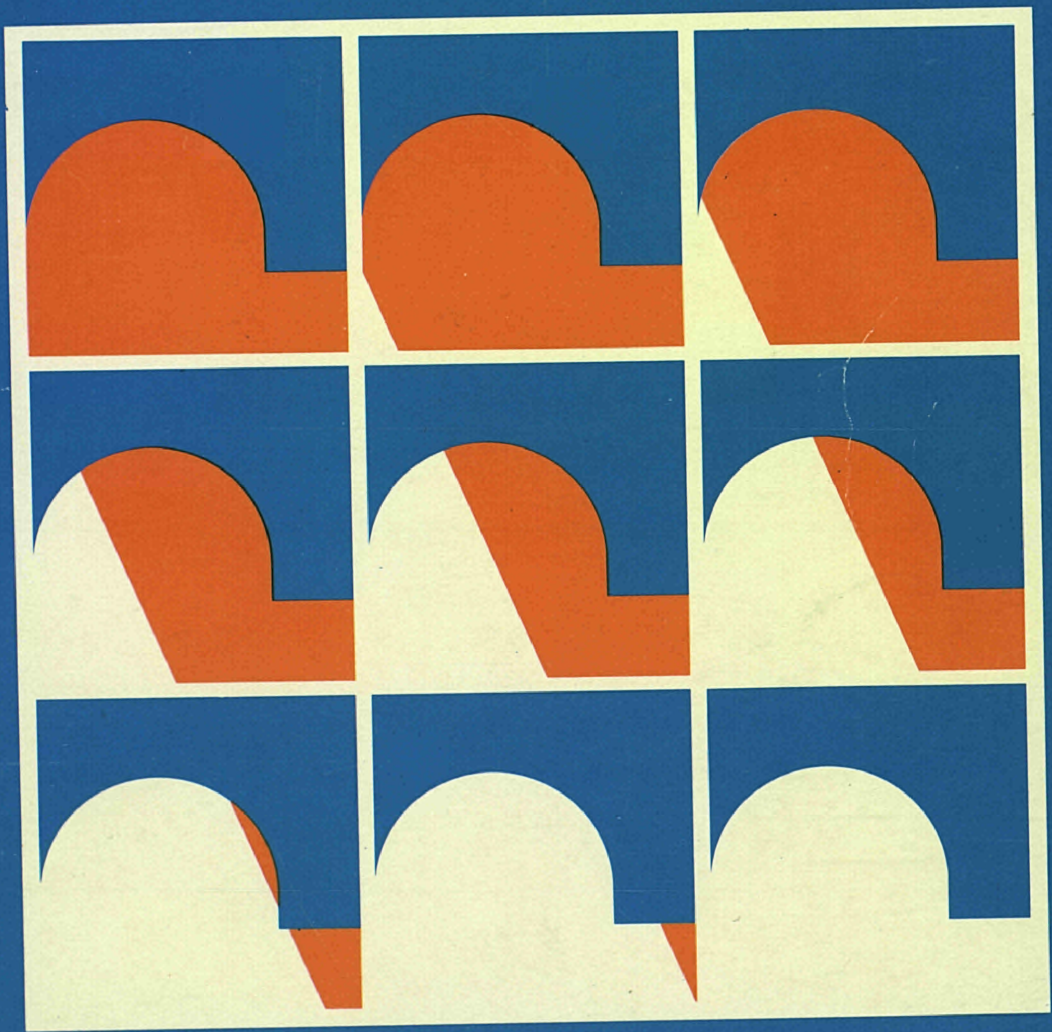
Commission of the European Communities

nuclear science and technology

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

Annual progress report 1991



Report

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FOREWORD

The following second Annual Progress Report summarises the activities of the European Communities R&D Programme on Decommissioning of Nuclear Installations for the year 1991. (For the first report s. Ref. /1/).

This programme was adopted by the EC Council in March 1989 /2/ 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.

This includes 2.3 million ECU voted by the European Parliament as additional funding for 1991, thus permitting a second call for research proposals /3/ for Area C (except pilot projects) of the programme. Of the 26 research proposals submitted to this call for proposals, 11 were selected; contract negotiations for these new projects were completed and most of them were signed before the end of 1991. As these and other contracts which become effective after July 1991 had not yet produced significant results, only the objectives, scope and work programme are reported in the following.

Work on the first phase of the four pilot dismantling projects was successfully completed and the respective contracts were amended to accommodate the work programme for the second phase. Also during 1991, Part B "Identification of guiding principles" was launched and a major action to collect data relevant to cost, occupational doses, working time and waste arisings was commissioned. Both activities are introduced briefly 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.

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/ 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.

/3/ 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.

CONTENTS

SECTION A: RESEARCH AND DEVELOPMENT PROJECTS — Page

SECTION B: IDENTIFICATION OF GUIDING PRINCIPLES

SECTION C: TESTING OF NEW TECHNIQUES IN PRACTICE

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, Postfach 90 01 60, D-W-6000 Frankfurt am Main 90
BNF	British Nuclear Fuels plc, Sellafield Works B403, UK-Seascale, Cumbria CA20 1PG
BS	Brenk Systemplanung, Heinrichsallee 38, D-W-5100 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-sur-Yvette
CEA-Valrhô	Commissariat à l'Energie atomique, Centre de la Vallée du Rhône, B.P. N° 171, F-30205 Bagnols-sur-Cèze Cedex
CIEMAT	Centro de Investigaciones Energéticas Medioambientales y Tecnológicas, Avenida Complutense 22, E-28040 Madrid
COGEMA	Compagnie 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-W-7000 Stuttgart 80
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	Société anonyme d'Explosifs et Produits chimiques, rue de la Dynamite, F-13310 Saint-Martin de Crau
FHGF	Fachhochschule Giessen-Friedberg, Wiesenstrasse 14, D-W-6300 Giessen
Framatome	Framatome, Tour Fiat Cedex 16, F-92084 Paris-la-Défense
Goodwin	Goodwin Air Plasma Ltd, Kernan Drive, UK-Loughborough, Leist. LE11 0JF

KA	Kraftanlagen Aktiengesellschaft, Im Breitspiel 7, Postfach 10 34 20, D-W-6900 Heidelberg
KEMA	N.V. Keuring van Elektrotechnische Materialen, Utrechtseweg 310, NL-6812 ET Arnhem
KfK	Kernforschungszentrum Karlsruhe, D-W-7514 Eggenstein-Leopoldshafen
KKWR	Kernkraftwerk Rheinsberg, D-O1955 Rheinsberg
KRB	Kernkraftwerk RWE-Bayernwerk GmbH, Postfach, D-W-8871 Gundremmingen
LAINSA	Limpiezas y Acondicionamientos Industriales S.A., El Payeter 13, E-46008 Valencia
NIS	NIS Ingenieurgesellschaft mbH, Donaustrasse 23, D-W-6450 Hanau
NNC	National Nuclear Corporation Ltd, Booths Hall, Chelford Rd, UK-Knutsford, Cheshire WA16 8QZ
Noell	Noell GmbH-Nuklear Service, Postfach 6260, D-W-8700 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-W-4300 Essen
RWTHA	Rheinisch-Westfälische Technische Hochschule Aachen, Reutershagweg 4, D-W-5100 Aachen
SCK/CEN	Studiecentrum voor Kernenergie/Centre d'Etudes de l'Energie Nucléaire, Boeretang 200, B-2400 Mol
SG	Siempelkamp Giesserei GmbH & Co, Siempelkampstr. 45, D-W-4150 Krefeld 1
Siemens-KWU	Siemens AG, Bereich Energieerzeugung KWU, Hammerbacherstraße 12-14, D-W-8520 Erlangen
Siemens-BEW	Siemens AG Brennelementewerk Hanau, Rodenbacher Chaussee 6, D-W-6450 Hanau
SSP	Stangenberg, Schnellenbach und Partner GmbH, Postfach 10 28 69, D-W-4630 Bochum 1
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-W-8000 München 21
TÜV-SWD	Technischer Überwachungsverein Südwestdeutschland e.V., Dudenstrasse 28, D-W-6800 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, PF 6009, D-W-3000 Hannover
VAK	Versuchsatomkraftwerk Kahl GmbH, Postfach 6, D-W-8756 Karlstein am Rhein

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, e.g. 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 shut-down 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 1991, 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:	J P HARTLEY, TEL	
	Phone: 44/81/575 47 23	Fax: 44/81/575 44 04

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. NNC Work programme

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

An inspection strategy was established and detailed plans for the inspection of reinforced concrete structures at Berkeley and Bradwell nuclear power stations have been prepared. The inspection and laboratory testing of concrete samples aims to provide the data required to assess the time taken to initiate reinforcement corrosion. Data from surveys undertaken in the preceding CEC decommissioning programme has been tabulated and detailed analysis of the data is to start shortly. Preliminary investigations of the impact echo and integrated optical fibre non destructive testing techniques has also been undertaken.

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.

The designs for Berkeley, Hunterston 'A' and Latina power stations which are typical of Magnox installation have been reviewed in order to establish that there is sufficient data and understanding to proceed with decommissioning when the end of their generating lifetimes are reached.

During the literature search information has been obtained from some 30 published papers relating to the design and decommissioning of similar plant.

Progress and Results

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

An inspection strategy was established. Reinforced concrete structures at two nuclear power stations; Bradwell that was inspected during the preceding CEC programme and Berkeley that was not inspected as part of the earlier CEC programmes; have been selected and detailed inspection plans were produced. The inspections and associated laboratory testing of concrete samples aim to yield the data required by predictive models that will be used to assess the life of the structures. The predictive models are designed to assess the time taken to initiate reinforcement corrosion attributable to chloride ions or to carbonation of the concrete.

2. Compilation of Systematic Database (B.1.3)

Data from surveys undertaken in the preceding CEC programme has been collated into a tabular format suitable for inputting into a database. The data has been initially classified as either a material property e.g. compressive strength (Mpa) or as a survey result e.g. depth of concrete cover (mm), depth of carbonation (mm) and level of chloride (% wt of sample) etc. Detailed analysis of the results is to start shortly.

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

The programme for the prestressing options aspect of the work has been delayed.

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

Initial investigations were undertaken into the equipment, working methods applications and capability of the impact echo and integrated optical fibre non-destructive testing techniques. These early investigations suggest that these techniques will be unable to clearly detect the presence of small flaws at the concrete steel interface in prestressed concrete structures.

5. Study of Parameters (B.2.2)

The investigations dealt with the long term behaviour of prestressed concrete pressure vessels (PCPV) during the post operational state. At first vessels with full operational levels of prestress and with the prestress losses associated with the post operational period were examined. Consideration was given to further reducing the level of prestress by relaxation of the prestressing steel. The PCPV of the Thorium High Temperature Reactor (THTR) Nuclear Power Plant Hamm-Uentrop which was depressurised after shut down in August 1990 was used as a reference vessel for these investigations.

The first task consisted of the compilation of the boundary conditions with regard to loads and material behaviour for the analyses of the long-term behaviour of this vessel in the post-operational state. These conditions were characterised by the absence of an internal pressure load and the reduction in the temperatures in the vessel walls.

The strain behaviour of PCPV in the post-operational state was determined from data provided by the continuous safety examinations of the THTR-PCPV. The comparison of calculated and measured concrete strains shows, that the concrete creep was overestimated by the assumed creep law (which was defined to be independent of concrete age), and that the applicability of this creep law in connection with the law of linear superposition especially after long-term release of internal pressure needs to be investigated in more detail.

6. Design Review (B.3.1)

Individual reactor designs have been reviewed and the state of operation established. Key items of plant have been identified for a more detailed assessment and the design of these items is being considered together with environmental factor with the object of identifying potential degradation mechanisms, such as the formation of nitric acid.

Berkeley reached the end of its generating life in March 1989 and is the lead station for decommissioning in the United Kingdom. Discharge of all fuel is scheduled for completion by mid 1992.

During discussions with representatives from Nuclear Electric and Scottish Nuclear it was explained that a three stage decommissioning programme is currently proposed in the UK, comprising:-

Stage 1: Removal of all fuel: (2-3 years)
Care and protective maintenance I (30 years)

Stage 2: Construction of a safe store (5 years)
Care and protective maintenance II (100 years)

Stage 3: Complete dismantlement and removal of all remaining plant and structures
Final clearance of site.

Data relating to the current condition, risk or nitric acid formation, corrosion resistance and rate of deterioration is being accumulated in order to assess possible loss of integrity for structural materials and components during the prolonged 'care and maintenance' stage following fuel removal.

7. Prediction of Nitric Acid Concentration (B.3.2)

During Stage I decommissioning, whilst fuel is being removed from the reactor, gamma doses will be relatively high. Early on during this stage a carbon dioxide based atmosphere will be maintained and the risk of formation of nitric acid will be negligible.

However, after a relatively short period of defuelling (i.e. months) air will be admitted and although activity levels will be decreasing it is anticipated that nitric acid vapour will be formed by gamma radiolysis of moist air. The gamma dose rates associated with selected key components will decay from the 'end of generation'. Knowing environmental details such as relative humidity, temperature, containment volume, flow etc, the rate of formation of nitric acid can be calculated to ensure that proposed handling/dismantling strategies will not be prejudiced by corrosion arising from conditions experienced during the decay stage.

Although by drying the air it is possible to minimise the level and concentration of nitric acid formed, it is known that nitric acid levels in excess of 100 vppm give rise to increased general corrosion rates and even lower levels are associated with the stress corrosion of unheat-treated carbon steel. Thus careful assessment of this area is necessary to ensure that optimum environmental control is defined.

8. Literature Survey (B.3.3)

A list of the published papers which have been consulted during this review is attached. Where appropriate reference has also been made to unlisted material which is confidential to the UK nuclear industry.

References

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/2/ Watston, J, & Macrae, W, Latina Power Station BNES journal, April 1964 Vol 3 No 2.

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2. <u>AREA No. 2</u>: 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

At the end of 1991, six research contracts relating to Area No. 2 were at the stage of execution.

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

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).

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 OBTAINED RESULTS

Summary of main issues

The design, calibration and testing of the prototype ultrasound transducer and generator has been completed.

Experimental tests both on non active and radioactive materials (coming from BWR-Garigliano) has been performed.

Progress and Results

1. Evaluation, selection and acquisition of special ultrasonic transducers (B.1.)

During the 1991 design, calibration and testing of the prototype ultrasound transducer and generator has been completed. Moreover tests with non-active AISI 304 pipe allowed to determine the optimum working condition (coupler diameter, ultrasound frequency and power) for subsequent decontamination tests.

For first tests the selected transducer is of the Langevine type (e.g. precompressed) and basically consists of a piezoelectric ceramic material disc forced between two metallic blocks; it is applied on the external surface of the pipe by a front conic coupler (sized on a 1/4 wavelength).

2. Decontamination Tests (B.2.)

2.1. Testing Materials

For initial testing AISI 304 material was used both non-active and radioactive. The external diameter and thickness are 48.3 mm and 3.7 mm respectively.

Radioactive material comes from BWR Garigliano and it belonged to an auxiliary pipe (draining line) of the primary system.

2.2. Test Devices (B.2.1.)

The experimental devices used for the testing with non-active pipe are schematically shown in Figures 1 and 2.

The experimental apparatus used for the testing with radioactive-material is shown in Figure 3.

In order to evaluate the effect of the velocity on the process efficiency tests have been performed also in the DECO-Loop.

2.3. Decontamination Tests on Contaminated Specimens (B.2.2.)

Preliminary laboratory tests on AISI 304 radioactive specimens have been performed in order to evaluate the effectiveness of the decontamination process both on straight pipes and on complex geometry samples.

Tests have been performed in flowing chemicals (HNO_3 5% + HF 1.5%) with the ultrasound transducer directly applied on the external surface of the tested material.

Results in Table I indicate that the decontamination process can reduce the residual activity of straight pipes well below the limit of 1 Bq/cm^2 with a uniform and minimal base metal removal (less than about 20 μm). Furthermore, results suggest that direct application of a transducer on a component surface can drastically reduce the residual activity in zones such as welds and dead legs where recontamination occurs if only flowing chemicals are used.

Decontamination results on a sample with a welded area are shown in Figure 4.

Table I: Results of the tests with radioactive AISI 304 pipes.

TEST N°	1 Static CH	2 Static CH+US	3 Flowing CH	4 Flowing CH+US	5 Flowing CH	6 Flowing CH+US
INITIAL ACTIVITY Bq/cm^2 (60Co)	8321.30	8853.59	585.11	736.71	6574.04	1589.79
RESIDUAL ACTIVITY						
Bq/cm^2 (60Co)	486.38	4.70	182.94	<0.1	2594.89	0.59
Bq/g (60Co)	251.77	2.43	109.83	<0.1	1768.55	0.22
DF	5.5	619.2	1.5	5758.0	1.2	2345.4
DE (%)	81.76	99.84	31.86	99.98	13.99	99.96
AV. PENETRATION (μm)	191.57	213.29	3.07	14.61	4.65	23.48

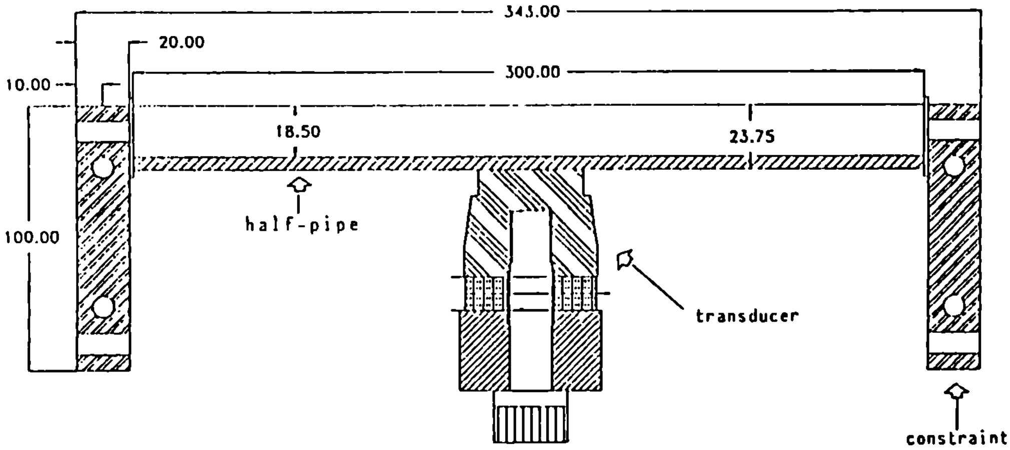


Figure 1 - Frontal view of the experimental devices used for the vibration amplitude tests, on half pipe.

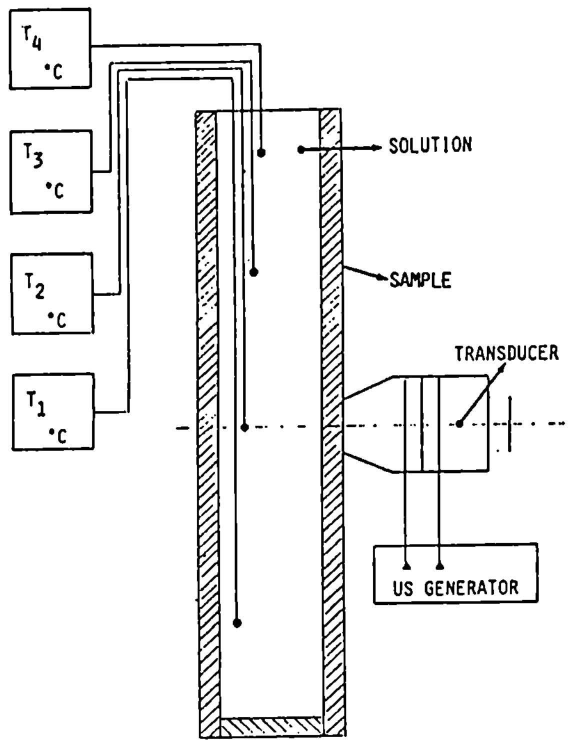


Figure 2 - Experimental device used for static test, on wole pipe.

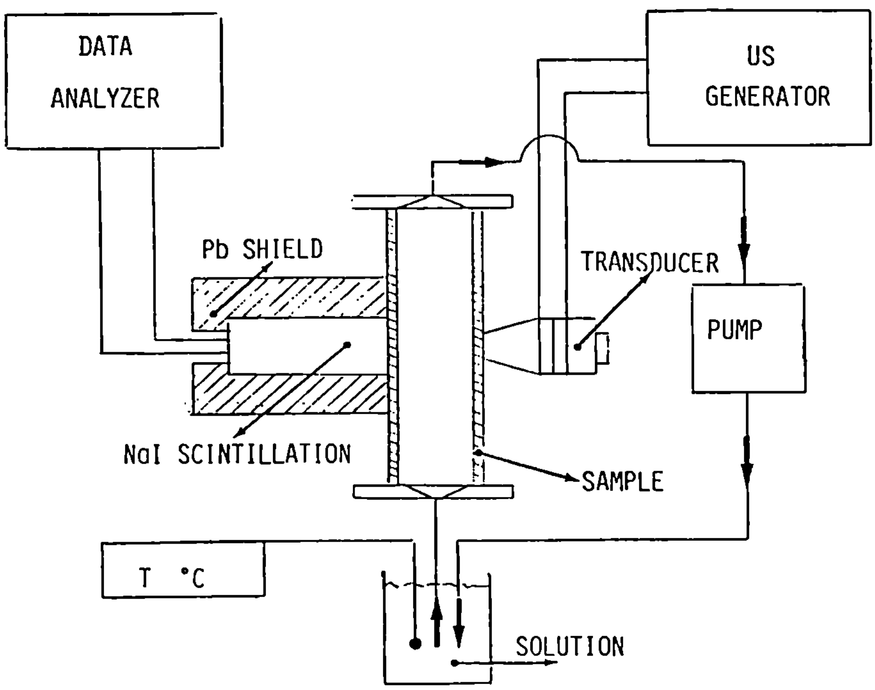
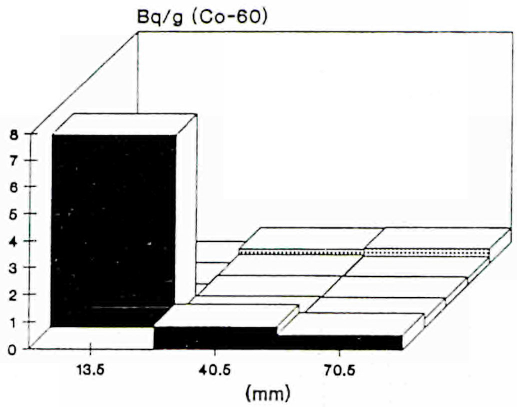
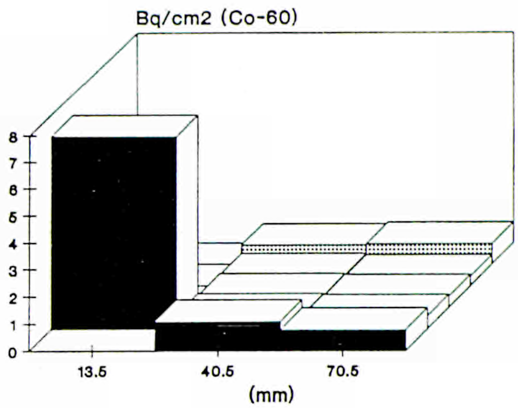


Figure 3 - Flow sheet of the experimental loop for tests with radioactive materials.





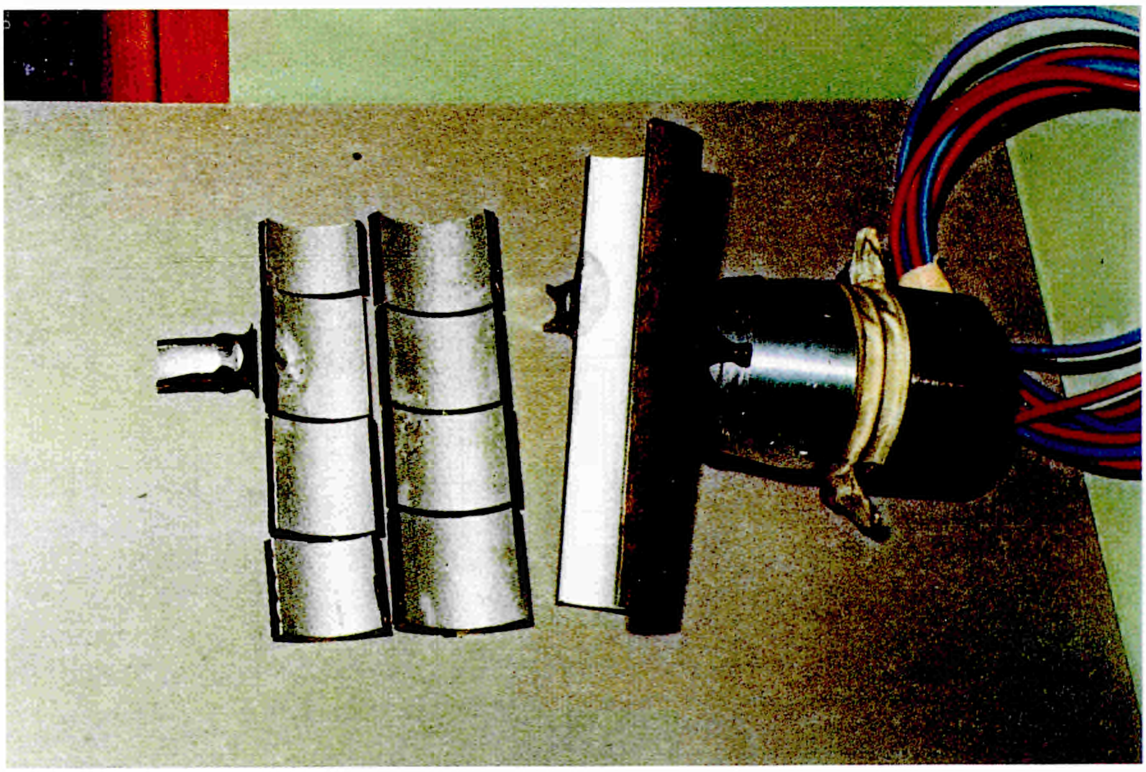


Figure 4 - Residual radioactivity distribution on sample n° 6 after decontamination.

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 OBTAINED RESULTS

Summary of Main Issues

In cyclovoltametric, potentiostatic and galvanostatic analyses, it was possible to verify the assumptions as to the anodic dissolution mechanism of potassium bromide (KBr) and potassium chloride (KCl). The dissolution mechanism when using the potassium fluoride electrolyte and glycolelectrolytes still remains to be investigated and clarified.

The replacement of KBr by KF as conductive salt showed positive results. Satisfactory removal rates along with a good anode current yield could be achieved.

The electrolyte along with KF as conductive salt demonstrated that the scattering 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 noted on the electrode.

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

Progress and Results

1. Quantitative Analyses as to the Dissolution Mechanism (B1)

When the potassium halogenides in water/water along with acetyl-acetone/ water along with isopropanol, and the overall electrolyte composition underwent cyclovoltametric analyses, it became evident that when adding isopropanol, this does not have any effect on the anodic behaviour of conductive salt, water and acetylaceton.

Moreover, the assumptions as to the anodic dissolution mechanism when using potassium bromide and potassium chloride were verified.

The potentiostatic analyses showed that when using potassium chloride and potassium bromide, distinct pitting was noted on the electrodes.

In the tests undertaken accordingly by us, micropitting was noted in the case of the potassium fluoride electrolyte.

In galvanostatic analyses it became evident that only with a KF electrolyte the so-called micropitting was noted which actually brings about the desired anodic removal. This result was also verified by subsequent tests. A visual check was made with a scanning electron microscope (SEM). With the KBr and KCl electrolyte deep pitting was noted.

In the range of 0.1 to 0.01 mol buffer per dm^3, the SEM images did not reveal a distinct effect of the phosphate buffer.

Moreover, it still remains open whether this low impact is a result of the buffer effect or of an inhibitory effect.

Due to the fact that the dissolution mechanism of the KF electrolyte is not yet exactly defined, further investigations and analyses are required. It might be possible that the metal and/or the respective ions are directly involved in the penetration reaction.

All investigations and analyses for conductive salts are repeated with the glycol electrolyte, and with the obtained results being then compared.

2. Optimization of the Aqueous Electrolyte by Replacing the Potassium Bromide by Other Conductive Salts (B2)

Out of many possible conductive salts and conductive salt compounds, potassium fluoride proved to be best suited to replace potassium bromide.

As shown in fig. 1, no pitting at all was noted when using potassium fluoride. One gets a uniform and smooth surface.

The recommended test parameters are:

Temperature:	40	°C
Current density:	10	A/dm^2
Voltage:	ca. 10	V
Electrode distance:	3 - 5	cm

Under these conditions it is possible to obtain with a smooth surface:

Current efficiency:	$\geq$ 84	%
Removal rate:	$\geq$ 0.9	microns/min
Removal efficiency:	$\geq$ 80	%

In a test extending over 16 hours, it was not possible to note any pitting, and this means that the service life is satisfactory.

3. Investigations of Scattering Behaviour and the Effects on Removal Efficiency, Surface Quality and Decontamination Factors (B 3)

The tests with the new aqueous electrolyte clearly demonstrated that the scattering power directly depends on the viscosity, and also on the applied current density. For example with an average viscosity of 12.2 m Pa*s, the maximum scattering power was attained, whereas with electrolytes of a higher or lower viscosity, the attained maximum efficiencies were at 64.4 % and 62.3 % respectively. Fig. 2 shows the scattering power in dependence of the current density.

A similar behaviour was also noted with a varying current density. Here too, it was noted that in all electrolytes the maximum was reached with a constant viscosity.

Fig. 3 shows the scattering power as a function of the viscosity. The below specified conditions were maintained:

Current density:	13 - 25	A/dm^2
Electrode surface:	10	cm^2
Waterbath temperature:	40	°C
Electrolysis duration:	30	minutes
Electrode distance:	3; 15	cm
Voltage:	13	V

In all tests the attained removal efficiencies ranged between 80 and 95 % !

4. Development of a Non-Aqueous Electrolyte (B 4)

Out of a great variety of organic solvents, glycol proved to be best suited to be used instead of water.

When using potassium fluoride as conductive salt, current efficiencies of 87 % were attained and removal efficiencies of 57 %. However, distinct pitting was noted, and that is why this electrolyte was not further examined.

When using potassium bromide as conductive salt, efficiencies of 87 - 95 % were attained, removal efficiencies of 84 - 93 %, and removal rates of 1.2 microns/min. No pitting was noted on the electrode.

The organic electrolyte underwent testing with some test parameters being modified accordingly.

The electrode surface quality worsened with an increasing isopropanol portion. With a low isopropanol portion, the surfaces shine and do not show any pitting.

Along with a reduced conductive salt concentration, the surface quality became worse.

Smooth and shining surfaces were noted as long as the electrolyte was new and the duration of the electrolysis rather short. After a longer electrolyte service life, however, a growing area with a dull surface was noted.

A water content of 2.8 % proved to be harmless. It is even possible to have a water content of 8 %, at maximum.

Water concentrations of 2.8 - 11 % increased the anode current efficiency from 70 to 77 %, whereas the removal efficiency went down from 88 to 66 %, and the removal rate went down from 1.77 to 1.06 microns/min.

The aqueous electrolyte along with potassium fluoride as conductive salt was found to be not suitable for pure iron materials.

For stainless steel, the previously stated good removal properties were actually verified. There were attained current efficiencies of 80 - 90 %, and removal rates of 1.7 - 1.9 microns/min.

The organic KBr/glycol electrolyte is suitable for all materials. The obtained surface qualities are comparable to those obtained so far in the tests.

5. Decontamination tests with contaminated samples (B 5)

Iron-59 with perfectly adhering surfaces and activities of 3000 - 6400 Bq/dm^2 were applied to stainless steel electrodes under the conditions specified below.

- Electrolyte:
  75 g $FeCl_2$ 4 H_2O
  105 g $FeCl_2$ 2 H_2O
  250 g H_2O

- Conditions:

Current	50	mA
Electrode surface	26	cm^2
Electrode distance	3	cm
Temperature	25 - 30	°C
pH value	0.7 - 1.5	
Current density	0.19	A/cm^2

- Pretreatment:
  Surface sandblasted, cleaned with water, and pickling for at least 5 minutes in 16 %-HCl.

The aqueous electrolyte with KF and the organic electrolyte with KBr as conductive salts were used in the decontamination tests.

It was possible to attain complete decontamination of all plates.

With aqueous electrolytes current efficiencies of 80 - 95 % were attained along with removal rates of 1.7 - 1.85 microns/min.

The values obtained for organic electrolytes were at 50 - 77 % current efficiency and a removal rate of 1.7 microns/min.

After decontamination and with the metal acetylacetone having been aspirated, the activities listed below were measured in the remaining electrolytic solution:

H_2O dist.	0.4	Bq/ml 'blind value'
KF/H_2O filtrate	1.1	Bq/ml
KBr/glycol filtrate	6.1	Bq/ml
$FeCl_2$ solution	213.9	Bq/ml
Flushing solution	1.3	Bq/ml

The values are negligible, i.e. the activity is primarily in the sediment.

6. Electrolyte Disposal (B 6)

No results are as yet available as far as the electrolyte disposal is concerned.

Fig. 1: Electrode No. 7
conductive salt potassium fluoride (KF)

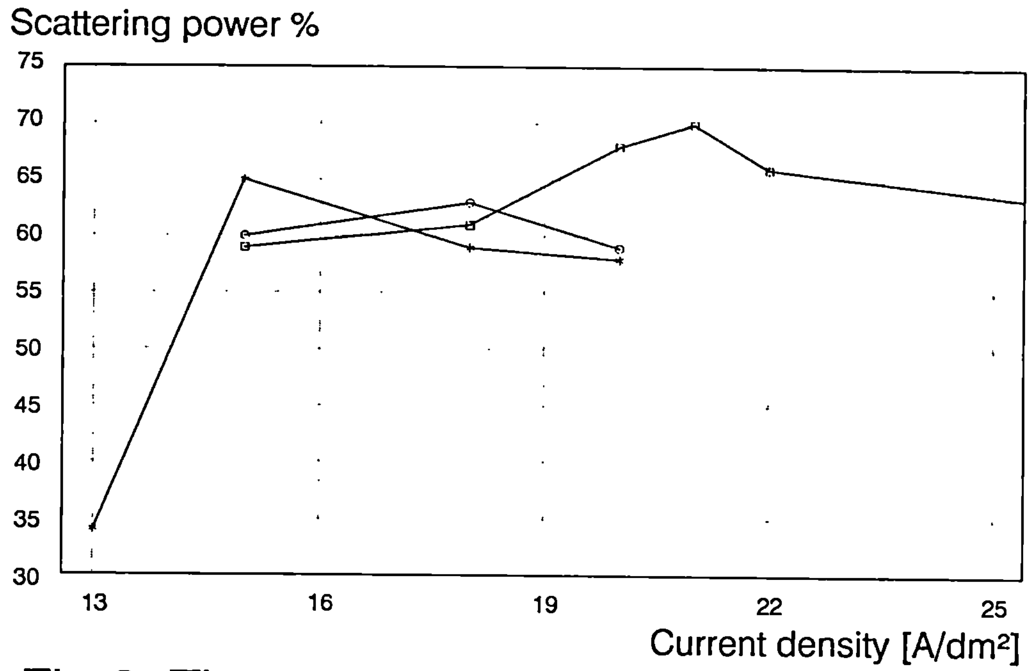


Fig.2: The scattering power as a function of the current dencity

○ Viscosity of the standard electrolyte
□ Viscosity 12,2 mPa s
∗ Viscosity 19,9 mPa s

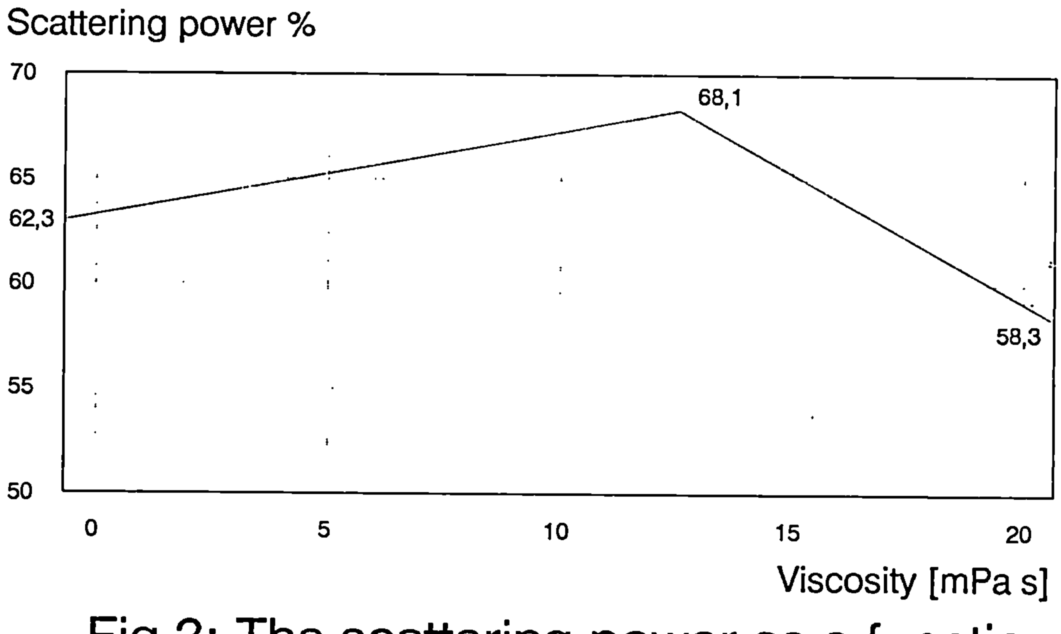


Fig.3: The scattering power as a function of the viscosity

Current density = 20 A/dm²

2.3. MICROWAVE SYSTEM TO SCARIFY CONCRETE SURFACES

Contractor: ENEA, Casaccia
Contract No.: FI2D-0024
Work Period: January 1991 - December 1992
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 obtained results

Summary of main issues

The design and the construction of the prototype microwave system including the support structure and the suction unit have been carried out. The design work has been performed by Sistemi e Tecnologie per l'Energia (ST), the construction of the microwave system has been executed by Alter and that of the mobile support structure by Sacchi; Nilfisk has supplied the suction and filtration unit. The complete apparatus has been assembled at Alter's workshop and submitted to acceptance tests.

At the Trisaia Research Centre, where the apparatus will be tested, two reinforced concrete platforms have been built for the cold tests.

Progress and results

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

The microwave system consists of three 5.5kW magnetrons generating 2450MHz microwaves; the system is equipped with rigid wave guides, irradiation heads, isolators to protect the magnetrons from reflected waves, transformers and ventilators (see Figure 1).

The microwave system, except for the lower part of the wave guides, is enclosed in a sealed stainless steel casing with inlets for electric cables and cooling water tubes.

The power supply unit and the control unit are located at a distance of 20m or more from the microwave system and are connected to it by means of flexible cables; flexible piping carries the cooling water from the plant mains to the casing to cool the magnetrons. The control system is capable of varying (between 0 and 100%) the operating power of each magnetron and of indicating this power and also that reflected from the surface being scarified.

The irradiation heads, which are protected by a teflon flange permeable to radiation but not to dust, are surrounded by a suction bell connected to the rigid wave guides by means of a vertically adjustable collar. The outer edge of the bell is equipped with a silicon rubber skirt that minimises the inlet of air and the dispersal of radiation. The bell is connected by means of a flexible vacuum hose to the suction system.

This system (see Figure 2) consists of a 220l drum in which most of the debris from the scarified surface are separated and of a main container downstream of the drum with two filtering systems, one capable of separating 99,5% of 2μ powders and one with absolute filters. A vacuum fan is installed on the top of the main container and moves the debris from the suction bell to the containers. These are mounted on casters which allow them to follow the movements of the microwave system when necessary.

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

The support structure (see Figure 3) consists of a 1300x2600x600mm frame in carbon steel coated with decontaminable paint. The double pair of guide rails of the frame allow the carriage with the microwave system to move backwards and forwards longitudinally for a length of travel of 1500mm at a constant rate variable between 1 and 5mm/sec; the movement is obtained by means of two chains operated by a motor bolted on the frame and remotely controlled from the control unit.

The distance of the irradiation heads from the surface to be scarified, the standoff, may vary from 15 to 40mm and is obtained by means of four screws connecting the microwave casing to the carriage; when the required standoff has been obtained the casing is fixed to the carriage with set screws, and by means of the adjustable collar the suction bell's skirt is brought in contact with the surface to scarify.

Due to eight idle wheels running on the double pair of rails, the carriage in its various positions on the structure will always be at the same distance from the wall or floor.

Translation along the third axis by means of wheels connected to the four corners of the structure is obtained by hand or in hot conditions with manipulators. Four screws at the corners allow the support structure to be fixed in the desired position when this is reached.

At the Trisaia Centre two reinforced concrete platforms have been built that will be used for the cold tests. Their dimensions are: 3000x2500x170mm. The two platforms are made, one with 325kg/cm^2 (ultimate compressive stress), the other with 425kg/cm^2 cement and are reinforced with steel bars of different sizes; the top and bottom sides of each platform, divided longitudinally, are equipped respectively with 24 and 12mm and with 8mm rebars: one part of the bottom side has no rebars. It will be thus possible to perform scarification tests with different types of cement and steel configurations using both sides of the platforms.

3. <u>Trial operations on a non-radioactive concrete wall</u> (B.3.)

The different parts of the equipment described above have been assembled at the manufacture's plant and have undergone acceptance tests.

The tests comprised visual inspections, dimensional checks and functional tests of the equipment. Subsequently three scarification tests were performed with results considered satisfactory on a preliminary basis; the surface layer of the reinforced concrete was shattered and the debris were transferred to the separation unit.

The system will be shipped to the Trisaia Centre in the first week in February 1992.

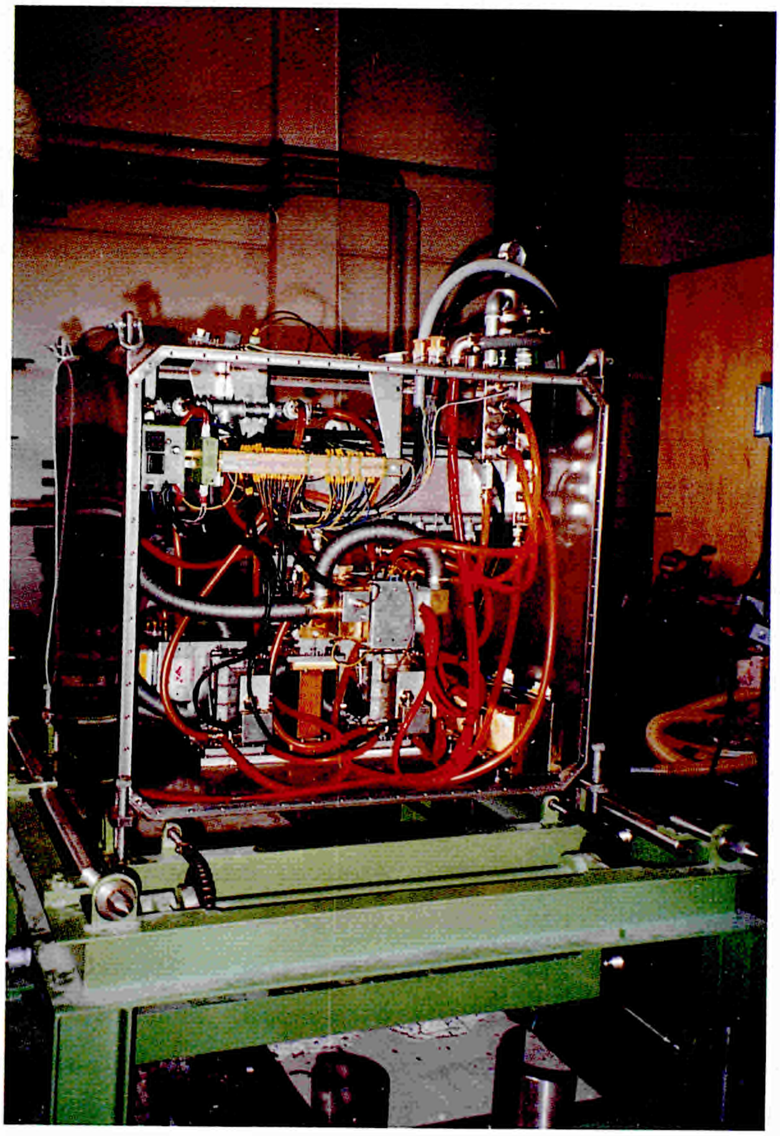


Figure 1: Microwave system

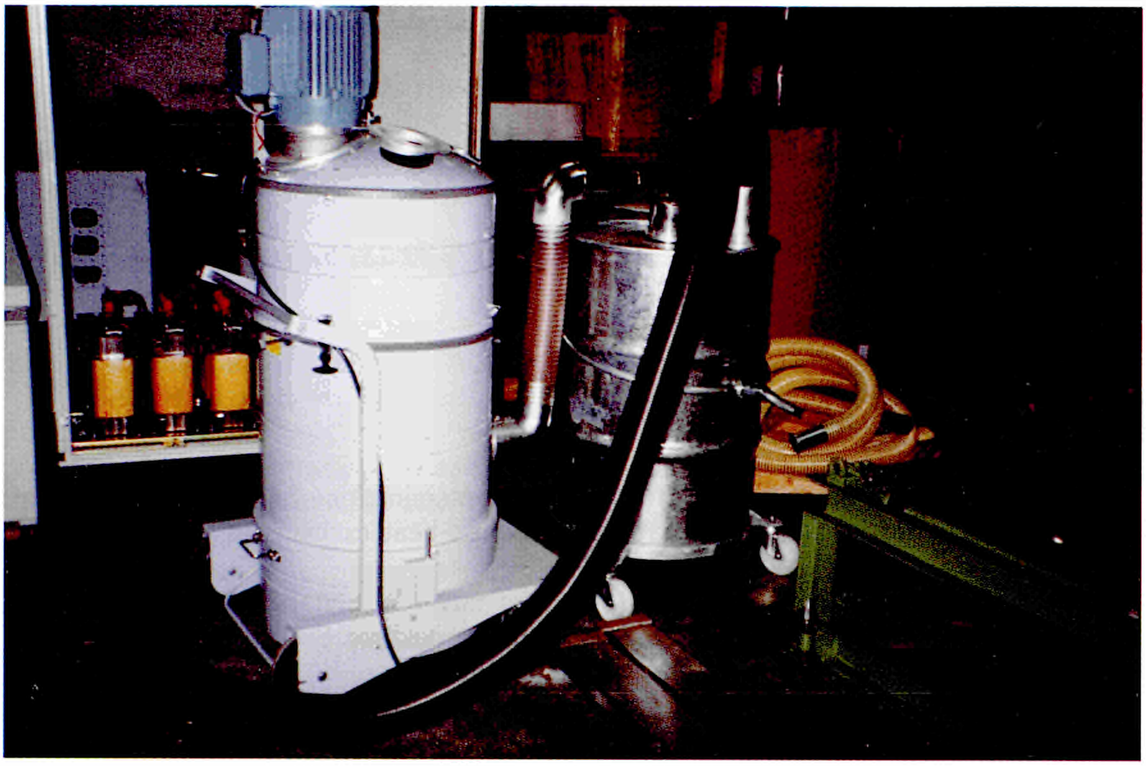


Figure 2: Suction apparatus

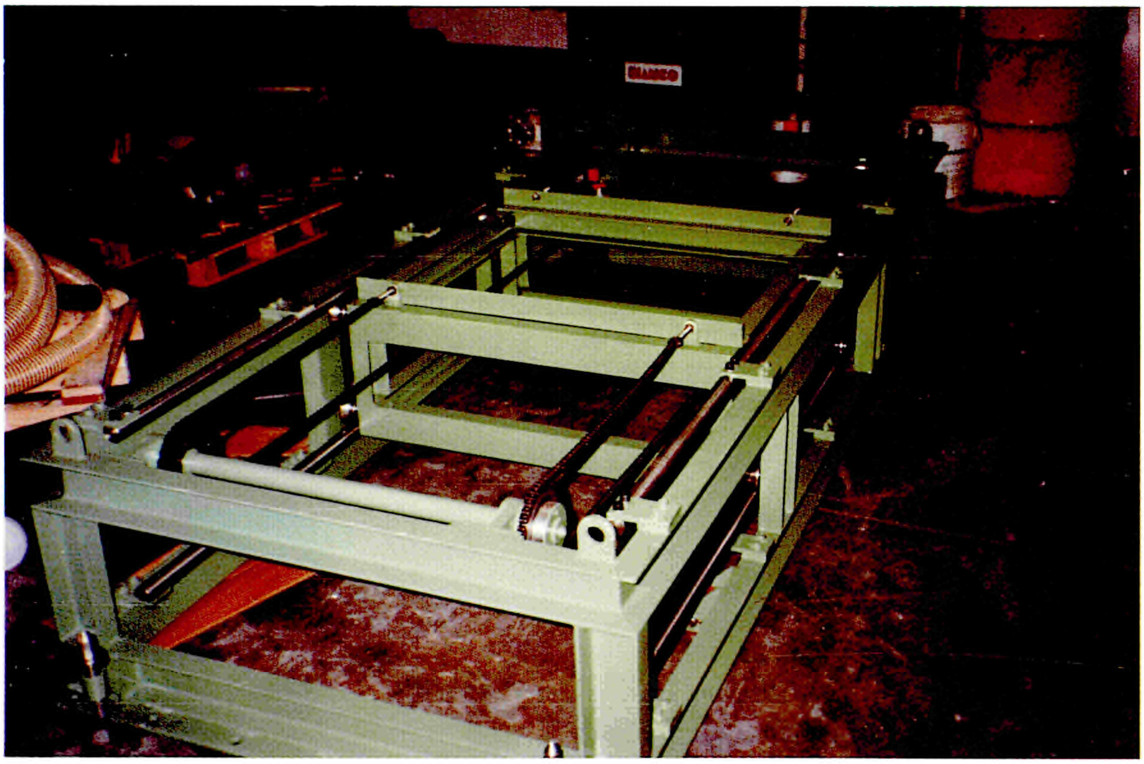


Figure 3: Support structure

2.4. DECONTAMINATION OF LARGE-VOLUME NUCLEAR COMPONENTS USING FOAMS

Contractors: CEA-Cad, AEA Winf
Contract No.: FI2D-0035
Work Period: October 1990 - March 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

The first specimens taken in 1991 from the carbon steel deaerator storage vessel of the Winfrith SGHWR (Steam Generating Heavy Water Reactor) allowed:

- characterization of the oxide deposit and its contamination,
- selection of the reagents with which the component could be treated and specification of their formulation,
- application of foam decontamination at laboratory level.

Progress and results

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

The first analyses carried out at Winfrith on the component in ferritic steel revealed the presence of an oxide layer (150 to 350 μm) made of 40% haematite and 60% magnetite. The presence of 10% chromium oxide and 5% nickel oxide is also noted in the surface oxide layer (5 to 10 μm). The radiochemical activity of the oxide layer is due to Co-60 (420 KBq/m^2) and to Mn-54 (2 KBq/m^2).

The selection of reagents was made at Cadarache following a series of tests carried out on test specimens from Winfrith in a liquid medium.

The treatment which proved the most effective is applied at ambient temperature in two stages:

- a 2h degreasing process using 3 mol/l soda followed by rinsing,
- an oxide etching process by applying a 2 mol/l sulfuric acid solution for a period of 4 to 5 h followed by thorough rinsing.

In these conditions, the residual activity obtained varies from 0.5 to 1.5 Bq/cm^2.

In order to apply these reagents in a foam-like medium, a formulation that was initially developed for the decontamination of G2 gas-graphite reactor valves was employed. For both decontamination stages a mixture of two surface-reactive agents was used:

- a sulfobetaine (Amonyl 675B) which is an amphoteric surface-active agent,
- a glycoside alkyd-ether (Oramix CG 110-60, which is a non-ionic surface-active agent).

A destabilizing reagent, pentanol 2 in a soda medium, is also used, allowing control of the foam's humidity and draining of the liquid in the film.

In sulfuroxide medium (dehydrating acid), the last mentioned alcohol had to be replaced by methyl 4 pentanol 2.

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

The formulations employed were tested with the help of a pilot test including:

- a mixer (capacity 6 l),
- a static foam generator,
- a decontamination reactor with a capacity of 30 l and height of 1 m.

The operating conditions imposed for these tests were the following:

Degreasing stage:

	soda: 3 mol/l
air flow: 180 l/h	Amonyl 675 B: 0.30%
liquid flow: 16.4 l/h	Oramix CG 110-60: 0.40%
	pentanol 2: 0.40%

Etching stage:

	H_2SO_4: 2 mol/l
air flow: 180 l/h	Amonyl 675 B: 0.45%
liquid flow: 13.8 l/h	Oramix CG 110-60: 0.6%
	methyl 4 pentanol 2: 0.6%

The results obtained on the Winfrith test samples are grouped in Table I and allow the following remarks to be noted:

- decontamination by foam loaded with chemical reagents requires a longer treatment period than by a liquid medium (8 instead of 4 to 5 h);
- it is easier to embrittle the oxide in one step (test No. 8) than to carry out the rinsing over several steps (test No. 9).

Note: during tests 8 and 9, only 3.5% of activity displaced from the test specimen was in solution, the remaining oxide having been recovered on the filter.

Table I: Decontamination of test specimens from the Winfrith deaerator using foams, with nitrogen as gas vector

Test No.	Foam characteristics:		Time (h)	Radioactivity (Bq/cm^2)		Decontamination factor
	Reagent	Concentration (mol/l)		Initial	Residual	
7	H_2SO_4	2	0 8	46	 5.3	 8.7
8	NaOH then H_2SO_4	3 2	0 2 0 8	53 45	 45 $\leq$ 0.6	 1.2 $\geq$ 88
9	NaOH then H_2SO_4	3 2	0 2 0 4 6 8	80 73	 78 17 3 1.6	 1.0 4.7 26.7 50

3. Small pilot tests (B.3.)

Studies on a vessel with a capacity of 2.5 m^3 have been undertaken, with the aim of demonstrating the decontamination of test pieces, taken from the Winfrith Deaerator Storage Vessel, using rotating nozzles to spray the foam.

4. Secondary wastes treatment (B.4.)

Tests have been carried out at Winfrith, to investigate the destruction of the surface-active agents and alcohols present using the WETOX process (catalytic oxidation using H_2O_2) and the Ag II oxidation process.

Initial results indicate that both processes will effectively destroy the organic components. For example, in one test using the Ag II process, the total organic content was reduced from 8240 to 230 ppm.

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 - December 1992
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

Summary of main issues

An extensive bibliographic research on decommissioning, namely on chemical decontamination requirements , has allowed the state-of-art in this field to be defined.

On this basis and using historical data collected during the operations of the evaporator plant, some hypotheses on the characteristics of contamination layer were determined, in order to define the details of experimental work to be performed .

A cold mock-up reproducing main characteristics of the actual EUREX evaporator was designed;it has been completely assembled and located in an area of the EUREX plant in order to simulate as far as possible the real situation, utilising the same instrumentation

The main characteristics of the contamination and the performances of decontamination processes were investigated running cold and hot laboratory tests with specimens .

The tests were initiated later than scheduled, due to an administrative delay with the subcontract with Torino University ; for the future there should be no other changes in the schedule .

Progress and results

1.Theoretical studies (B.1.)

For the main characteristics of the evaporator see Table I . For the characteristics of Uranium and Plutonium solutions that were obtained in the tank during reprocessing campaigns see Table III.

A survey on technical literature in the field Ref./1,2/, showed that a number of chemical decontamination procedures has been extensively applied for removal of contaminants presents in a reprocessing plant (mainly Uranium and Plutonium) .

The Plutonium contamination , coming from the action of strongly nitric acid solutions and resulting in a diffused corrosion of the metallic surface /Ref.3,4/,is probably present as an oxide layer.However the chemico-physical interactions between the oxide layer and metal can be strongly affected by the thermal heating of the evaporator mantle ; moreover , highly localized Plutonium contamination can be present .

In addition to the classical chemical treatment , extensively described elsewhere , /Ref.1,2/ ,mainly based on acidic or redox processes , it was also envisaged to investigate a novel approach based on a new class of chemical decontamination agents , micellar aggregates / Ref.5,6,7,8/ coupled with specific extractants and dispersed in a gas-liquid phase .

The classical actinide extractants , such as TriButhylPhosphate(TBP),TriOctylPhosphineOxide(TOPO),Ac.EthylendiaminoTetracetic(EDTA),Di2(EthylHexyl)Phosphoric acid

(DEHPA) can be solubilized in an aqueous medium in two forms:

-aspecifically coupled in the surface region of non reactive (ionic or not) surfactant aggregates

-directly chemical bonded with alkylic chains ,which permit then to form greater aggregates with a higher concentration of active sites.

Some important factors for evaluating the performances of such micellar reagents are briefly reported:

-the extractants can be vehiculated in aqueous media .

-there is no need of organic solvents as complexing agents.

-the resulting wastes can be reduced in volume using these reagents as foam .Moreover an additional concentration can be obtained in two ways:

°thermal treatment (already described for analytical studies) /Ref.9,10,11/.

°ultrafiltration (described for wastes treatment) Ref./12,13,14/

- the surfactants , that potentially are not compatible with cementification of solid wastes,can be diluted or des troyed by means of:

°a chemical treatment (acetone or ethanol)/ Ref.7/

°a selection of appropriate hydrolizable surfactants /Ref.15,16 /

°a bacterial treatment / Ref.17 /

2.Laboratory tests (B.2.)

2.1. Selection of the reagents

Because it is not possible to perform any measurements on the type and degree of contamination present in the actual evaporator , a need was envisaged for a set of experimental tests in order to assess the contamination/corrosion mechanism and the performances of the proposed decontamination procedures .

The tests were performed utilising ring specimens made of INOX or EPE 21 , the alloy used for C-603 evaporator ; for their composition see Table III .

2.2 . Contamination

A first series of tests was conducted utilizing Uranium and eventually Cerium solutions (to simulate Plutonium) as contaminants , subsequently some ring specimens were treated directly with Plutonium solutions .

2.3 . Decontamination

The experimental tests are not yet complete and will be presented in the next Report .

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

3.1.Design and manifacture of a mock-up (B.3.1.)

The mock-up reproducing main characteristics of C-603 has been designed,manufactured and installed. It has been provided with the same instrumentation used for C-603. .

The material is stainless-steel with a glass window.

3.2.Simulation tests (B.3.2.)

The facility was tested, and operation will start in January 1992.

The contamination/decontamination tests should be runned in connections with results coming from laboratory tests .

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TABLE I - C-603 Evaporator characteristics

Type	Thermosyphon
Volume	13 l
Thermic exchange surface	0.3 m²
Connections	Welded

TABLE II - Final products of EUREX reprocessing campaigns

CAMPAIGN No.	U (g/l)	Pu (g/l)	H+ (M)	D (g/cm³)	NH_2OH (M)	CONCENTRATION FACTOR
No. 1 Evaporator feed	16.5	traces	0.2	1.03		17
Final products	280	traces	2.0	1.50		
No. 2 Evaporator feed	5.2	traces	0.5	1.05		35
Final products	180	traces	4.6	1.35		
No. 3 Evaporator feed	-	4.5	0.1	1.00	0.2	22
Final products	-	100	3.0	1.24	0.2	

TABLE III - Alloy composition (%)

C	Si	Cu	Fe	Mn	Cr
0.01	0.15	1.94	32	0.25	21
Ti	Al	Ce	Mo	Ni	S
0.89	0.12	0.10	3.06	40.04	

2.6. DECONTAMINATION TECHNIQUE USING A DISPERSED CHEMICAL AGENT

Contractors: BATTELLE INSTITUT Frankfurt
Contract No.: FI2D-0054
Work Period: September 1991 - December 1992
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 obtained results

Summary of main issues

In the period September to December 1991 working phase B.1. has been performed. B.1. covered lay out, construction and testing of an experimental set-up, capable of producing and depositing an ultrafine fog of a chemical agent on a target surface /1/, which consists in a first approach of a non-active sample. The system contains a closed loop, where fine droplets of an etching fluid are generated in the 2 μm size range via an ultrasonic transducer and deposited electrostatically on a metal target after passing a corona discharge section. The metal target consists of a rotating endless belt which faces the aerosol generator and a wiper on the back of the belt to collect the waste liquid for postinvestigation. To test the functions of the apparatus and to determine typical process parameters first experiments have been carried out successfully. The apparatus available now allows a direct automatic on line measurement of the reaction kinetics of the etching process. The system is ready for the planned experiments with austenitic steel to optimize the process with non radioactive samples in B.3. and to demonstrate the method using radioactive samples in B.4. determining the attainable decontamination factors.

Progress and results

1. Construction and testing of the experimental set-up (B.1.)

The apparatus to be developed has to fulfill several tasks :

a) An etching fluid has to be atomized
b) The generated fog has to be deposited uniformly on the target to form a thin film
c) After reaction of the target surface with the etching fluid to a metal salt the remaining etching fluid needs to be removed
d) There has to be a control of the amount of etched material

The main objective of the project is to demonstrate the attainable decontamination factors using a chemical agent dispersed as fog applied to radioactive samples. Therefore the tasks a)-d) are planned to be carried out within a compact closed system which can be easily handled in a glove box. In the first step the apparatus necessary for these experiments was planned and constructed. First tests with non radioactive samples have been performed.

The experimental set-up for metal etching experiments is shown schematically in Fig. 1 and as a photograph in Fig. 2. It contains an aerosol generator which consists of a glass container (1) with flat bottom, to which a 1.75 MHz piezoelectric transducer /2/ (3) with 20 mm diameter is epoxy connected from the outside for protection against the etching fluid. The thickness of the glass container bottom is identical with the wave length in glass at the operation frequency of 1.75 MHz. Therefore the wall is transparent for the ultrasonic power. The liquid level of the etching fluid container inside the aerosol generation tank is maintained at an optimized level around 5 cm by a mechanical level monitor (2). Liquid is continuously delivered from the reservoir bottle via teflon tubing. A liquid geyser forms fine droplets under the action of ultrasonic radiation pressure and resonance vibrations of microscopic gas bubbles. Fig. 3 shows a typical cloud formed by the "geyser atomizer". The typical parameters of the liquid aerosol formed are a 2 μm mass mean diameter with a typical narrow geometric standard deviation of about 1.3-1.5 assuming a log normal distribution. The generated aerosol fills the top part of the aerosol generator and is blown out by a teflon coated membrane pump (5). With the optimized air flow rate of about 25 l/min, the aerosol liquid mass output is about 3.5 ml/min etching fluid which results in a droplet number concentration of approximately $0.5 \cdot 10^8$ droplets/cm^3. The aerosol passes a 15 mm diameter horizontal teflon tube and enters a vertical glass tube which contains a high voltage power supply (7) in its upper part which delivers up to 20 kV to a corona tip, a metal needle with Pt coating at its tip and teflon coating at its periphery for protection against the etching fluid. The tip of the corona electrode emits electrons by field emission and ionizes the surrounding air. Negative gas ions are repulsed from the tip while positive ions are attracted. The ion cloud around the corona tip (8,9) when passed by the aerosol will result in a negatively charged aerosol. A grounded target faces the corona tip via a "bottom window" in the aerosol deposition tank (6). The distance between the corona tip and the metal target can be adjusted to ensure that the charged aerosol is uniformly deposited onto the target surface over the full window opening of 60 x 100 mm. The metal target consists of an endless, motor driven (10, 11) metal belt which rotates continuously over two polyethylene coated metal cylinders. An acid resistant rubber wiper (12) at the back of the belt removes and collects (13) the remaining etching fluid after a selectable reaction time which is typically between 10 and 50 seconds. This liquid wiping proved to be sufficient to remove the etching fluid and the formed metal salt from the belt surface without the necessity of an additional washing liquid. By measuring the pH-value of the etching fluid prior to and after the etching process the reaction kinetics of the metal/acid combinations can be controlled on-line to minimize the required etching fluid.

2. First tests with non radioactive metal samples for process verification (B.1. B.3.)
By changing the aerosol output rate (< 3.5 ml/min), the gas flow (< 25 l/min) and the target belt speed (5 cm/min <v< 50 cm/min), means the reaction time, a wide range of process parameter variations has been used to assess the performance characteristics of the apparatus. These parameters are set by means of the control unit (4). As a test material for the calibration and assessment of the apparatus several 70 mm wide 100 µm thick copper foils were treated by film etching with HNO_3 at concentrations between 2-40 % which corresponds to a molarity range between 0.3 and 6.

The mass of the removed metal obtained using different process parameters was checked by weighing the target belt prior to and after the etching process. In addition these results were cross checked via pH-value control measurements. The stoichiometric and reaction kinetics measurements were also repeated by bath etching using HNO_3 concentrations of 2-60 %. The achieved agreement between film and bath etching was satisfactory.

For the planned etching experiments in B.3. with CrNi steel the inner parts of the glass apparatus will have to be protected against hydrofluoric acid (HF) with an ethylene vinylacetat copolymer coating. The specific technology has already been tested and proved.

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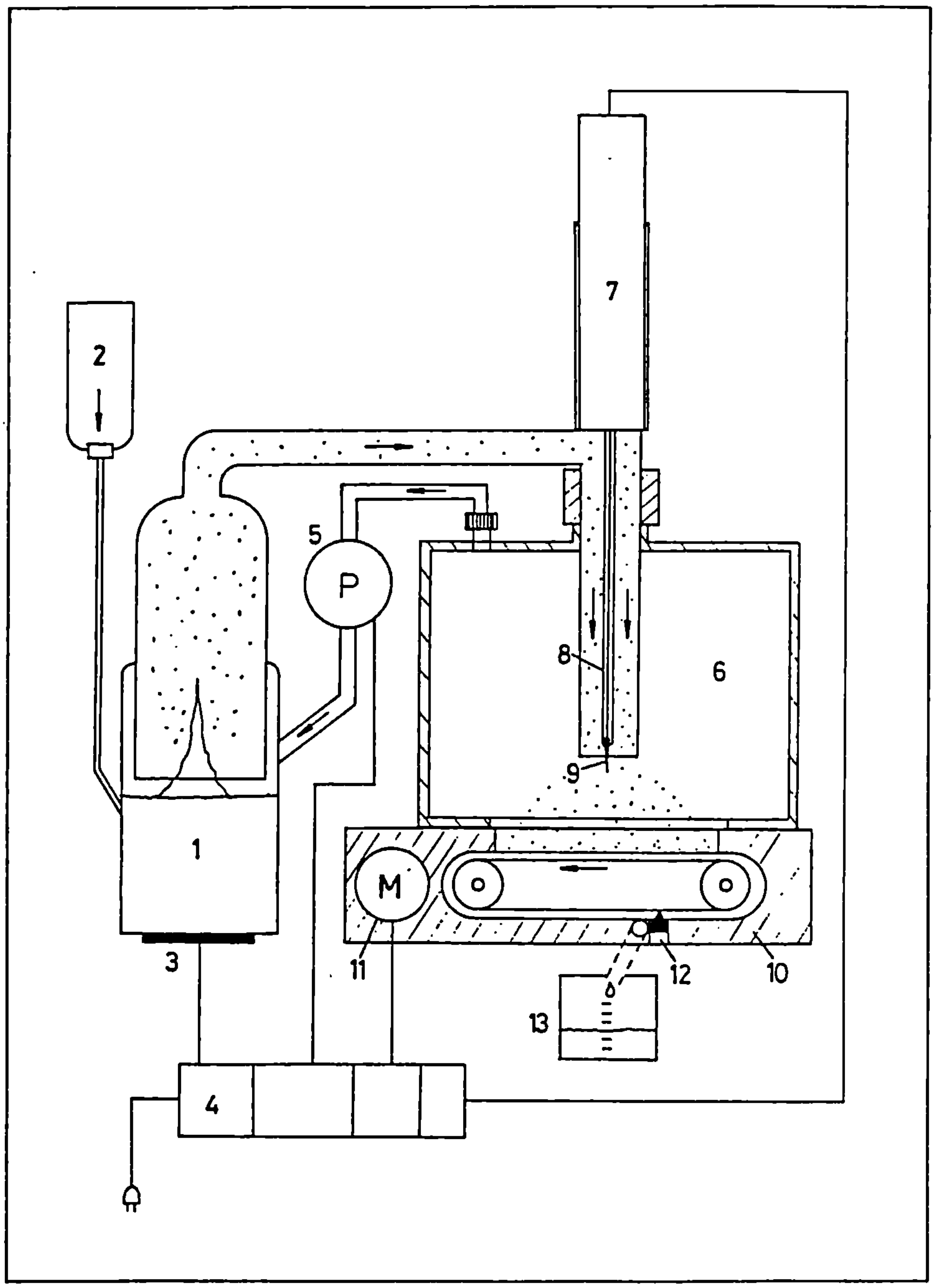


Figure 1 **Principle of the experimental set-up for demonstrating film etching using a chemical agent as a fog**

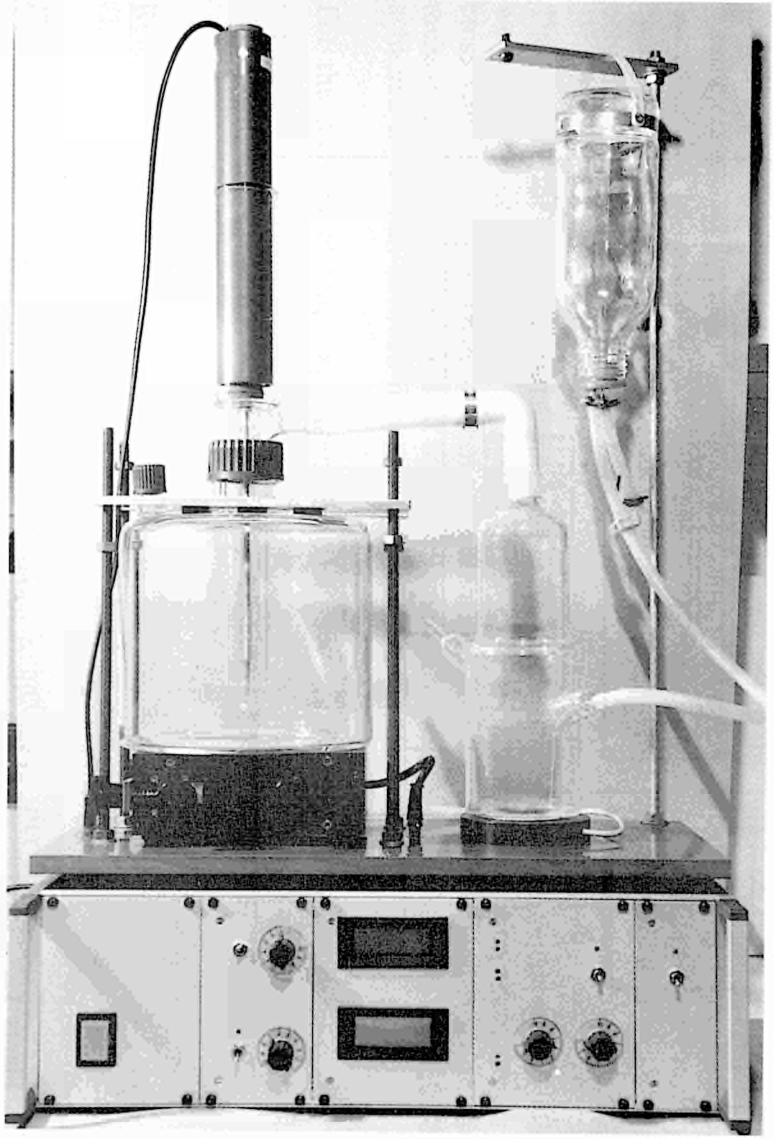


Figure 2 **Experimental set-up for the investigation of film etching using a chemical agent as a fog**

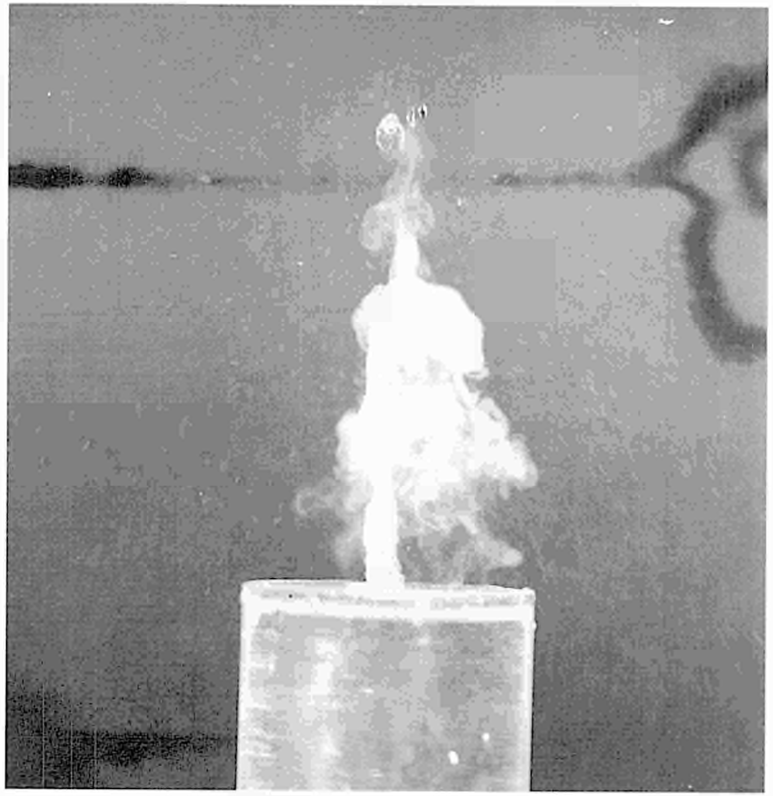


Figure 3 **Typical droplet cloud produced by a gyser atomizer**

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

At the end of 1991, eleven research contracts relating to Area No. 3 were at the stage of execution.

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

Contractor: TÜV Bayern
Contract No.: FI2D-0007
Work Period: July 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

In November 1990, a two-stage filter system (HFA and ARFA), each equipped with high-efficiency aerosol filters, was installed. ARFA is equipped with an automatic dust cleaning device. At the beginning, ARFA showed an average pressure difference of 150 Pa, which rose steady to nearby 1000 Pa during the first semester of 1991.

The pressure differences registrated since July 1991 averaged the 1000 Pa (figure 1) [use of a programmable data logger]. The exhaust air temperature remained at approx. 24 °C throughout the entire work period, i.e. from November 1990 to December 1991, whereas the relative humidity of the exhaust air in the above period averaged at 33%; fluctuations up to 12% may occur within 24 h (Figure 2).

Dismantling work has mainly been carried out with grinder (7 1/2 month), plasma arc (nearly 2 months) and milling (1 month) (Figure 3). The first trial discharge of the filter boxes showed that about 60 kg of dust had accumulated so far (Figure 4). Due to delay in the dismantling work of KKN Niederaichbach, remote-controlled dismantling is not expected to start before the end of 1992.

According to a statement by the decommissioning management of KKN Niederaichbach, increased generation of dust is not expected until 1992. The contract has therefore been subject of a time extension until June 1993.

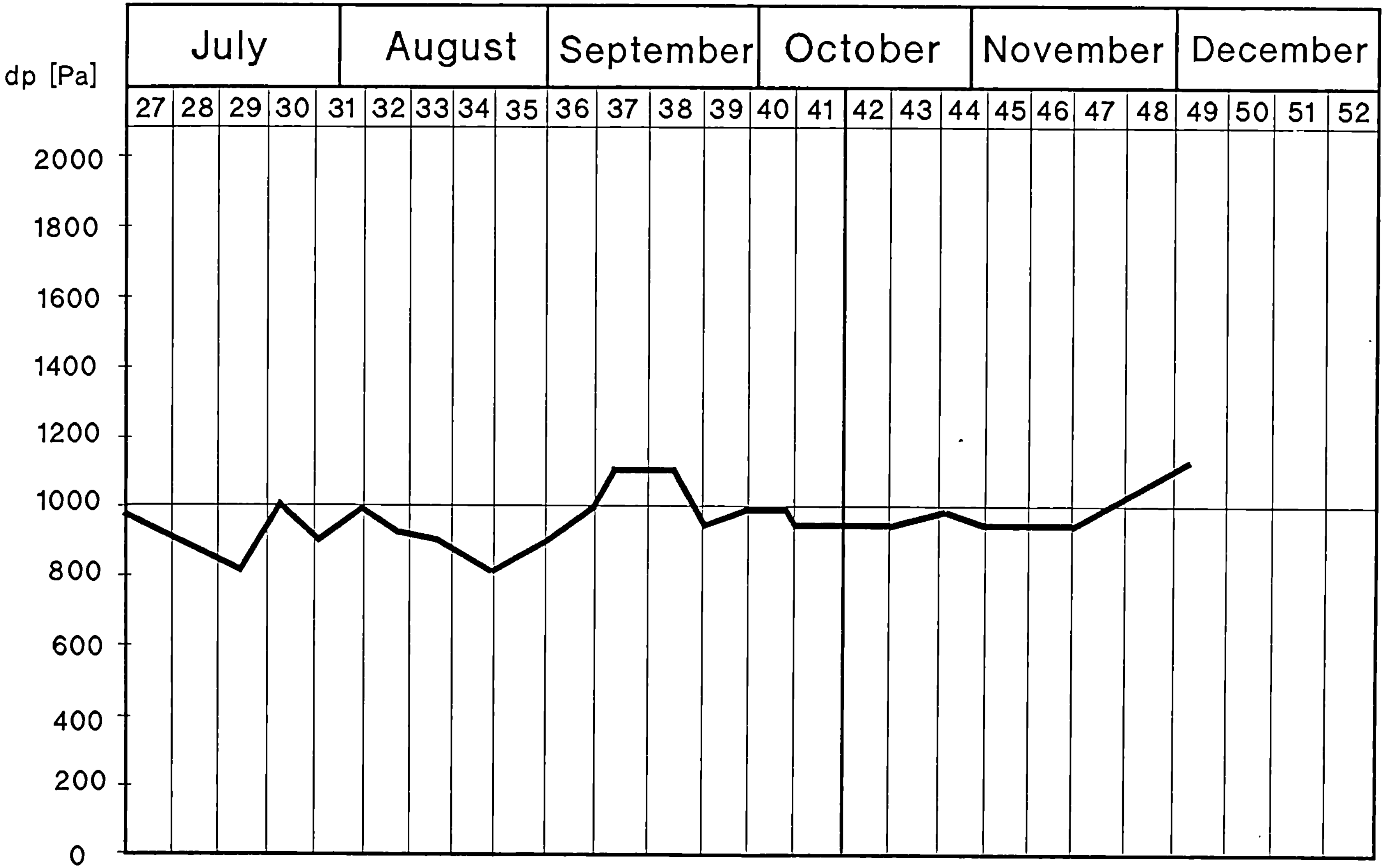


Figure 1 : Mean effective pressure difference of S-filters

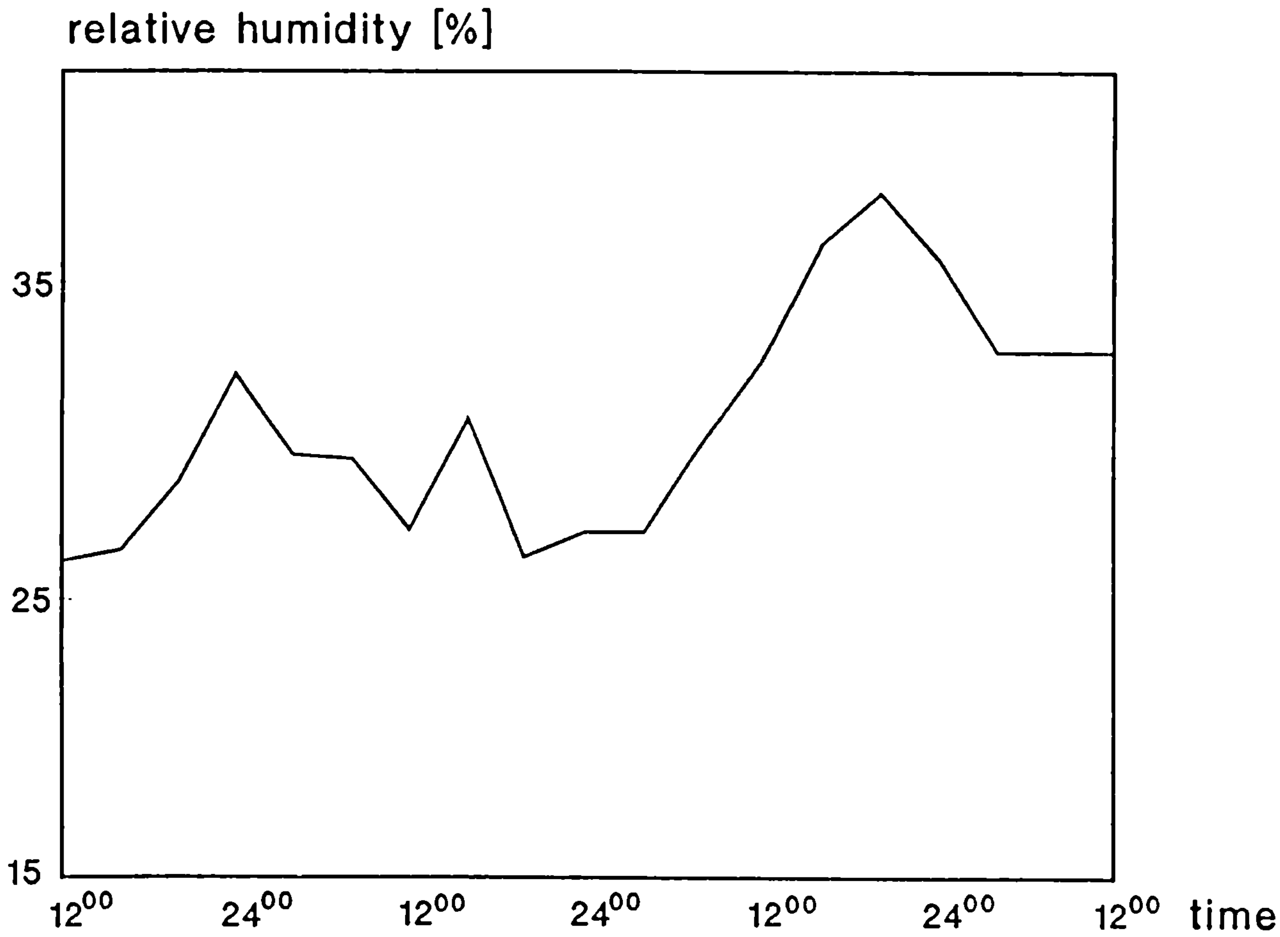


Figure 2: Relativ humidity of the exhaust air

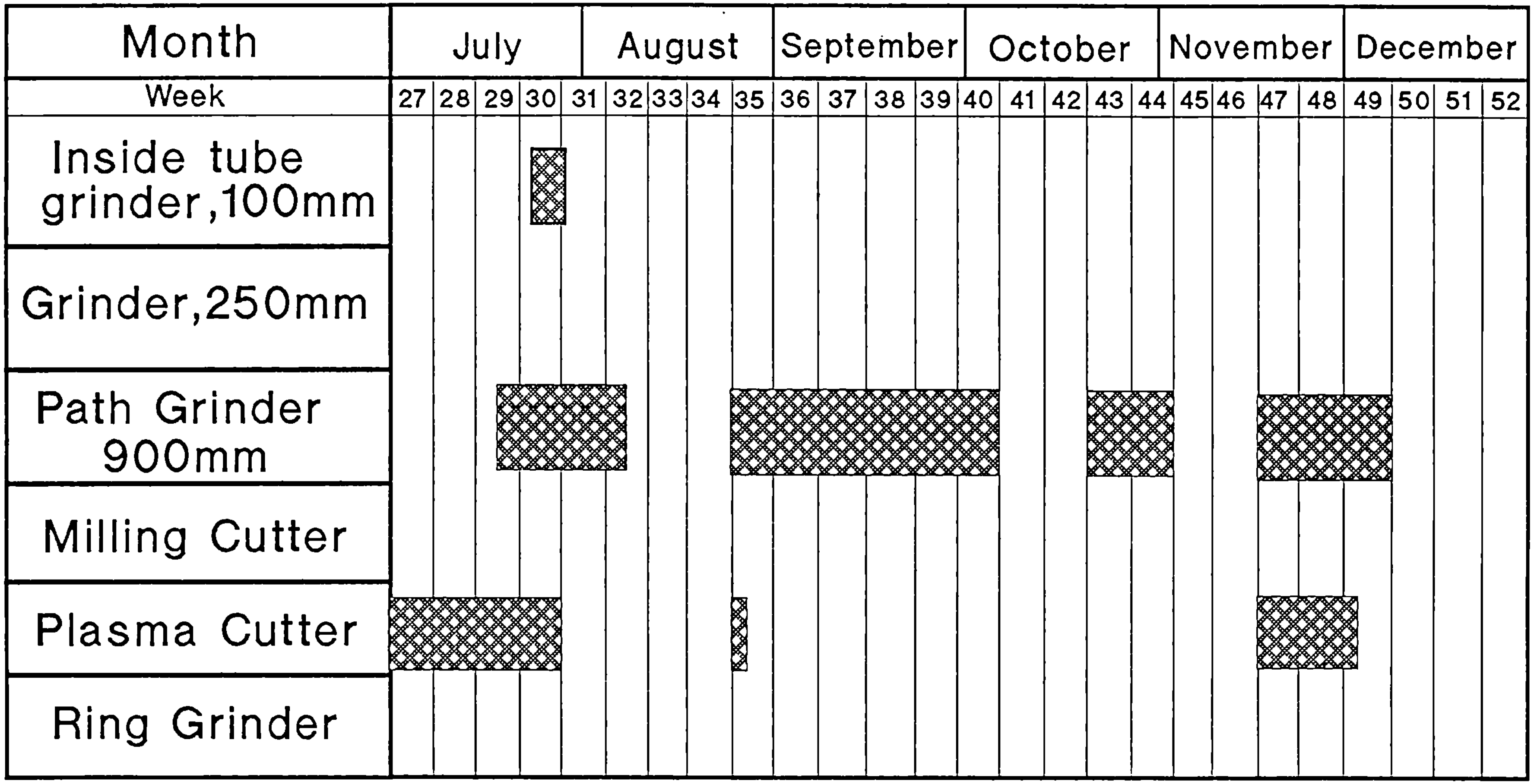


Figure 3 : Timetable for cutting methods 1991

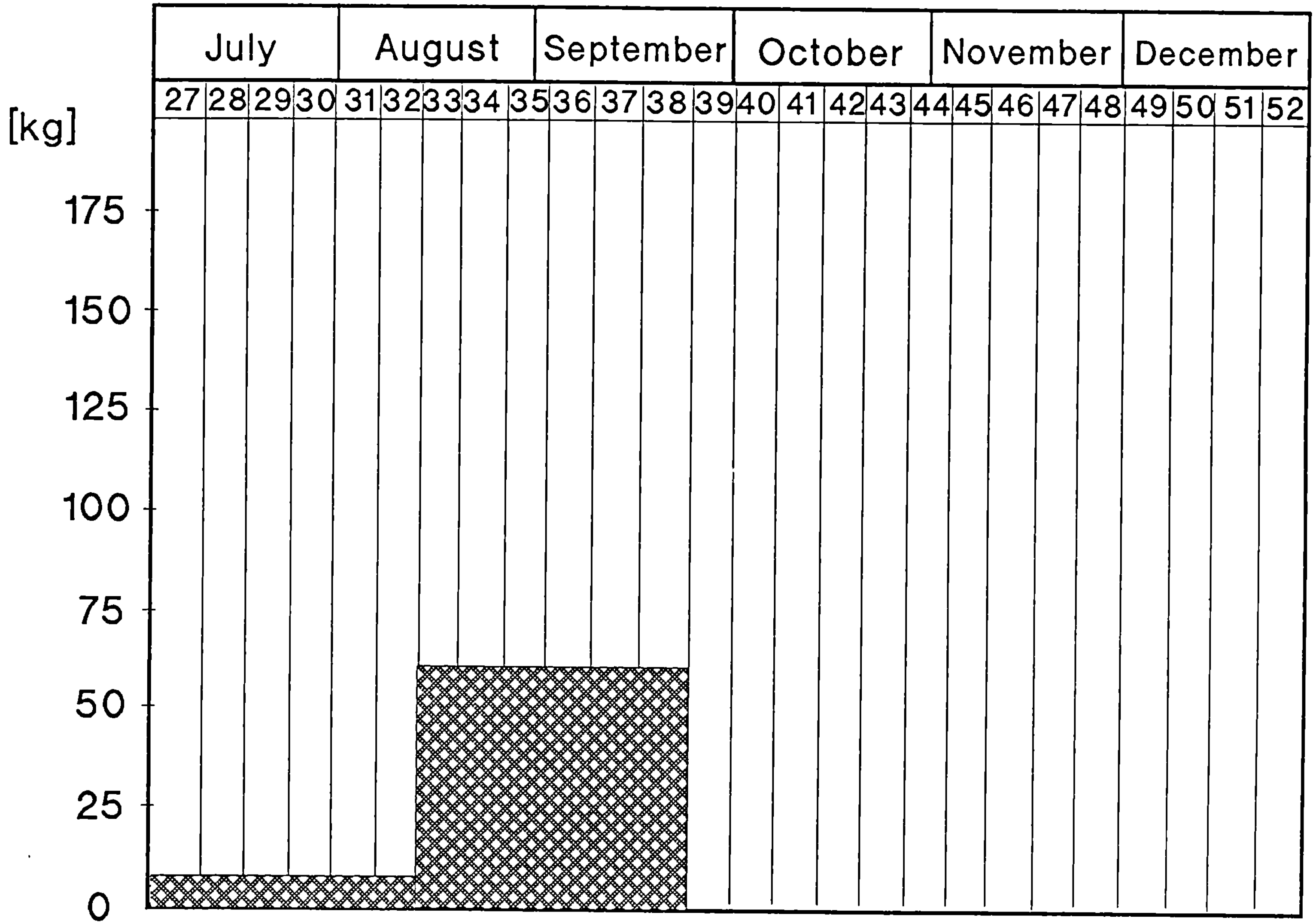


Figure 4 : Dust mass in the plastic bags

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 - June 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

For the remote-controlled application of abrasive water jets it is necessary to develop sensor systems to supervise both the state of the cutting tool and the cutting result.

Measuring the sucked-in flow rates gives an information about the state of the focusing tube (correlation of flow rate and diameter).

Instead of measuring the flow rate also the pressure loss in a specific length of the transport hose can be measured and correlated with the air flow rate. So the pressure loss can indicate the focus diameter, too.

The supervision of the cutting result was tried to realise by sound pressure measurement and frequency analysis. There are sometimes characteristic frequencies for cutting through which cannot be seen in case of kerfing. But the results of sound frequency analysis as well as the comparison of the sound intensity up to now are not reliable for controlling the cutting result.

So tests with a deflector plate and an accelerometer were carried out to find out the state of kerfing by the detection of reflected abrasives.

The development of new material for focusing nozzles gives an increase of the standing time by more than a factor of 40.

Progress and results

2. Development of controlling systems (B.2.)

2.1. Experimental setup

To develop useful sensor systems tests were carried out in a water basin. For controlling the tool different measuring devices were installed (fig. 1).

For the supervision of the state of the cutting head a system to measure the sucked-in air flow rate was adapted. Additionally the pressure loss in a special part of the transport hose was detected.

For the detection of the cutting result a hydrophone was used. The frequencies of the sound pressure were analysed by Fourier-Transformation. Sound frequencies were measured for cutting through and kerfing a standard workpiece (austenitic steel).

Additionally kerfing tests with a deflector plate were carried out to use the excitation of this plate to detect the fact of kerfing instead of cutting through.

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

The diameter of the focusing nozzle can be controlled by measuring the flow rate of the sucked-in air. Unfortunately this is not very easy to measure because the sucked-in air is used also to carry the abrasives into the cutting head; so in that case also abrasives have to pass the instrument and can destroy it.

So it seems to be more useful to control the diameter of the focus by measuring the pressure loss in a specific length of the transport hose. Fig. 2 gives the results of tests. The pressure loss is proportional to the sucked-in air flow rate and so it can be correlated with the focus diameter. In addition also rapid changes of the pressure loss indicate a change of the conditions of the abrasive transport (broken or clogged transport hose).

2.3. Development and adaptation of controlling methods to verify the cutting result (B.2.3.)

For remote controlled operation of the abrasive cutting process a method is necessary to detect the cutting result. In case of using abrasive water jets optical and mechanical sensor systems are difficult to apply. Because of the suspended particles the optical conditions in the water are very bad, additionally mechanical systems can be destroyed by reflected abrasives. The produced kerf is very small so there is no possibility to bring sensor systems in.

So the measurement of the sound pressure seemed to be useful method to detect the cutting result, because it is a non-contacting system which also is not sensitive against the particle load. To distinguish between cutting through and kerfing sound frequencies were analysed.

Fig. 3 gives the typical analysis of cutting through on one hand and kerfing on the other. All the tests were carried out on standard samples (size, material) under similar conditions (cutting parameters, position in the water basin). So, there is no influence of these parameters on the results.

But, unfortunately, up to now the **frequency** analysis does not seem to be a reliable method to detect the cutting result, because for both cutting methods - kerfing and cutting through - different analysis are occurring often, too. Up to now there isn't found out any reason for this scattering of the occurring frequencies.

Also differences in the sound **intensity** have not been reproducable. So they can't be used for the detection of the cutting result, too.

Fig. 4 gives the results of another method to detect the case of kerfing. Instead of a hydrophone an accelerometer is fixed on a plate very close to the focusing nozzle (see fig. 1). In case of kerfing reflected particles will hit this plate. The sensor is able to detect this signal.

This measuring method is also non-contacting and easy to apply. So further tests will be carried out to qualify this technique for controlling the cutting result.

3. Wear of the cutting head (B.3.)

Due to the fact that new designed materials are available now for the focusing nozzles the standing time of these parts has been extremely increased. Fig. 5 gives the results of comparative tests with normal standard focusing nozzles and the new material. The working parameters are normally causing a high rate of wear because of the high pressure and the high flow rate of abrasives. From the obtained results it can be calculated that the new nozzle material will allow about 100 h of operation time without changing the nozzle and without losses in cutting performance.

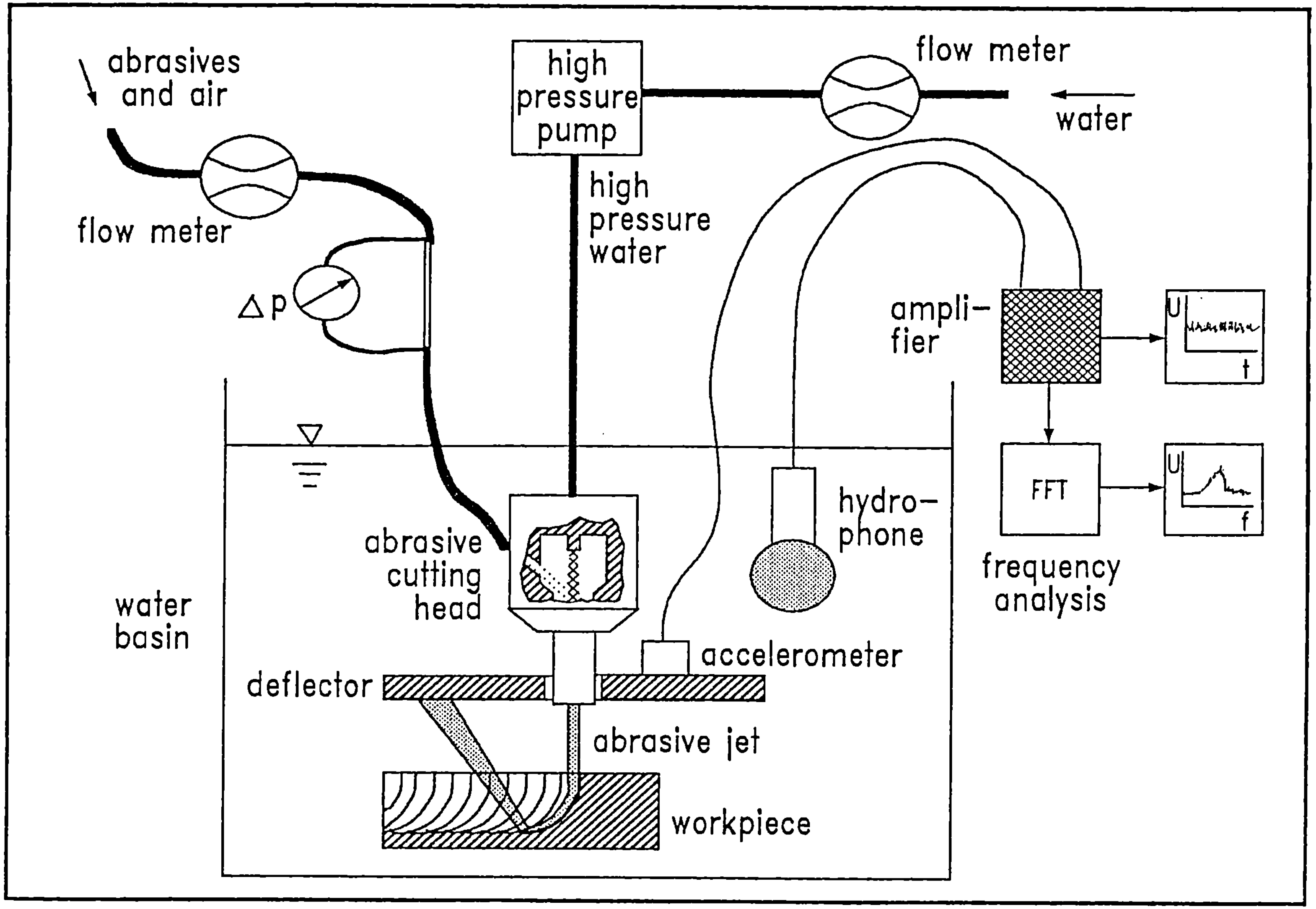


Figure 1 Experimental setup

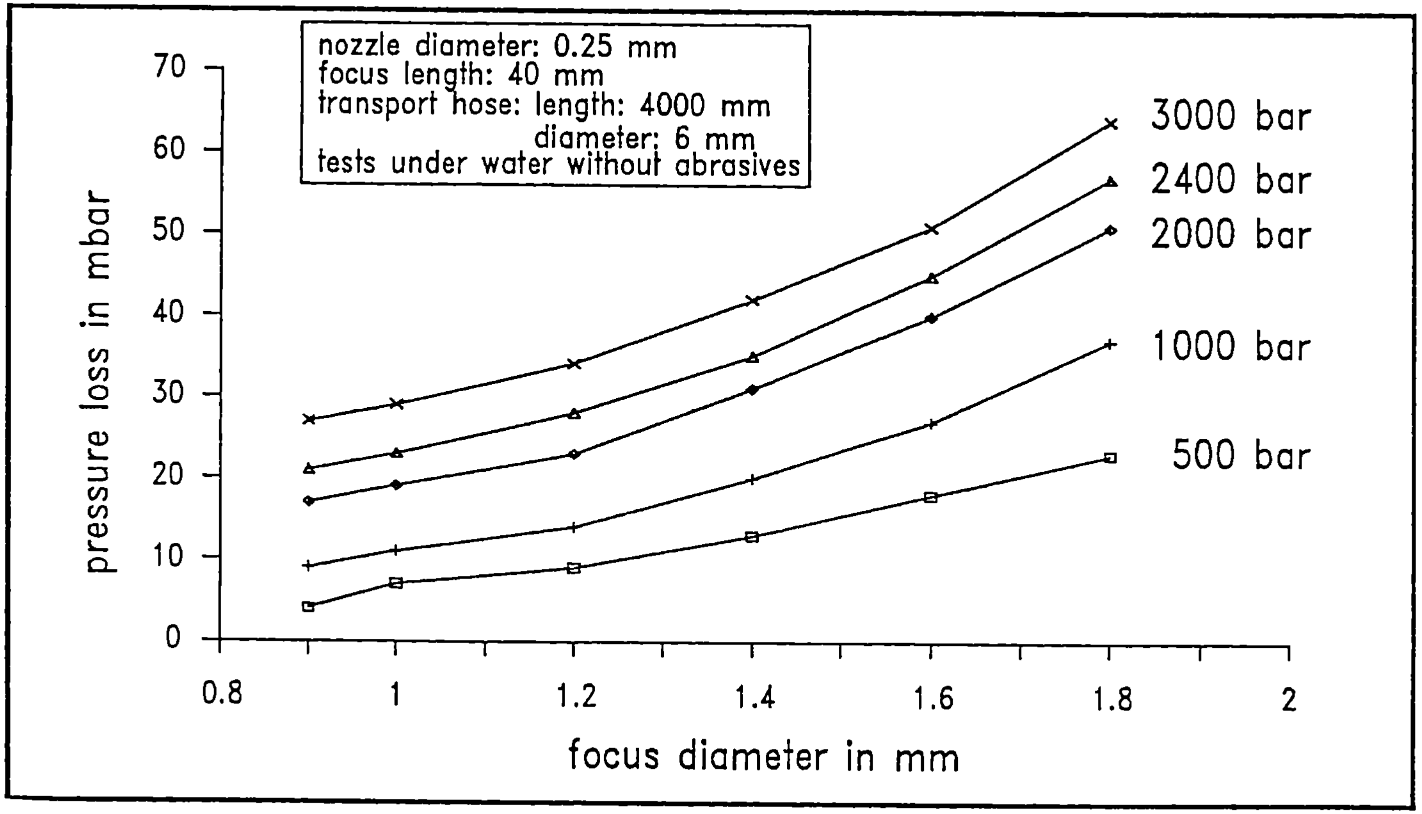


Figure 2 Effect of focus diameter on pressure loss

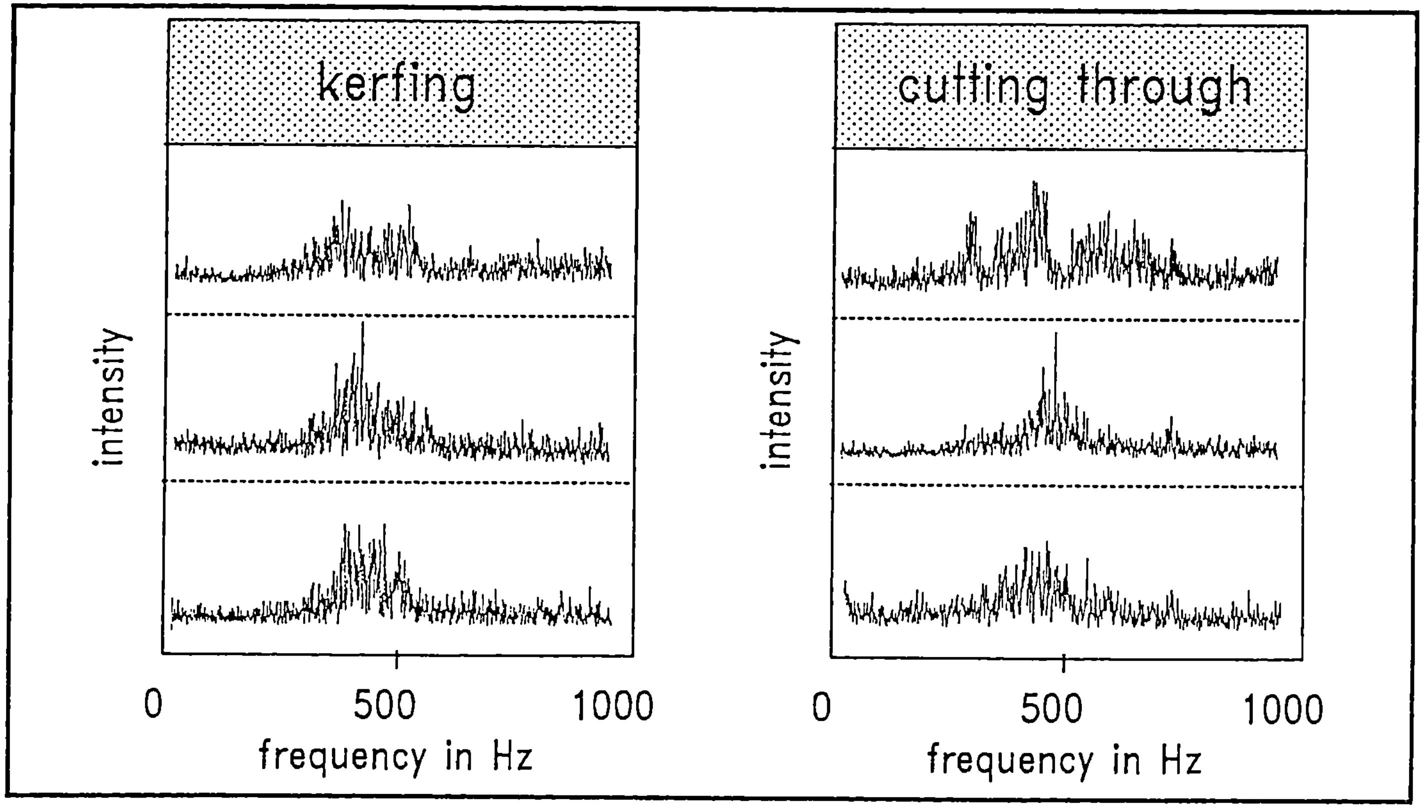


Figure 3 Comparison of sound intensity for cutting and kerfing

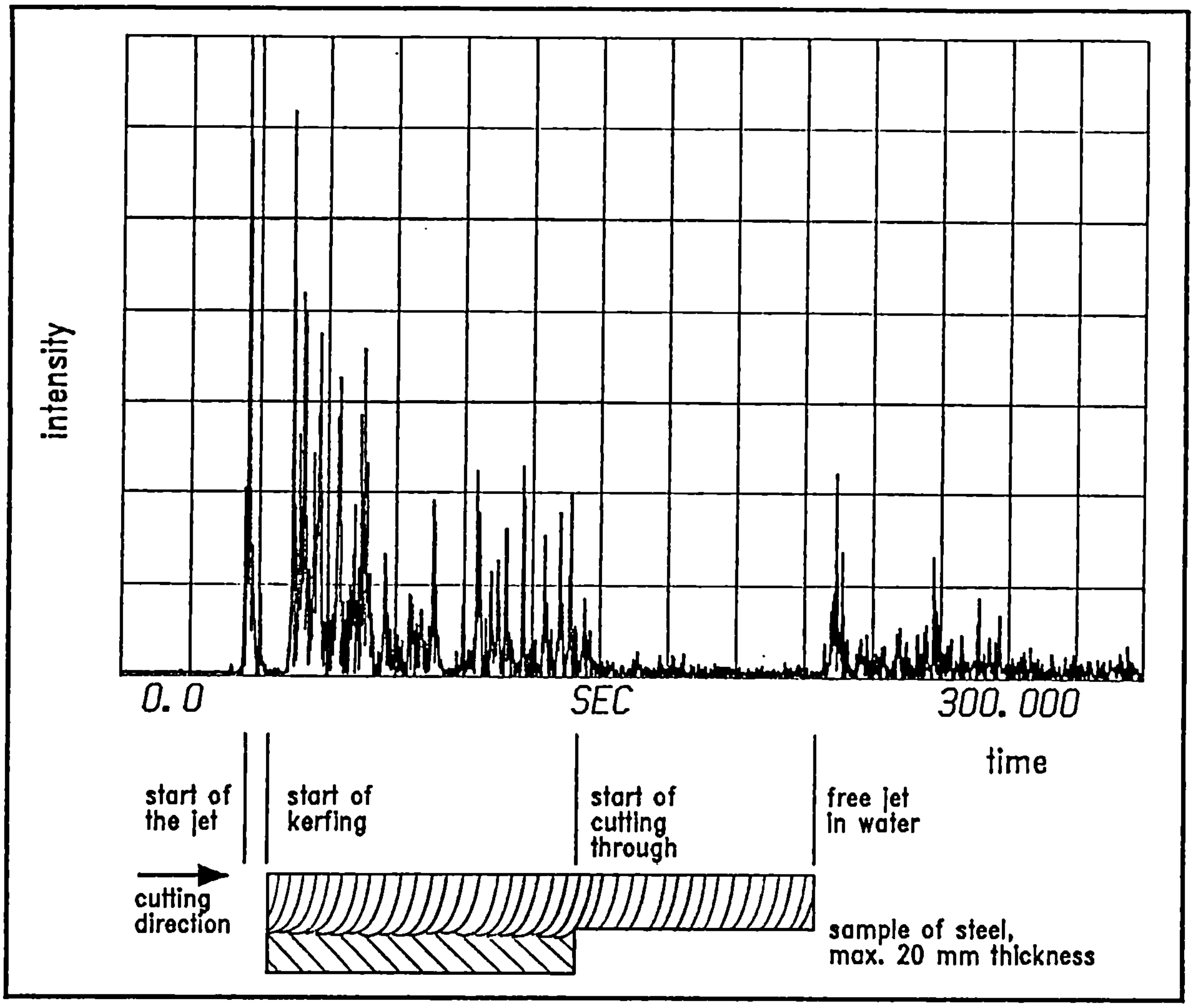


Figure 4 Detection of kerfing by an accelerometer

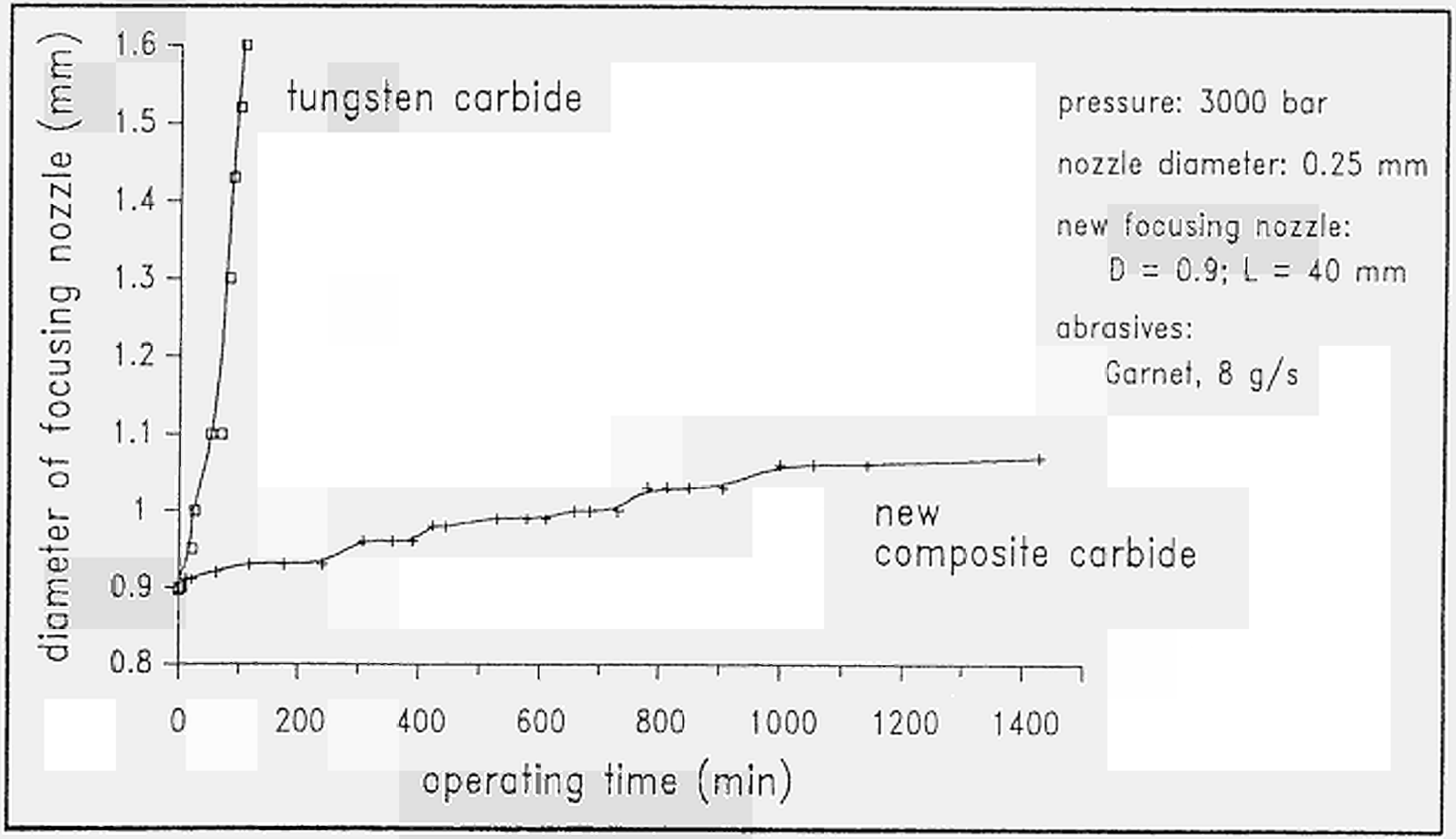


Figure 5 Wear of a new type of focusing nozzles

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.

C. Progress of Work and Obtained Results

Summary

The first phase of the work programme, "determination of basic charge parameters" (B.1.), has been completed. A first batch of 8 charges, differing in the manufacture of the confinement, were manufactured in January-February and fired against targets at different stand-offs (charge to target distance) in March. During four tests pressure transducers and accelerometers were also employed. Another batch of 18 charges was manufactured during April-May and tested against steel blocks at a fixed stand-off in June. These charges differed in : quantity of explosive, liner geometry, initiation and wave shaping. A charge was fired without target in order to better study its jet by means of X-rays. In October the analysis of the experimental data was completed, the configuration which gives the best performances was determined and suggestions for a further optimization were given.

In November the second phase of the work programme, "optimisation of the cutting charges" (B.2.), started and in December the preliminary theoretical studies (B.2.1.) were completed. An analytical computer code for the study of the effects of asymmetries on the jet from a linear shaped charge was developed and the programme for the firing tests of the second phase was defined.

Progress and Results

1. Determination of basic charge parameters (B.1.)

The most important parameters which characterise a high performance cutting charge have been experimentally investigated. Two sets of tests were carried out in areas especially equipped with respect to personnel safety and data survey under conditions deriving from an explosion.

2. First set of firings (B.1.2.)

Two batches of four charges each, both in the reference configuration but differing in the manufacture of the confinement (cold formed or machined), were manufactured and tested against different steel targets at various stand-offs. During some firings also time h stories of pressure and ground accelerations were detected by means of pressure transducers and accelerometers. Results of these tests were:
- the manufacture process of the light confinement does not influence the final cutting performances;
- an optimum stand-off is between 1.5-2.0 BW (Base Width).

3. Second set of firings (B.1.2.)

No. 18 charges, differing in : quantity of employed explosive, liner thickness, liner angle, type of initiation and detonation wave shape, were manufactured. No. 16 charges corresponding to 8 different configurations were fired against monolitic steel blocks positioned at a distance of 100 mm (2 BW). A reference charge was also fired without target in order to better study its jet by means of X-rays. In fig.1 two X-ray photographs of the jet at two different times are shown.

4. Analyses of the experimental data (B.1.3.)

The experimental results (penetration performances, X-ray photographs, pressure data) confirm the suppositions formulated during the first theoretical study and in addition show very interesting and useful fracture effects on some monolitic steel blocks. In fig.2 a photograph of two completely fractured 200mm-thick steel targets is reported.

Up to now, with a 50mm-BW charge the best performances are obtainable by means of a configuration characterized by:
- copper liner: thickness 1.5 mm, angle 100 degrees;
- central initiation;
- stand-off distance 100 mm.

In fig.3 the cutting performances of the tested linear shaped charges versus liner thickness, liner angle and stand-off distance, are presented. On the basis of these three plots it seems that better results can be achieved by optimizing further the liner thickness and the stand-off.

5. Optimisation of the cutting charges (B.2.)

On the basis of the results obtained in the first phase and of some theoretical progress in the understanding of the jet formation phenomenon in linear shaped charges, a new computer code was developed and the experimental programme for the phase 2 decided.

6. Effects of charge asymmetries (B.2.1.)

Using some suggestions reported in a recent work, /1/, an analytical computer code , called OTOLSC (OTO Linear Shaped Charge), for the study of effects of asymmetries on the jet from a linear shaped charge was developed. By means of this computer code the following results were obtained:
- a very critical tolerance is on the liner thickness;
- an almost critical tolerance is on the symmetry of the explosive;
- non critical tolerances are on initiation position and confinement thickness.

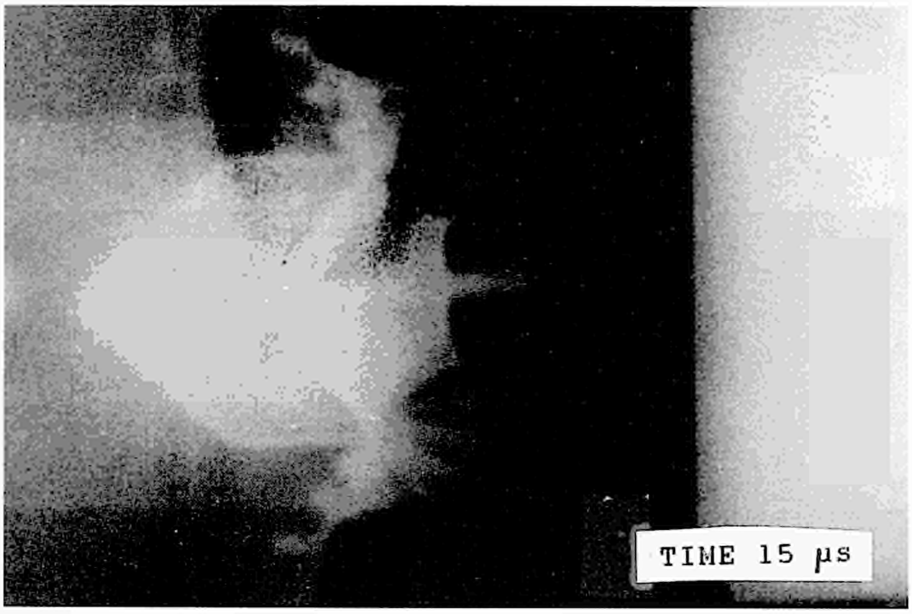
7. Experimental programme for the phase 2 (B.2.1.)

In order to completely eliminate the secondary effects of the projection of splinters from the confinement, it was decided to perform a first set of 6 firings by using unconfined charges. During this first subphase three different configurations, differing for quantity of explosive and type of explosive, will be tested.

On the basis of the results of these first trials, the final firings for the optimization of the liner thickness, stand-off and explosive quantity per unit length will be performed (10 tests). During this subphase the fracture effects on various targets will also be studied (4 tests).

References

/1/ D. C. Pack and J. P. Curtis, "On The Effects Of Asymmetries On The Jet From A Linear Shaped Charge", J.Appl.Phys. 67 (11), June 1990.



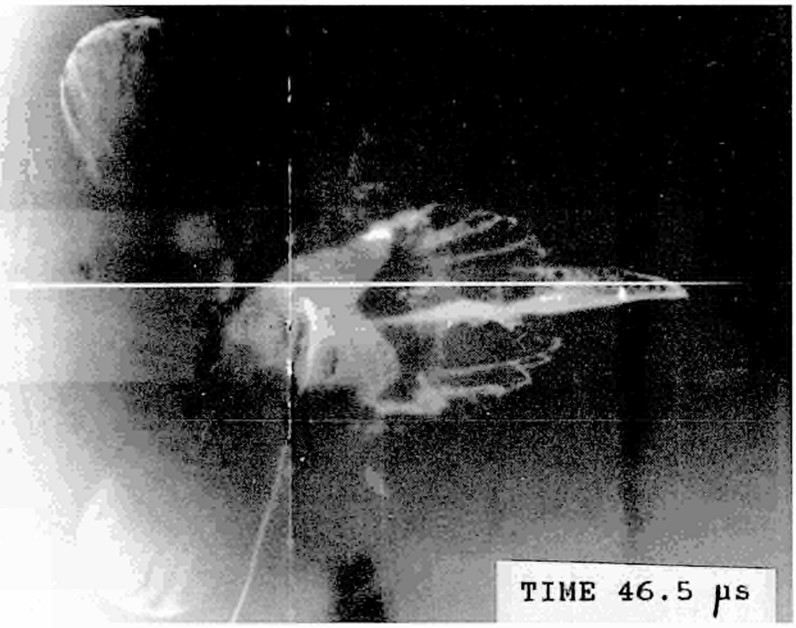


Figure 1: X-RAYS OF REFERENCE CUTTING CHARGE

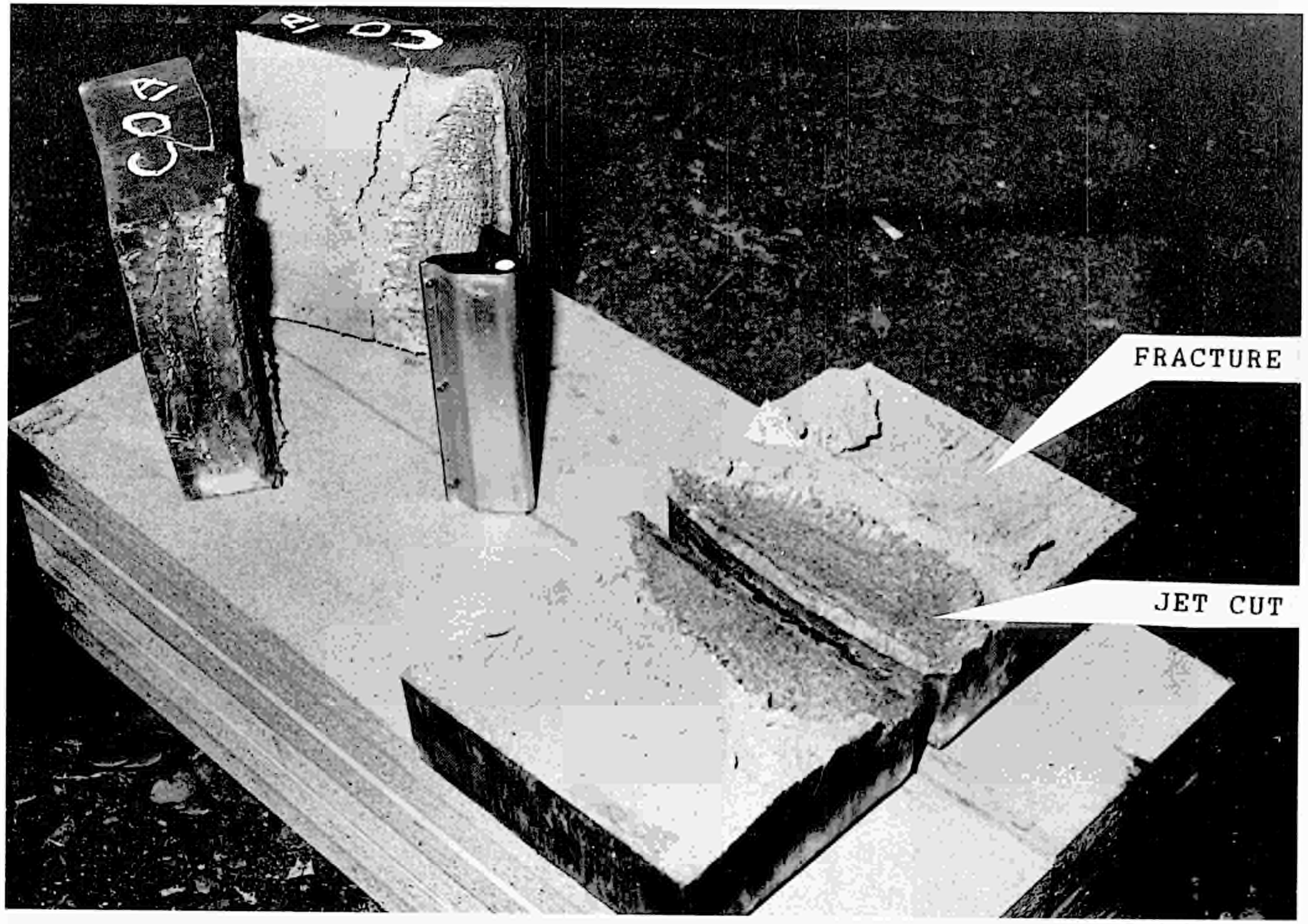


Figure 2: 200mm-THICK STEEL TARGETS

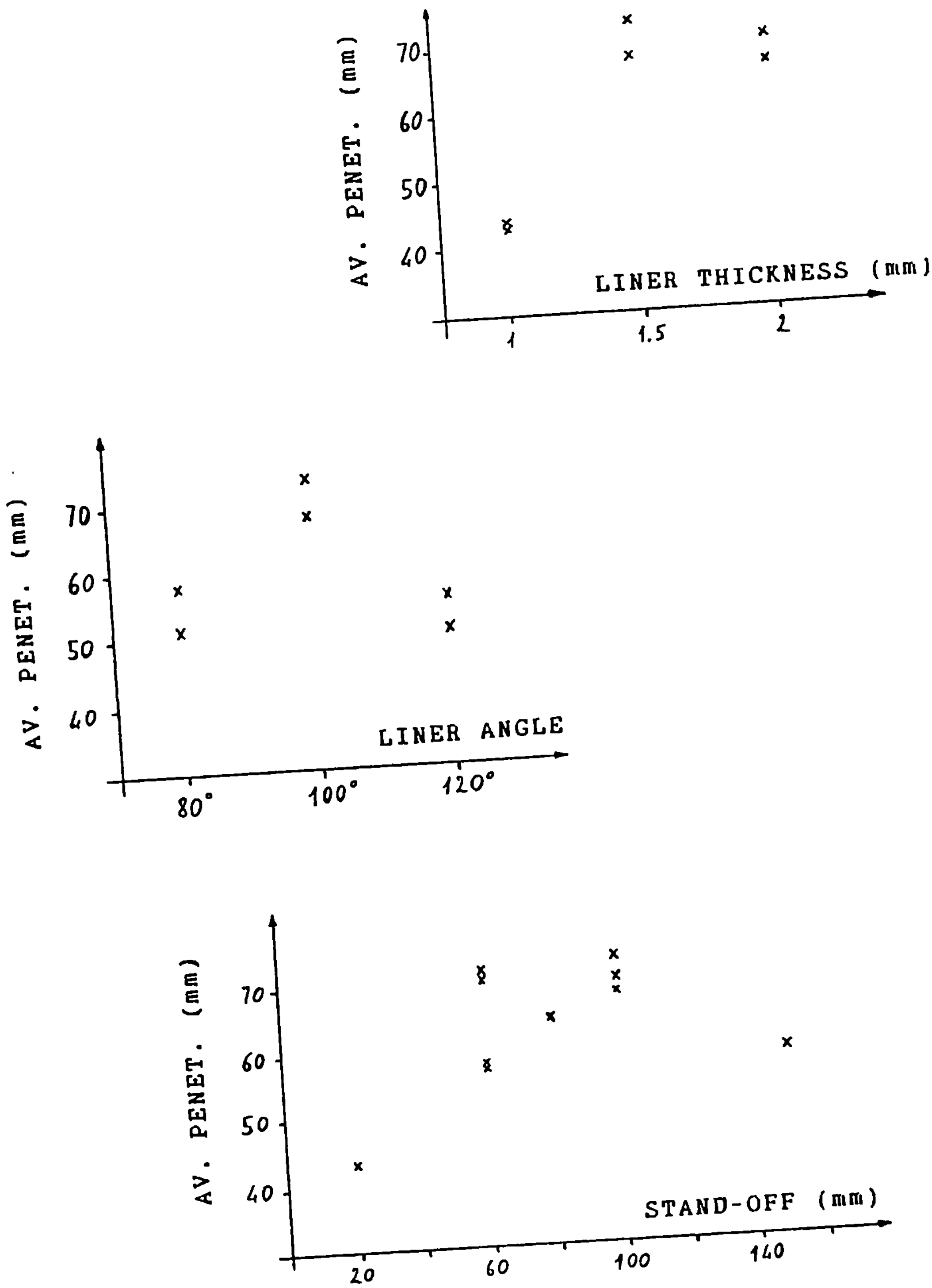


Figure 3: average cutting performances

3.4. EVALUATION OF THE SEGMENTATION BY VARIOUS CUTTING TECHNIQUES (PLASMA TORCH, ARC SAW, CIRCULAR DISC, ETC.)

Contractors: CEA Valrhô, CEA-Sac
Contract No.: FI2D-0013
Work Period: October 1990 - June 1992
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 alternative 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

The facility where the cuts take place, the analytical techniques and some test specifications have been described in the first annual progress report. Due to the different cutting time of the various tested tools, we have studied carefully the test procedures in order to compare the tools in the same conditions (at the steady state).

The optimal conditions of cutting to minimize as much as possible the volume of wastes have been emphasized.

All the measurements relating to the cutting with the grinder, the reciprocating saw, the plasma arc torch and the air arc torch are almost completed. First comparisons have been drawn from these results.

These comparisons concern the using range of the tools, the wear of the most exposed parts, the cutting speed (versus the thickness of the steel plate), the production of secondary wastes and their repartition (sedimented dross, attached slag, deposits on the cell walls, aerosols in exhaust duct).

Progress and results

B.1. Preparation of the testing cell

The preparation of the testing cell, the analytical techniques and most of the test specifications have been described in the precedent report. We have yet emphasized in the test specifications the necessity to have the most representative procedure of the cutting tests in order to compare the different tools in the same conditions.

We have selected three methods of measurements :

a) the measurements is done during a transitional period between starting and steady state noted "samplings" (Figure 1).
   It is thus necessary to introduce a correction factor to reach the real value for the steady state.
b) stop of the ventilation of the cell ventilation during six minutes from the beginning of the cutting (characteristic time to reach the steady state). Then starting of the nominal ventilation and of the measurements (Figure 2).
c) the samplings begin in the same time as the cutting. Cutting is first stopped and samplings go on until all the air of the cell has been cleaned (Figure 3).

The three methods are in good agreement when applying the correction factor to the concerned results.

The second method (b) has been used for almost all the tests save when using the air-arc torch. In that case, the tool is manually operated and the third method (c) run.

B.2. Results of cutting tests

B.4. Four cutting tools (over five) have been studied and their secondary emissions measured. We can thus give some first comparisons.

For the practical range of use of the tested tools, we can emphasize that :

- the grinder and the reciprocating saw are not adapted to cut steel plates with thicknesses superior or equal to 30 mm.
- the air-arc toch has to be used in several passes for thicknesses superior to 30 mm.
- the plasma arc torch has been a satisfactory tool for all the considered thicknesses.

The fastest tool is the plasma arc torch, its cutting speed is respectively multiplied by 2, 10 and 50 compared to the one the air-arc torch, the grinder and the reciprocating saw when cutting 10 mm thickness steel plates. These ratios increase (for air-arc torch and reciprocating saw) in case of 30 mm thickness steel plates (Figure 4).

With regards to the secondary emission produces, the following tendencies can be underlined :

- the air-arc tool produces the largest amount of wastes when cutting 30 mm thicknesses steel plates, its production of sedimented drosses is multiplied by 2, 3 and 4 compared respectively to the one of the grinder, the plasma arc torch and the reciprocating saw (Figure 5).
- the deposits on the walls of the cell are more important for the grinder than for the other tools, this is probably due to the spark jets induced by this tool.
- the wear of the plasma nozzle and of the blade of the reciprocating saw are negligible compared to the ones of the air-arc electrode and of the blade of the grinder.
- the reciprocating saw produces the least aerosols.

The size distribution of aerosols are often bimodal and sometimes trimodal. The different modes are a consequence of the aerosol formation, mainly mechanical erosion and fusion.

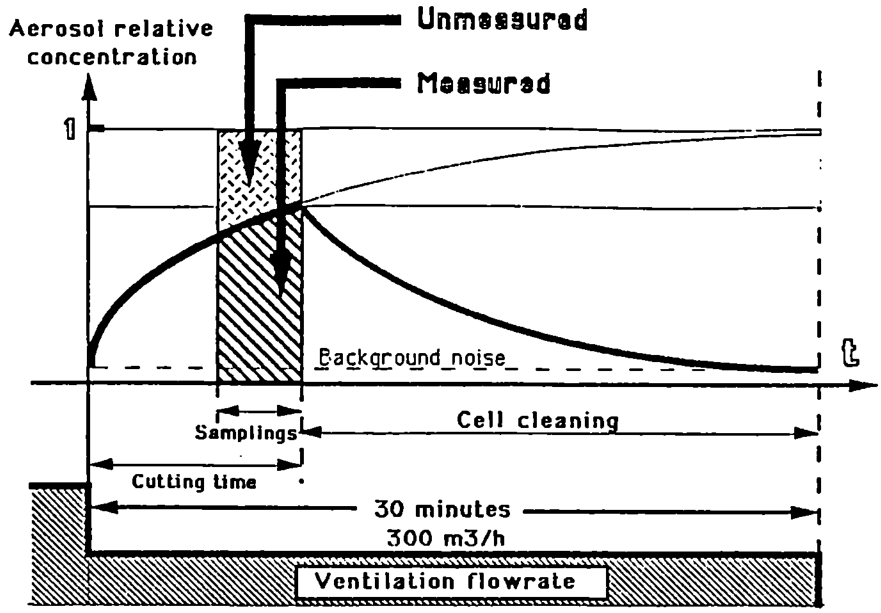


Figure 1 : First method for sampling aerosols

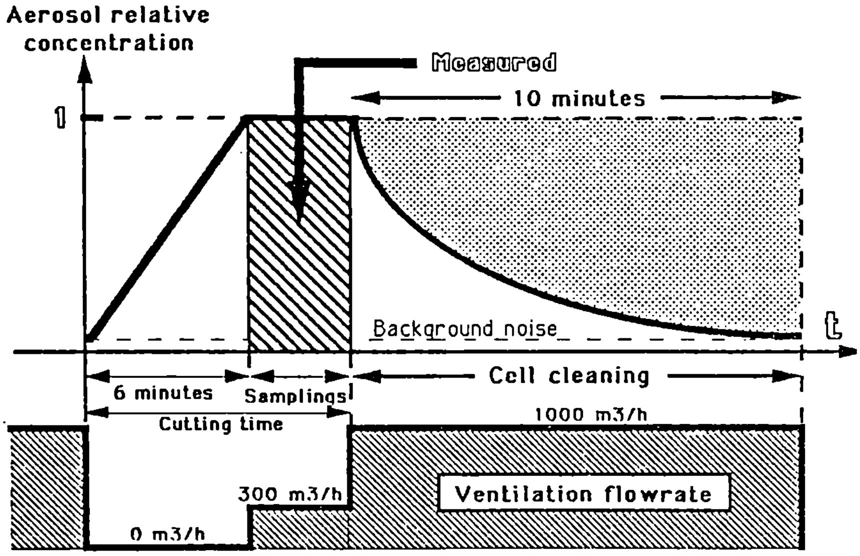


Figure 2 : Second method for sampling aerosols

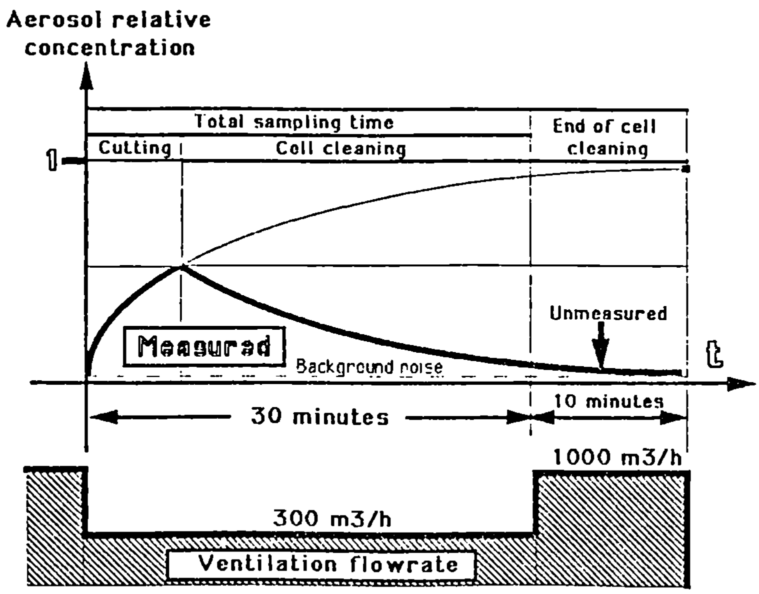


Figure 3 : Third method for sampling aerosols

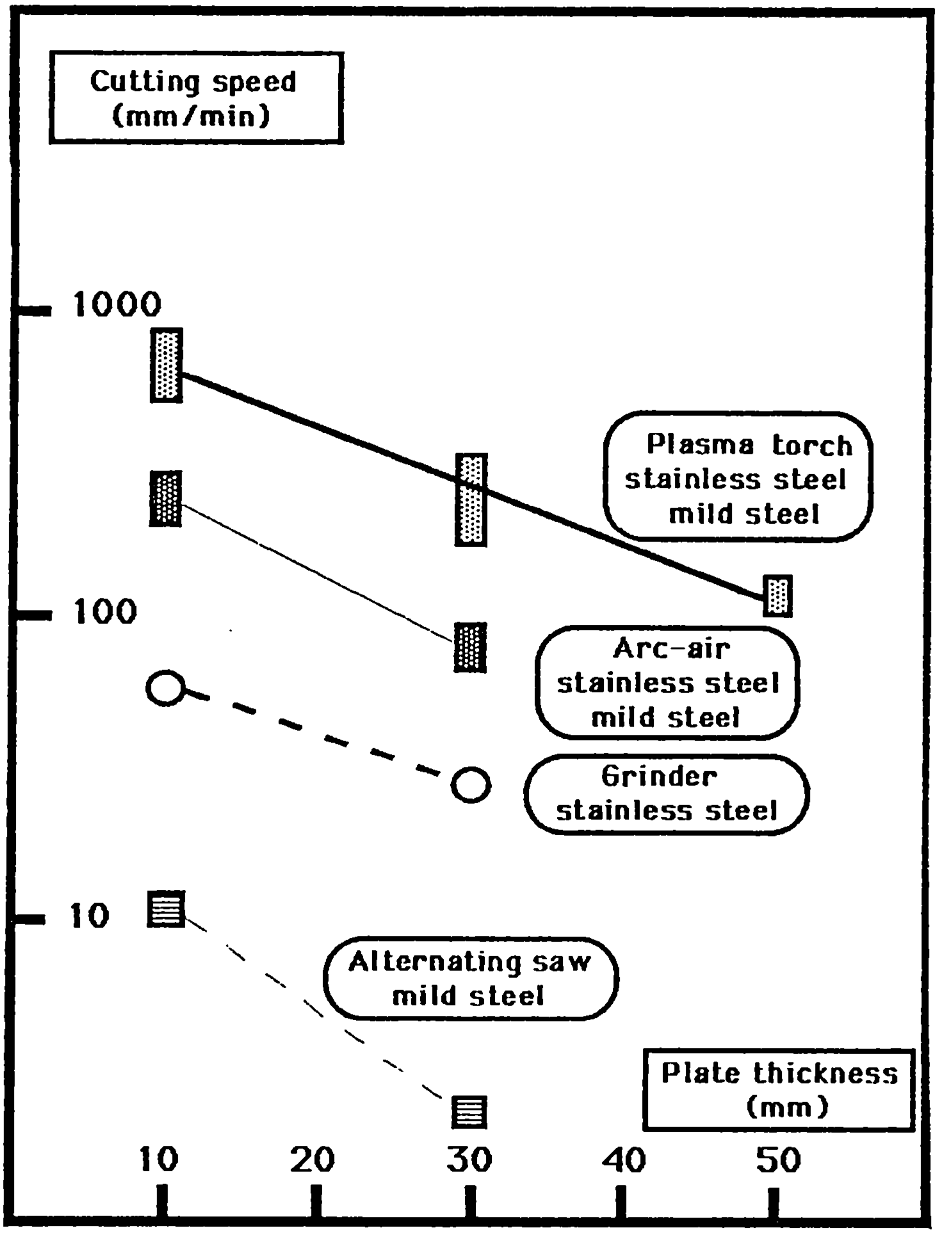


Figure 4 : Optimal cutting speed versus the thickness of stainless steel and mild steel plates for different tools.

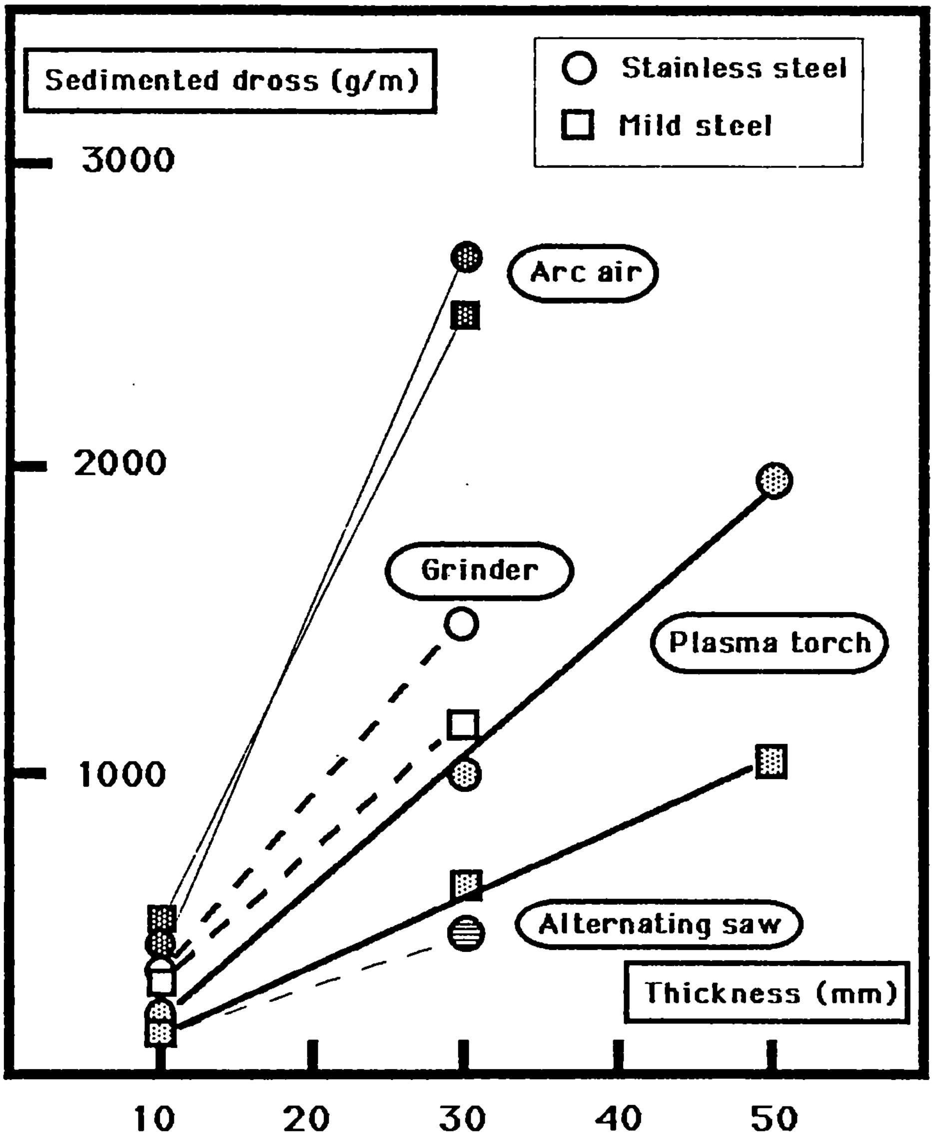


Figure 5 : Sedimented dross production versus the thickness of the plate.

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: R LEAUTIER, CEA Cad, SST/SMP/GP
Phone: 33/42 25 73 08 Fax: 33/42 25 32 62

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 1991 was connected with four main aims, in relation with the topics of work programme packages B.2, B.3, B4, B.5.

.The Pegase facilities such as the moving device and test tank have been studied, realized and installed in accordance with the requirements necessary to cut great thicknesses. The modification of the current nuclearized torch is being made. A 100 amp torch and a cooling system have been ordered. The study of supplying the torch with high voltage and intensity was analyzed

.Tests on the usable system and plasma saw, and the cutting process have been made in order to qualify working parameters. Then different material thicknesses were cut with a consumable electrode using stainless steel wire. Model n°1 (diagrid 200 mm, 20 mm borings) was cut with the plasma saw. The driving device for the movement of the plasma saw was completed and tested.

. As regards the control system for the given application, the inductive sensor was defined to satisfy accuracy, ability for process control, resistance to water and dirt, small dimensions, shielding against electrical and mechanical influences, functions for stainless steel materials. The development of a hole and edge detection algorithm and the specification for the microprocessor control unit have been made. Software modules have been specified and partly implemented.

.Research has continued so as to find samples for radioactive sample testing.

Progress and results

1. Development of the plasma torch, adaptation of the moving device and test tank (B 2)

In view of the accuracies required for cutting great thicknesses a new manipulator has been designed. The new design aims to limit deflection of the extremity of the manipulator in normal and transitory work phases, in order to meet the above requirements.

In order to carry out realistic tests, we need considerable dimensions for our models. Therefore we have studied a new test tank (see figure 1). In order to analyse the distribution and balance of the by-products the tank bottom has been equipped with a special pump and sump. This system allows the separate collection of sedimented dross and suspended particles(see figure 2).

In order to improve the capacity of our plasma torch it seemed necessary to increase the arc voltage (up to 300 volts) and the intensity (up to 1000 amps). Furthermore, increasing the cooling flowrate and lowering the inlet temperature is planned.

Unfortunately, we had some problems because of difficulties in finding a supply set able to supply 300 volts and 1000 Amps on the market.

Thus we finally decided to continue the work by :

- Modifying our current TD 600 N plasma torch by using a 8 mm diameter electrode, and parametric adaptation of the more adequate diameter tuyere (see figure 3)
- Ordering a PT 15 plasma torch from the L Tech Company. This torch is already qualified for 1000 Amps
- Ordering a torch cooling system.
- Carrying out tests with our four power units in parallel and series parallel secondary connections. Thus it should be possible to reach 1000 Amps with 200 volts and 340 Amps with 300 volts.
- The results of these parametric tests should confirm by extrapolation our investigation, i.e. high intensity and high voltage are necessary to cut great thicknesses

2. Development of other tools (B.3)

. As regards the consumable electrode tool, differents types of stainless tubes were cut, and it was found that tubes with greater wall thicknesses could be cut more easily in relation to other tubes.

Furthermore different material thicknesses were cut and it was found, that cutting quality depends on the efficiency of the water jet. Quality is relatively satisfactory up to a depth of 40 mm. Problems appeared when trying to cut a 110 mm diameter tube with 54 mm diameter bolt and a 3 mm wire. Other tests will be made with a wire larger in diameter (from 3mm to 4 mm) and a second welding rectifier connected in parallel in order to increase amperage.

. Concerning the plama saw, a motoring device for the up and down movement of the torch and for its rotation has been realized. Cuts with the plasma saw were performed on a workpiece of a 200 mm maximum thickness with borings of 20 mm in diameter (see figure 4).

It was found that this tool will only work acceptably when the plasma saw cuts from the top to the bottom of the workpiece. Difficulties appeared when cutting upwards, attached slag remains in the kerf, secondary arcs increase the wear of the nozzle and may even destroy the tool.

Thus it is impossible to produce the required width of kerf by rotating the saw at a single angle, as was proved manually. The result leads to the conclusion that it will be very difficult to automate the cutting process.

Further tests will deal with the improvement of the automatic control of the moving device (see figure 5).

. Regarding the control system, further studies were done on observing and positioning the cutting process. A single CCD camera cannot offer a sufficient image. It will be necessary to use two cameras with a self-adjusting iris to illuminate the surroundings in order to negate the brightening effect of the plasma arc.

The figure 6 shows the principle and the counterpart to connect tools to the Cadarache manipulator.

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

. For the given application the inductive sensor has to fulfill accuracy specifications, process control specifications and general requirements. Two inductive sensors have been tested, one with a 30 mm and one with a 15 mm diameter. The sensors are embedded in water resistant metallic cases and so protected against dirt, mechanical stress and external electromagnetic fields. The tests showed the most useful parameters such as measuring range, diameter of the sensor material of the workpiece, influence of water on the distance curves. Also one main task for the process control is the detection of workpiece edges and holes. Finally for geometrical tracing of the workpieces during the teaching phase, a 15 mm diameter inductive sensor was chosen. Further tests were made with a high precision X/Y/Z axis manipulator to obtain sensor signals similar to those expected from the manipulator of the Cadarache Laboratory. In order to achieve an absolute measuring error of less than ± 1 mm several algorithms have been tested

Regarding the process control, the first tests showed a dependence between the nominal distance of the inductive sensor and the resulting dynamics of the distance curves during a lateral movement above the workpiece. The evaluation algorithm for the inductive sensor distance signal used a kind of adaptive trigger value for position determination of the holes. A short overview of the software structure and the data flow is given in figure 7.

For that purpose 3 large software modules will be developed : the user interface module, the general management module, the manipulator interface module.

Furthermore, for an automatic determination of the trigger values a proportionnal correlation between the starting values and the trigger values was estimated.

This proportional correlation could not prove its function during the following tests with different nominal distances.

Therefore a trigger curve has been determined so as to obtain the correct trigger value dependent on the nominal distance above the workpiece. This trigger is shown in figure 8.

When the area of influence is known, the algorithm for detecting holes and edges can be defined as follows : Reading the sensor signal over the trajectory, approximate determination of hole and edge, determination of nominal distances at the workpiece, determination of trigger value from the trigger curve, sequential search for edges. Some of the results are presented in figure 9.

. During the project period, the electrical interfaces between the components of the system have been defined. Stepper motors have been chosen as driving units, and are interfaced with the system control microprocessor through a digital interface.

The interface for the inductive sensor allows a bidirectional data transfer for the exchange of messages, commands and thickness values.

A suitable user interface for the manual control unit for the four axes manipulator has been developed.

Figure 10 shows the layout of the manual control panel.

For direct control of the four axes an analogous joystick with two degrees of free motion has been mounted into the panel

4. Preparation of radioactive samples (B.5)

Contacts have been made to obtain radioactive samples from French dismantling sites. As a result, the Marcoule site will supply samples, from the G3 reactor.

The features of these samples are :
- Parts of tube 800 mm in diameter
  L = 800 mm l = 200 mm e = 15 mm
- Rate of contamination : between 10^2 and 10^3 Bq Cm^{-2}
- Radio elements Co 60, Ni 63, Fe 55

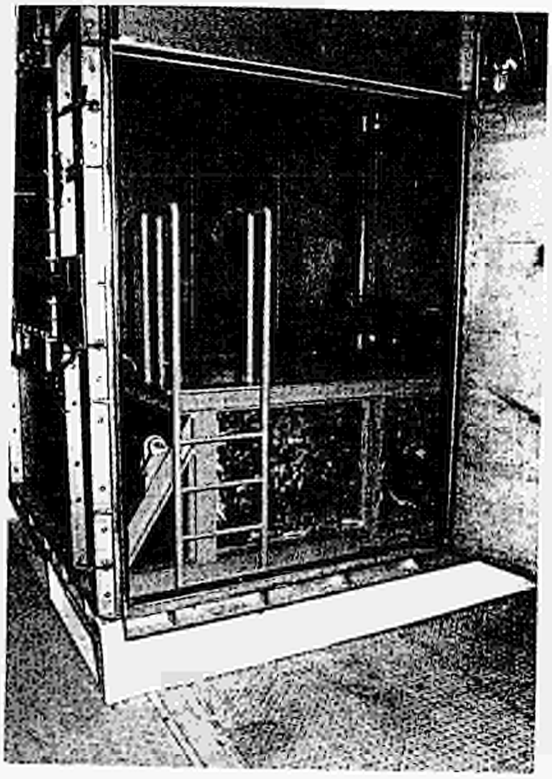
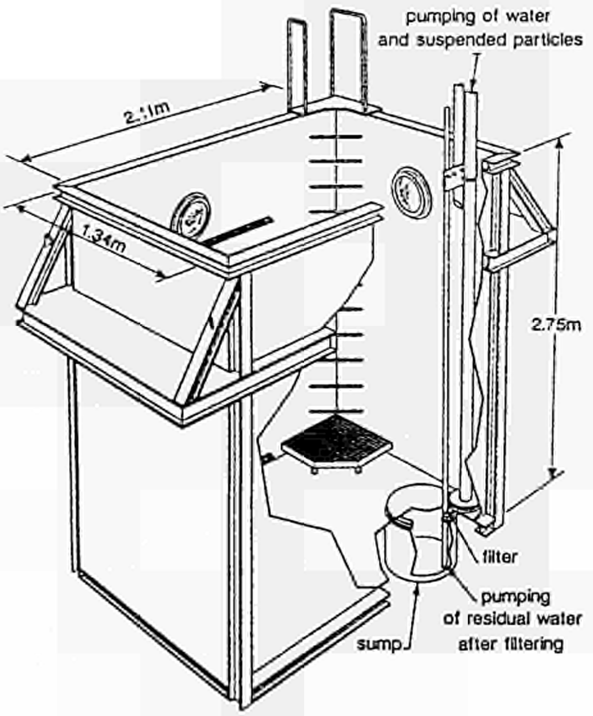


Figure 1 :
Manipulator
and test tank
inside the containment



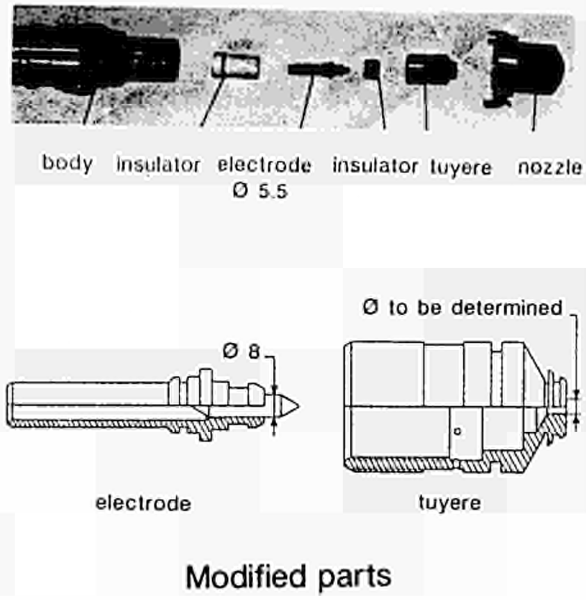
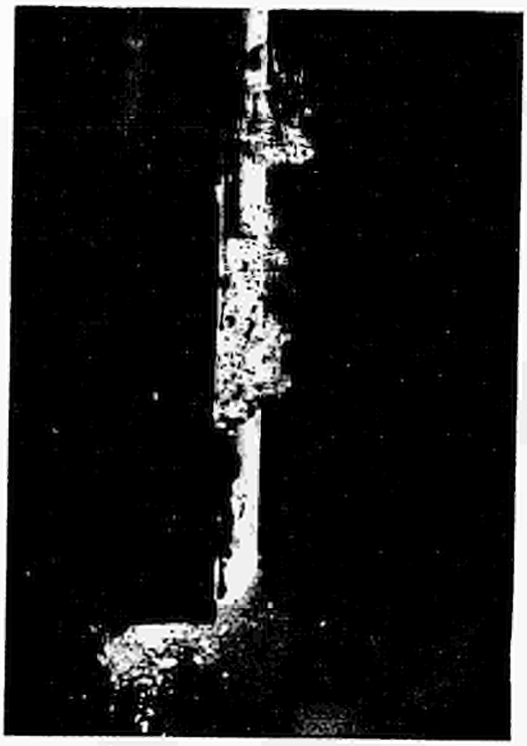
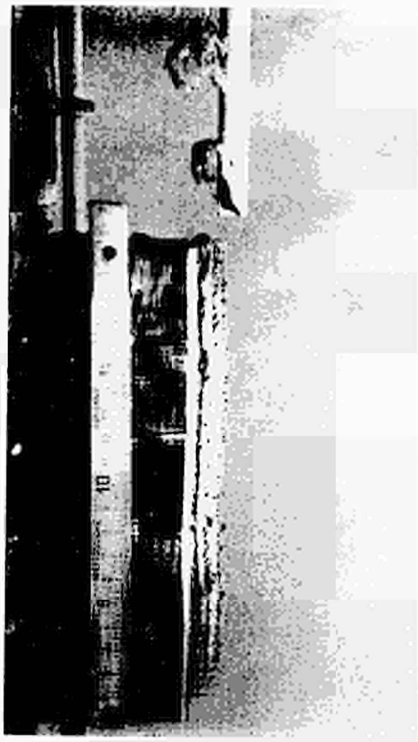


Figure 2 : Drawing of the new tank

Figure 3 : The TD 600 N torch



a) Moving to start position



b) Nozzle at bottom position

Figure 4 : Plasma compass saw in operation cutting 200mm stainless steel

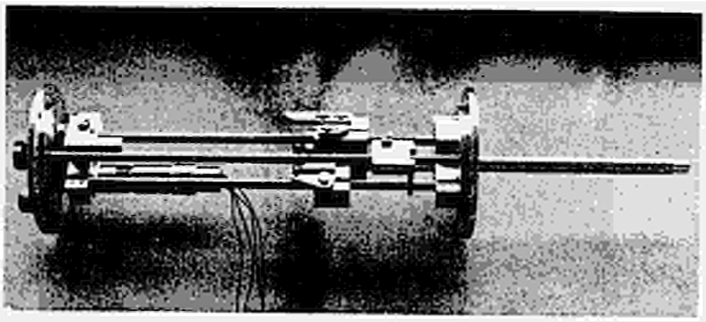


Figure 5 : Details of the moving device for the plasma saw (shown without housing)

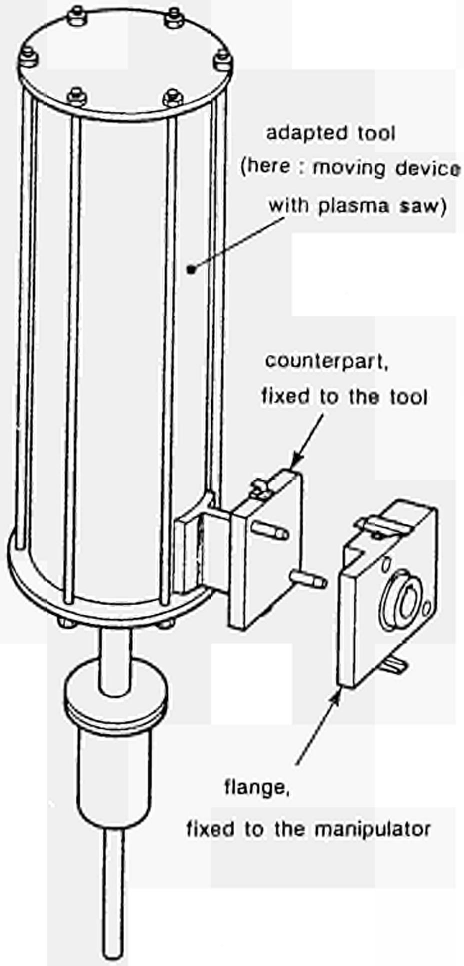


Figure 6 : Principle of connecting different tools to the manipulator

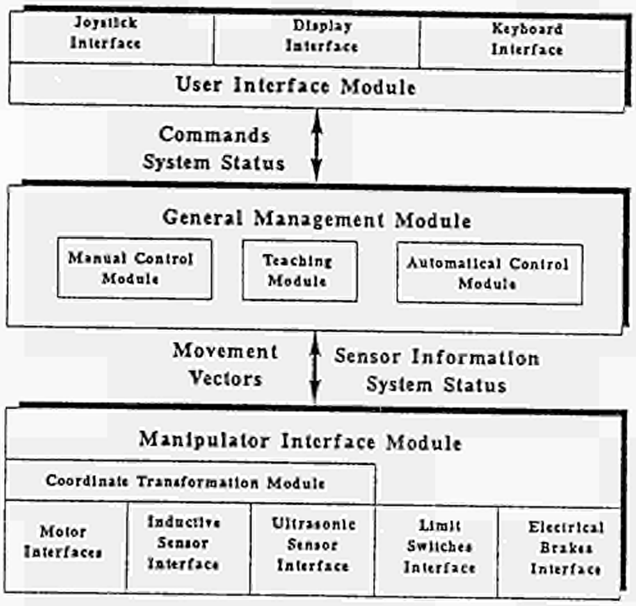


Figure 7 : Software structure and data flow

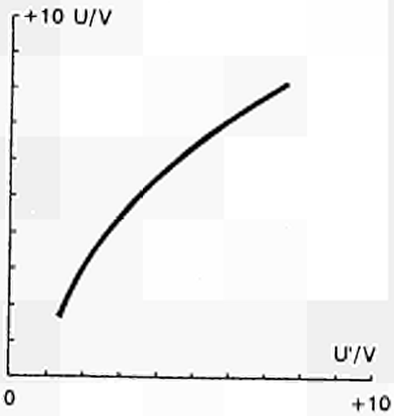


Figure 8 : Trigger values in dependency of nominal distances

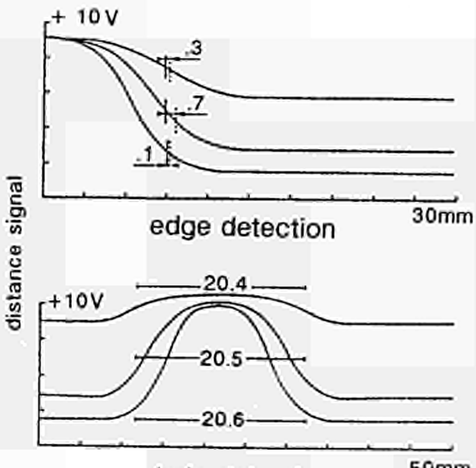


Figure 9 : Accuracy with different nominal distances

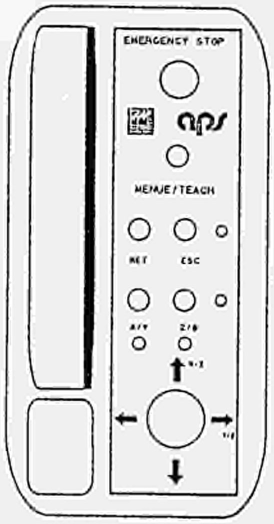


Figure 10 : Layout of manual controller front panel

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: M T CROSS, AEA Wind.
Phone: 44/9467/72432 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 OBTAINED RESULTS

1. Summary of main issues

Significant progress continues to be made on this contract with successful completion of tasks B3, B5 and B6. Due to the rephasing of the WAGR operations programme the requirement for cutting stay-tubes was brought forward in the programme. The consequence of this early requirement is the progress of task B6 and B7. The remote deployment system has been designed, manufactured, commissioned and operated on the reactor using the specially adapted plasma torch. The XYZθ deployment rig was used to locate the plasma torch at distances of up to 10 metres.

A specially developed camera system was also used to carryout the remote inspection of the stay-tubes. This was necessitated because of the early date of the operation. Full mock-up tests have been completed and operators fully trained on using the equipment. In August 1991 a total of 130 tubes were successfully cut and inspected. The first stage of the control software has been completed and a small test rig constructed to test the software.

2. Torch Adaptation Design Study (Task B2)

The design study for the method of attaching, rotating and positioning the plasma torch for accurate location inside the small bore pipework has been successfully completed. The unit selected for the task was an Econocut 30-60 amp system. This unit has a fixed current output of either 30 or 60 amperes. The torch selected was an air plasma unit which had a compact and relatively simple design. Modifications to the torch were required to enable it to fit down the 37mm ID pipework.

Modifications to the torch head included the removal of a large bulbous section at the rear and also reducing the shaft diameter. A tufnol 'nose' was added to provide additional protection to the nozzle during cutting, also a sleeve was fitted to cover the head during removal to and from the deployment system. Experience during trials had shown the torch nozzle to be susceptible to damage during transfer. The air plasma torch selected is a contact type torch and requires the nozzle to remain in contact with the work piece to provide the earth return circuit. There is no pilot-arc initiation circuit, instead the arc is initiated between the electrode tip and the inside of the nozzle. A High Frequency voltage is required to initiate the breakdown of the air to form a plasma.

A full series of trials have been completed to determine the ideal cutting speeds to ensure complete penetration of the insulated stay-tubes.

A detailed assessment was undertaken to determine the requirement for a remote deployment system. The conclusion of the assessment was then used to allow manufacture of the system. Details of the requirements are given below:

Movement required in 4 axes - X, Y, Z and θ
Remote deployment of cutting torch up to a distance of 10m
Self alignment of cutting torch above the tube
Nozzle protection during transfers
Support framework for deployment tube

The manufacture and commissioning of this deployment system was completed in 1991. The system was fully remote, being controlled by Joystick and viewed by closed circuit television. The control system incorporates a number of interlocks to minimise the risk of cutting in the wrong location.

A tufnol sleeve was also added to act as insulation between the torch head and the torch holder. During early cutting trials it was found that double arcing tended to occur between the threaded connections at the top of the torch head and the holder. The current jumped across the air gap to short circuit the torch tip. The tufnol sleeve was extended and this has now cured the problem.

Prior to its operational use a comprehensive training programme was undertaken to ensure operators were fully conversant with the system. An operating manual and technical descriptive manual were also produced in support of the training. A second version of the modified cutting torch has also been successfully tested, this torch incorporates a built-in acoustic sensor and its associated cabling.

3. Examination of cutting parameters (Task B3)

A full assessment of the cutting parameters has been completed based on the Econocut system. A 30 ampere current is used with an air-flow rate of 200 litres/min at 5 bar pressure. The variables which have been defined as affecting cut quality are initial dwell period, cutting speed and the amount of cut overlap. These cut parameters were examined for two types of pipe. The first type was mild steel with an internal diameter (ID) of 67mm and a 5mm wall thickness. The second pipe included an insulating foil wrapped around the outside, the foil consisted of ten layers of dimple foil 0.1mm thick x 1.2mm apart.

The dwell period selected for insulated pipe was between 5-10 seconds. A cutting speed of 200mm/min was used for non-insulated pipework and a reduced speed of 150mm/min was used for insulated pipework. The recommended overlap for insulated pipe is between 5°-10°. With respect to nozzle wear it was found that over 25 cuts could be satisfactorily performed without the need for replacement. However, due to the relative inexpense of these items it was recommended that the nozzle and electrode be changed whenever the unit was removed from the deployment system (approximately every 12 cuts). This had the additional advantage of reducing potential dose-uptake due to nozzle failure and also reducing operational times.

4. Control System Development (Task B4)

The development of the control system has been defined in two stages:

Stage (I) Development of a system capable of monitoring and storing signal data during cutting operations.

Stage (II) Closing the control loop to produce an operational feedback system.

The Stage (I) work has been completed, the system comprises an Intel multibus computer system and associated signal conditioning circuiting. Interaction with the user is via a VDU and keyboard. The writing of the system software has been completed this year, although a significantly longer period of time than expected was required. This additional time reflects the nature of writing low level software that interfaces with hardware. Initial trials have shown that High Frequency voltages generated during arc-striking are causing interference to the control system. This was anticipated and work is in-hand to identify the exact nature of this problem.

5. Testing of Full Scale Mock-up (Task B6)

Due to the early requirement of the cutting equipment the full scale testing on a mock-up was brought forward in the programme and completed this year. The programme involved the cutting of small and large diameter tubes using the air-plasma torch deployed from the remote XYZ frame. Specially designed inspection cameras were also used on the mock-up and incorporated into the control system.

Due to the short timescales it was not possible at this stage to use the acoustic monitoring technique to assess the cut. The inspection was performed using two purpose built cameras which are also deployed from the XYZ system and are controlled from the desk console. The initial specification for a 20 metre plasma cable was not achieved due to inconsistant arc-striking. Several measures were taken to improve the consistency resulting in a reduction in cable length to 13 metres and the removal of some 2m of screening.

6. Operational Performance

During the period between July 1991 - August 1991 the complete cutting system was deployed on the WAGR pile cap and was engaged in stay tube cutting operations. A total of 100 stay tube (67mm ID) and 30 flux scanning tubes (37mm ID) were cut without any major problems. In addition to a fixed focus camera positioned on the Z axis a dedicated camera system consisting of 7 cameras was also employed to give a total view of operations in all locations.

The XYZθ deployment system was successively moved to different positions along the pile cap floor to allow direct access to the standpipes. As stated previously during each relocation of the cutting torch the nozzle tip was routinely replaced to avoid excessive wear. Torch failures did in fact occur on two occasions and each failure was due directly to mis-alignment of the torch inside the pipes being cut. Both failures occurred in the small diameter tube where the clearance between the bore of the pipe and the torch was only 3mm. This resulted in arcing between the rear exposed conductor of the torch and the wall of the pipe resulting in complete destruction of the torch.

During the actual cutting operations several recordings were made of the acoustic signals produced during cutting. These recordings will now form part of a library of recordings which will be used to develop the alogrithm for the control system.

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 - May 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

Cutting tests were carried out during the period from July '91 to December '91, so as to verify the various constituents (rubber and plastic for vulcanization and steel cable) making up the diamond wire.

Progress and results

1. Cutting tests on non-contaminated concrete structures (B.6.)
   The cutting tests on the highly reinforced concrete structure (test mock-up) started in September. The necessary quantity of water for cutting reinforced concrete is usually 2000/2500 liters per hour. During these tests, coolant consumption was reduced to 850 liters per hour by adding some emulsion (2%) to the water.
   The cutting tests are going on and it seems that further reduction of coolant consumption will be achievable.
   Fig. 1 shows the cutting of a concrete structure with a diamond cable.
2. Development of suitable materials for the cable vulcanization (B.2.)
   During cutting, using a diamond wire vulcanized with rubber and reduced coolant, the temperature increased and, as a result, rubber swelled up and the cable broke, whereas with a diamond wire vulcanized with plastic, no relevant problems have happened by now.
3. Selection and improvement of a suitable steel cable (B.4.)
   As far as the steel cable is concerned, no essential difference was found between the steel cable diameter 4.8 mm and the steel cable diameter 4.9 mm. Nevertheless, considering the high mechanical stress the steel cable must support, a 4.9 mm diameter cable will be used.
   Main cutting parameters to-date:

Cutting speed:	Sq. Mts/hours = 1/1.5
Wire speed:	20 Mts/sec.
Diamond wire consumption:	Sq. Mts./L. Mts. = 0.40/1
Coolant consumption:	Lts./Sq. Mts. = 1275/1

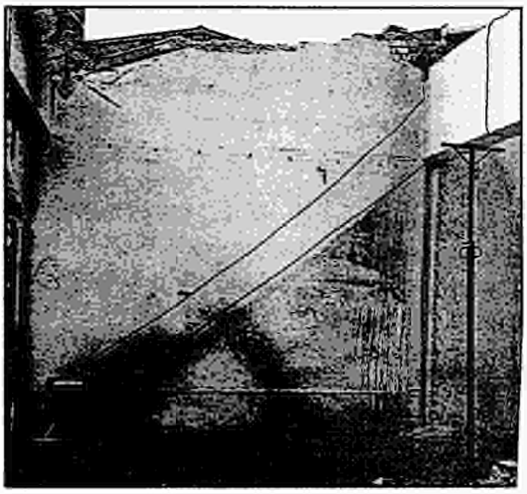


Figure 1: Cutting of a concrete structure with a diamond cable.

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 - August 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

Summary of main issues

Phase B1 of the Programme, which concerns assessment of the beam-workpiece interaction for CO lasers, and comparisons with CO_2 lasers, has almost been completed. In the later Phase B2, direct assessments of the cutting capabilities of the two laser types will be made, but the emphasis of the first phase is to make some fundamental studies on two of the key processes which influence the beam-workpiece interaction, namely the Fresnel absorption of the beam at the molten metal surface, and the plasma formation at the beam-workpiece interaction point. The aim is to demonstrate experimentally that, at the shorter CO (Cf. CO_2) wavelength, there is a higher absorption of the beam (which should give more efficient cutting), a reduced tendency to form plasma (which is deleterious because it blocks the beam), and a lower absorption of the beam in the plasma. It is acknowledged that the experimental work is challenging, but if successful, it is expected to give rather direct guidance on the advantages of CO processing for realistic beam-workpiece interaction conditions. The progress to date can be summarised as follows:-

Detailed beam profiling of the AEA and DLR CO lasers has been carried out (important because focused spot intensities govern strongly the beam-workpiece interaction process). AEA has set up a temporary 20 m beam line to bring its CO beam into the workstation of its 5 kW CO_2 laser; the CO_2 laser beam is being used to create 'real world' conditions of plasma and molten metal, the absorption of which at 5 μm is being assessed using the CO radiation as probe beam. The beam line is also being used to assess the influence of gaseous contaminants in the beam line on the transmitted laser power and beam quality, since this will yield information relevant to decommissioning applications. DLR has commenced studies of plasma ignition for CO and CO_2 beams incident on stainless steel. The equipment for the measurement of beam reflectivity from solid and liquid (hot) metal surfaces has been constructed. The main diagnostic is an integrating Ulbricht sphere. Calculations have been carried out, resulting in a specification of a silicon lens with small aberrations so that plasma ignition threshold is expected to be attainable with the DLR CO laser.

Progress and results

1. Work at AEA Technology

Long distance laser beam profile measurements (B1) This is an important experiment because long distance laser beam propagation is highly relevant to decommissioning applications. Two factors are being investigated: the laser beam mode (and far-field divergence), which is relevant because of focusing and beam line dimensions considerations, and the possible effects due to absorption of the laser radiation by vapours in the beam line. This absorption can be caused especially by water vapour, because the CO laser radiation wavelength range lies inside a very strong infrared absorption band of the water molecule, but other species of (organic) molecules may also cause significant absorption. It is already known from previous work with CO_2 lasers that, for high beam power ($\geq$ a few 100W), absorption by vapours will not only reduce the beam power available at the workpiece, but, more importantly, cause thermal blooming of the laser beam. Thermal blooming is the effect of convective air currents in the beam line, with associated refractive index variations, driven by the heat from the laser beam due to absorption. Thermal blooming will affect the quality of the laser beam mode severely and can, in extreme cases, increase the divergence of the beam to the extent that almost no power is available at the workpiece. In the case of a CO_2 laser, absorption and thermal blooming can be prevented by purging the beam line with dry, clean air at moderate flow velocities (about 0.5m/s).

A 20m beam line, linking the CO laser to one of the workstations of the 5kW CO_2 laser, has been set up for the experiments. It contains a collimating telescope consisting of a pair of concave mirrors. The near field beam profiling results, obtained with a Laserscope UFF100 and shown in Figure 1, indicate a divergence of 3.34 mrad, averaged over a variety of laser beam powers. Since the position of the beam waist has not yet been established, the data has been analysed in two ways:

First the data points were fitted by the method of least squares with a hyperbola representing the envelope of a zero order mode beam. The variable parameters were wavelength (λ), waist position and waist diameter (φ). The usual diffraction relationship had to be satisfied: $\alpha\varphi = 4\lambda/\pi$, where α is the full angle beam divergence. A solution was found for λ=5.5μm where the waist was predicted to lie inside the laser. If this is indeed the case, the laser is operating very close to the diffraction limit. Secondly, it was assumed that the waist lies at the output window and therefore has an inferred diameter of 6mm. The (divergence x waist) product is now 10 μm, i.e. approximately 1.6 times the diffraction limit at λ=5μm.

Far field intensity measurements (also shown in Figure 1) were made by placing the Laserscope near the focus position of a 400 mm focal length spherical collimating mirror. Figure 2 shows the spot size as a function of focus position. Taking into account effects from astigmatism, the results indicate a (divergencexwaist) product of 13.3μm. This is approximately twice the diffraction limit, a result which is fairly consistent with the second method of interpreting the near-field profiles above.

The beneficial effect of purging the beam line with clean, dry air was also assessed. For these experiments, two pyroelectric laser power meters were used - one as near to the laser output window as possible and the other at a position about 4m away from the end of the 20m beam line i.e. at a distance of 16m. The laser power was varied between 20 and 110W by means of the Xe content of the laser gas and the HV potential difference across the laser discharge (it was known from previous experiments that these variations have only a very small, negligible effect on the laser beam mode), and the power entering and emerging from the beam line was measured with and without air purge. Figure 3 shows the results, indicating that there is a significant improvement resulting from purging (the dry air supply had a dewpoint of about -25°C, compared to an estimated 10°C for environmental air in the laboratory). The reduction in power even with dry air is due to absorption by the mirrors in the beam line and beam collimator (seven in total). Figure 4 shows the measured beam diameter (in one dimension), using a pyroelectric detector masked with a small hole, scanned across the laser beam at the end of the beam line. No significant changes in beam diameter could be measured, indicating that the laser power is too low for thermal blooming to occur. The absorption of the CO laser radiation takes place within the ν_2 vibration-rotation band of the water molecule. This band is centred at λ=6.25μm but still has a significant absorption within the CO laser radiation range i.e. λ=4.9-5.5μm. Within this range, the average half-transmittance water vapour density is approximately 0.14 gram/cm² /1/. For a 16m long section, this is equivalent to 0.09 grams of water per litre of air. At a dewpoint of 10°C, the equivalent density of water vapour in the laboratory air is about 0.01gm/l, and at a dewpoint of -25°C, about 0.5×10^{-3}gm/l for the dry air supply. Hence, according to Goody /1/, the absorbed proportion of the radiation by the dry air is very small (less than 1%), but about 20% by the 'wet' laboratory air (in the model by Goody /1/, although defining a half-transmittance value, the transmission does not show a simple inversely exponentially dependence on the water content of the air). This is in fair agreement with the results as shown in Figure 3.

Laser beam absorption by liquid metal and in laser-produced plasma (B1)

Using the 20m beam line, the CO laser beam can be used to 'probe' the absorption by a plasma created above a metal surface irradiated by the 5kW CO_2 laser beam, and compared to the absorption of a second probe beam from the CO_2 laser. The same equipment can easily be adapted to carry out experiments to measure the absorption by liquid metal, created by the high power CO_2 laser beam.

The theory of Fresnel absorption by metal surfaces of radiation predicts that for clean, polished surfaces the absorption increases with radiation wavelength. However, in a 'real-world' cut kerf, the relevant surface will be at high temperature, probably molten, and at least partially oxidised. Theoretical predictions under these conditions become extremely difficult and it is hence very important to attempt experimental measurements. In the case of absorption of the laser beam by plasma at the workpiece, the process can be described by the Beer law, which states that the magnitude of the transmitted radiation is inversely exponentially dependent on the distance travelled through the plasma. The absorption mechanism is predominantly inverse Brehmsstrahlung, for which it can be shown that a x2 reduction in laser wavelength suggests a x4 reduction in absorption coefficient. However, a detailed prediction in a 'real-world' plasma will be difficult and it is considered important to attempt experimental measurements of absorption at 10.6μm and 5μm.

Measurements of Laser Beam Absorptivity in Laser Plasma (B1) Investigations of the absorptivity of CO_2 and CO laser beams in plasma produced above a CO_2 laser weld have been undertaken. Welds were produced in steel and aluminium alloy by focusing the beam from Culham's 5kW CO_2 laser CL5 with an on-axis spherical mirror, having a focal length 300mm, on to the surface of the metal pipes which were held in a rotary chuck. Low power probe beams from the CO_2 and CO lasers were focused in the plasma region above the workpiece, and the transmitted laser beam power was measured by a power meter. The probe beam could be traversed through the plasma over a distance of about 5mm from the weld centreline. First, the beam from the CO laser was used to probe the plasma created by a 4kW CO_2 laser weld in mild steel. The power in the probe beam was about 5W and a CaF_2 filter, which transmits ~95% of the radiation at 5μm and only ~10% at 10.6μm, was inserted in front of the power meter to reduce the scattered CO_2 laser beam signal. Nevertheless, a satisfactory signal to noise ratio has not yet been achieved and further work is still required. Second, a small fraction of the CL5 CO_2 laser beam was used to probe the plasma. Aluminium alloy pipe was also used instead of mild steel to increase the plasma. A reasonable signal to noise ratio was achieved which enabled the measurement of some absorption of the CO_2 laser probe beam by the weld plasma. However, to obtain a detectable effect, the probe beam had to be directed very close (~2mm) to the surface of the workpiece which caused some interruption of the probe beam by the rotating pipe. Therefore, a series of CO_2 laser probe signals versus probe beam position were recorded with and without the pipe in place and with and without a CO_2 laser weld taking place. From the results so obtained, it is estimated that the plasma absorptivity at 10μm in the aluminium alloy welding plasma decreases from ~20% to 10% as the height above the workpiece is increased from ~1 to ~2mm.

The duration of B1 has had to be extended until April 1992, partly because of the challenge of the work, and partly because of staff changes. However, it is believed its successful completion is attainable. For the same reasons, commencement of B2 has been delayed but its completion on schedule is still expected.

2. Work at DLR

CO Laser Beam Quality Investigations (B1) In order to get comparable beam characteristics of CO and CO_2 lasers at beam-workpiece interaction experiments, detailed profile measurements of the CO laser beam have been carried out. Table I shows a summary of the results obtained. Different beam diameters and corresponding beam powers can be achieved using apertures of different diameters inside the resonator. The beam quality is characterised by the factor K, which is unity for a Gaussian beam profile. Subtracting the calculated spherical aberration of the optical system from the measured focal spot diameter yields a corrected factor K_{CORR} which is a measure of the beam quality alone. In the case of the multiline CO laser beam the chromatic aberration is still included, which is essential only in the cases of lenses of short focal length. Towards the end of the year, measurements of 'best-form' silicon lenses for 5μm wavelength have been carried out. Compared to the CaF_2 lenses an increase of the maximum intensity up to more than 10^6W/cm^2 can be expected at a beam diameter of 12mm and a beam power of 550W.

Plasma Ignition Experiments (B1) In order to determine the different behaviour of a 5μm (CO laser) and 10.6μm (CO_2 laser) wavelength beam interacting with metal surfaces using argon as a shielding gas the experimental setup shown in Figure 5 has been established. It is well known from material processing with CO_2 lasers that deep welding in stainless steel occurs at focal spot intensities of several 10^7W/cm^2, associated with the appearance of a blue shining argon plasma. This argon plasma enhances the coupling efficiency of the laser radiation to the metal surface, but only up to intensities of several $10^7 W/cm^2$ when the plasma separates from the metal surface, absorbs the laser radiation and prevents it from further interaction with the metal surface. Figure 6 shows this behaviour as a result of theoretical calculations by Beyer /2/. As the absorption coefficient of the plasma is approximately proportional to the inverse square of the wavelength of the laser radiation, the absorption of the CO laser radiation should be one quarter of that of the CO_2 laser, and also the onset of plasma ignition is shifted to higher focal intensities. The processing range of the CO laser is also indicated in Figure 6 - it is estimated to be shifted to intensities one order of magnitude higher than for CO_2 lasers. In the experiments, the target material was stainless steel moving at a speed of about 5cm/s. The focusing lens had a focal length of 63.5mm and the shielding gas was argon (the beam properties are given in Table I). Using the CO_2 laser, argon plasma formation and deep welding could be observed at peak power densities of at least $2.5 \times 10^6 W/cm^2$. Using the CO laser, at the same power densities and similar focal spot size the deep welding effect occurred, but no argon plasma could be observed. The bright yellow-white light emitted from above the metal surface seemed to be produced by excited metal vapour. Figure 7 shows a polished cross-section of the CO_2 and CO welding seems respectively. Using the new silicon lenses described above, it should be possible to attain the argon plasma limit also for the CO laser wavelengths. The experiments will be done at the beginning of 1992.

Absorption measurements (B1) Literature studies have been carried out to find experimental data of the absorption coefficient of molten metal surfaces and to choose an adequate experimental method. Figure 8 shows a schematic view of the experimental arrangement. The interaction between the laser beam and the target takes place in the interior of an Ulbricht-sphere. The reflected laser light is integrated by this sphere and can be measured by means of a detector. An interference filter stops any radiation except the wavelengths of the CO laser. The laser operates in a single pulse mode so that starting with a solid surface of the target the variation of the reflectivity can be observed when the surface liquefies. In addition, by means of the thermocouple the part of the laser beam which is absorbed by the target can be evaluated. Theoretical calculations considering the available performance data of the CO laser resulted in target dimensions of 5mm diameter and 2.5mm length, to achieve a temperature rise of several 10° K during laser interaction, which can be measured accurately. The experimental arrangement has been constructed and finished during the second half of the year, but due to a leakage problem in the laser system the start of the measurements had to be delayed to the beginning of 1992.

References

/1/ GOODY, R.M., Atmospheric Radiation I Theoretical Basis, Clarendon Press, Oxford (1964), pages 157 and 189.

/2/ BEYER, E., Einfluss des laserinduzierten Plasmas beim Schweissen mit CO_2 Lasern, Schweisstechnische Forschungsberichte Band 2, DVS, Düsseldorf (1985).

Table I. Beam quality of the CO laser. DLR's CO laser characteristic beam data for different optical systems. R_B: beam diameter. R_F: focus diameter. I_{MAX}: maximum focal intensity. I_{MIT}: Medium focal intensity. Sph. Aberr.: calculated spherical aberration. K_{CORR}:beam quality factor K corrected for spherical and chromatic aberration. $K=(\lambda f/\pi)/R_F R_B$.

Optics	P_{laser}	R_B (mm)	R_F (μm)	I_{MAX} (W/cm^2)	I_{MIT} (W/cm^2)	K	Sph.Aberr. (μm)	K_{CORR}
mirror f=150mm	100W TEM_{00}	3.3 ±5%	94	8.5×10^5 ±5%	3.5×10^5 ±3%	0.78		
lens f=127mm			79	1.2×10^6 ±5%	5.0×10^5 ±3%	0.79	0.6	0.80
lens f=63.5mm			55	2.8×10^6 ±5%	1.0×10^6 ±3%	0.56	2.5	0.59
lens f=63.5mm	350W medium order	6 ±5%	129	1.6×10^6 ±5%	6.8×10^5 ±3%	0.13	14	0.15
lens (silicon) f=50mm			90	2.8×10^6 ±5%	1.4×10^6 ±3%	0.15	3.5	0.16
mirror f=150mm	300W high order	7.75 ±5%	290	2.0×10^5 ±5%	1.5×10^5 ±3%	0.11		
lens f=63.5mm			140	1.0×10^6 ±5%	6.5×10^5 ±3%	0.095	28	0.12

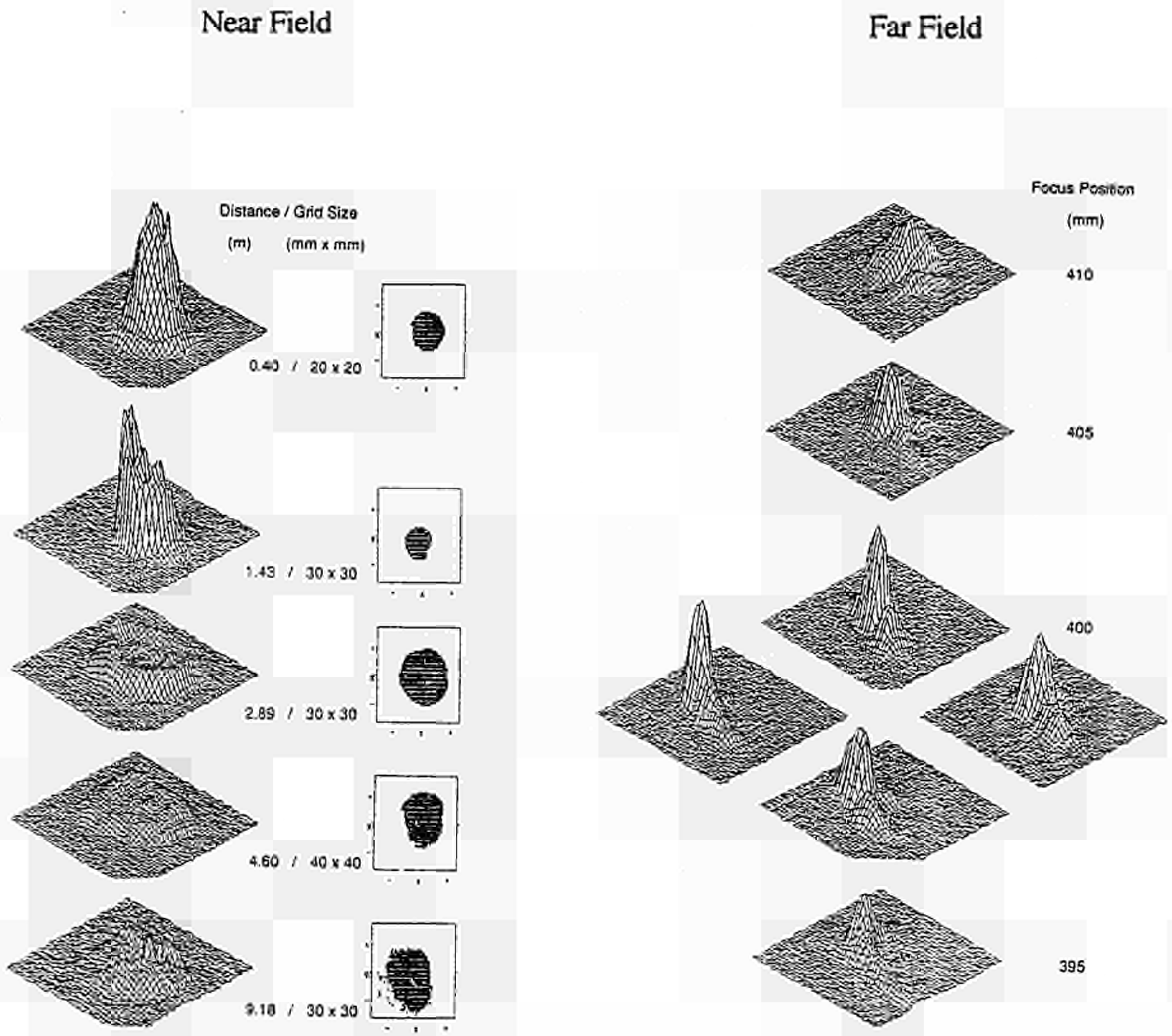


Figure 1. Near field and far field laser radial intensity profiles and $1/e^2$ intensity contours. In the far field profiles, the grid size is 1mm×1mm.

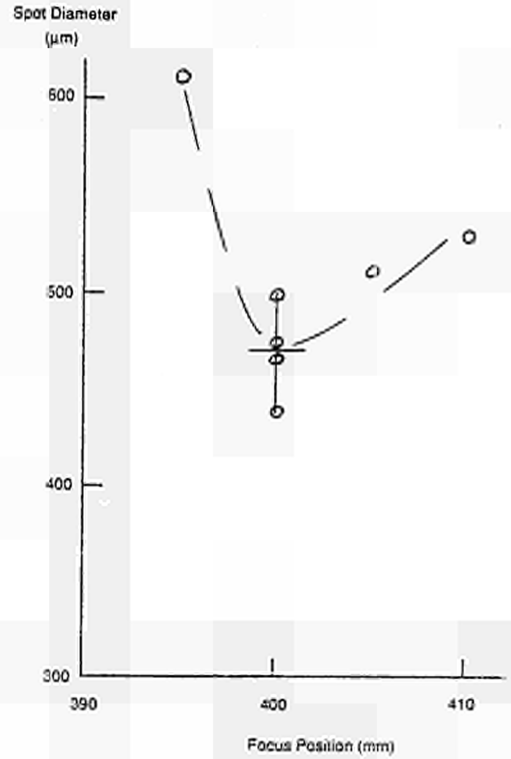


Figure 2. CO laser focal spot size versus focus position.

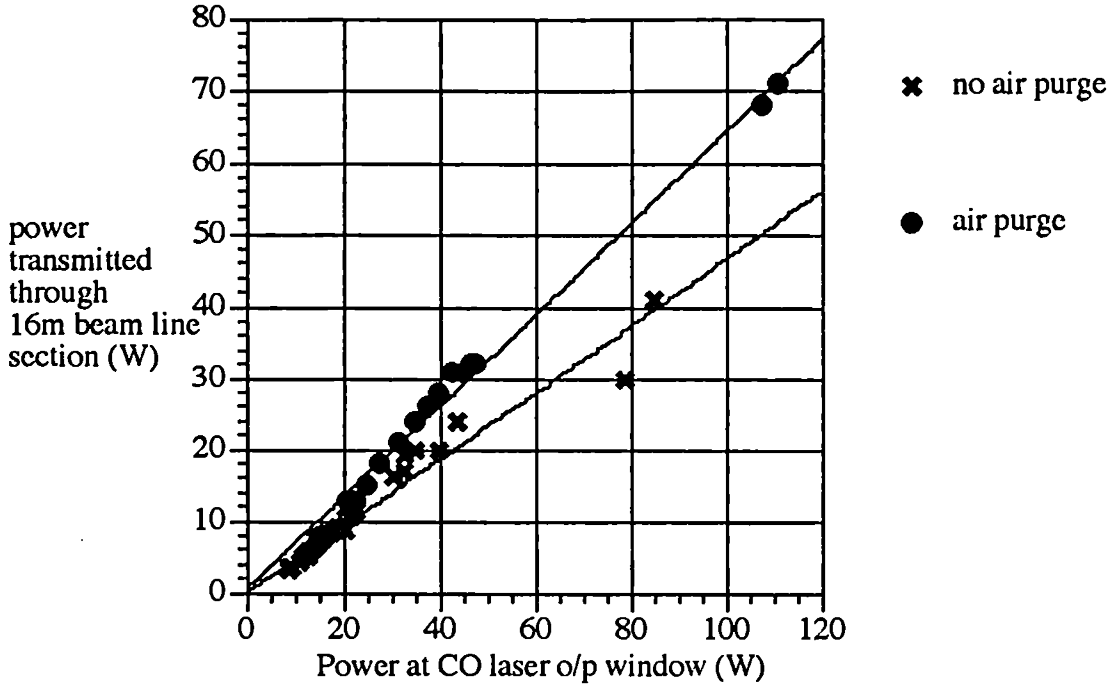


Figure 3. The transmitted power through a 16m long section of the beam line, with and without dry air purge.

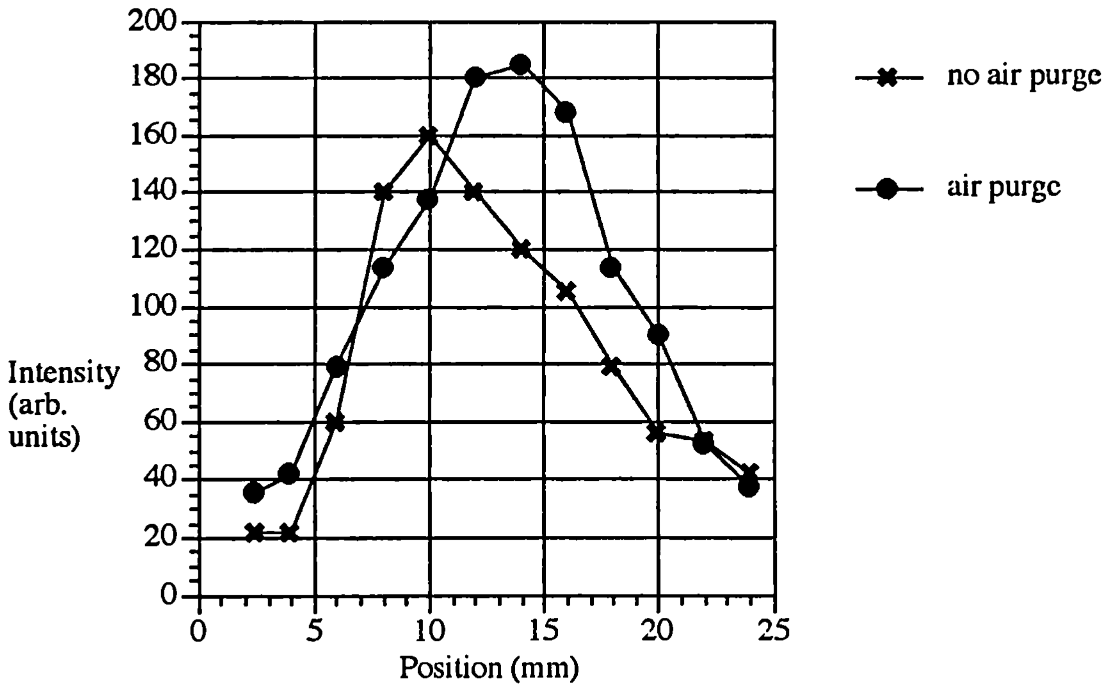


Figure 4. The measured CO laser beam profile after having traversed a 16m long section of beam line, with and without dry air purge.

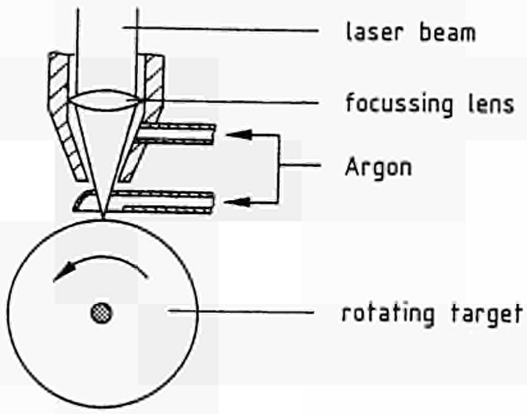


Figure 5. Experimental setup (schematic) for plasma ignition measurements.

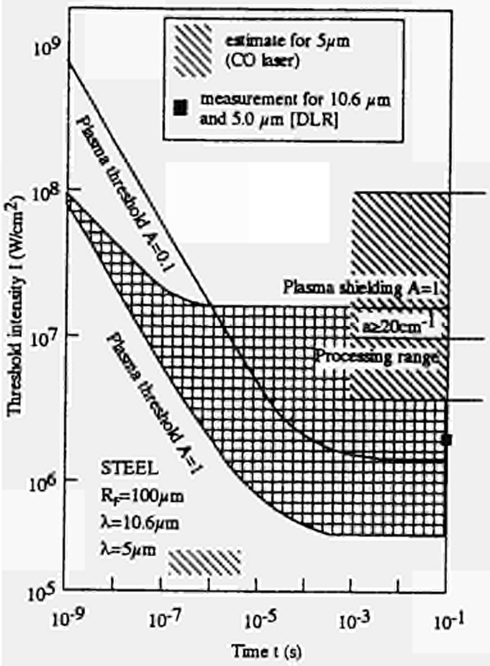


Figure 6. Plasma threshold intensity for steel/argon, calculated after Beyer /2/.

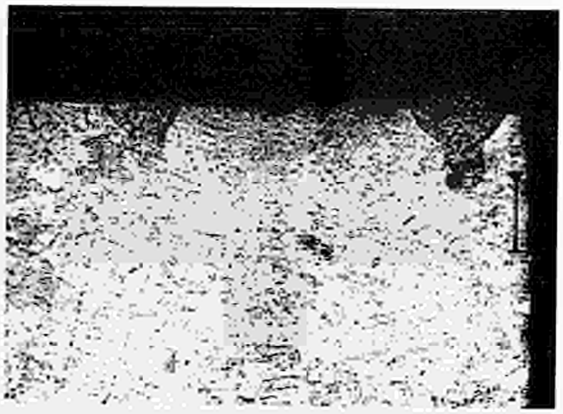


Figure 7. Polished cross-section of deep weld seems in stainless steel. Length of scale (top right): 700μm. Left: CO_2 laser; beam power 470W, max. intensity 2.5×10^6 W/cm^2; right: CO laser, beam power 540W, max. intensity 2.5×10^6W/cm^2.

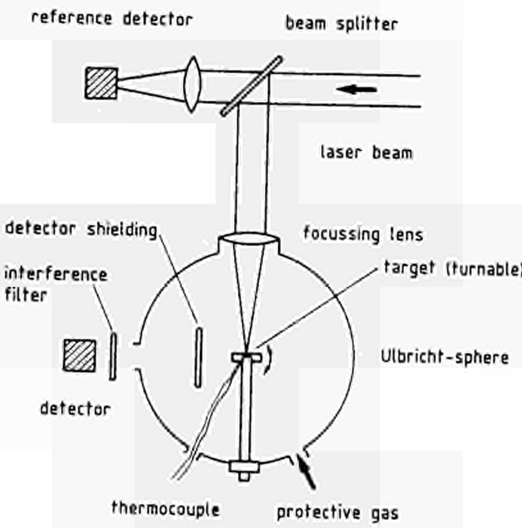


Figure 8. Schematic view of the Ulbricht integrating sphere experimental setup for absorption measurements.

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 - April 1992
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 obtained results

Summary of main issues

The cutting device has been optimized in order to use less plastic (a plastic ring supports the hollow-shaped charge), less explosive (the quantity of explosive must be constant behind the dihedral) and a steel dihedral instead of a copper one (copper is a poison for the re-melting of cut steel).

The agreement files have been drawn up and are now examined by UDIN before sending to the authority.

Experimental shootings have been carried out on a full-sized mock-up and on confining mocks-up.

The calculations of the confining box to be set around the hollow charges are now on the agenda.

Progress and results

B.1. Definition of cutting-power of dihedral-shaped charges.

In order to avoid a copper deposit which is a poison for the re-melting of cut steel, new cutting devices using steel dihedral have been adapted from the ones which were defined in 1990.

The following table gives the correspondence between the thickness of mild steel to be cut and the main dimensions of the ring section of the hollow-charge.

	Cutting device				
Thickness of mild steel tobe cut	Reference mm	height mm	width mm	Stand-off mm	Dihedral thickness mm
10	L 10	20.5	17	8.5	1
15	L 15	26.5	22	11	1.5
20	L 20	32.5	32.5	13.5	1.5

The results of the experimental shootings are the following :

Reference of the cutting device	Thickness of the mild steel to be cut (mm)	Thickness cut
L 10	10	12
L 15	15	16
L 20	20	21

These results show that the hollow charges have correct dimensions and confirm that the performances of copper and steel are very close.

B.1.2. Establisment of the agreement files

The safety files have been drawn up and are now examined by UDIN before sending to the different competent departements. The main difficulty is to confine the gases and the shock-wave produced by the detonation.

The agreement of the Nitroroc N, used as explosive for the cutting fuses, has been granted by the national authority : INERIS.

B.1.4. Calculation of the minimum quantity of explosive (first back shield)

A confining shield must be set around the cutting devices in order to confine the gases and the plastic projections produced by the explosive charges.

Experimental shootings have been carried out using small charges in confining mocks-up, the first two mocks-up broke into pieces, test proceed with the third one. The confining shield will obviously be heavy and bulky.

Calculations of the confining shield will shortly be made according to three cutting cases : 15 mm, 20 mm and 25 mm steel thicknesses.

B.2. Pre-test series with bounded steel samples

Tests have been carried out using prototype hollow-charges on a full-sized mock-up of a G2's pipe (length : 5 m, diameter : 1.2 m, thickness : 20 mm, material : mild steel).

The cutting occurred in good conditions but the effects on the environment have been very important : a 3 mm thick steel shield placed at 1 m behind the hollow-charge, was bored through in plastic projections, the static pressures mesured at 4 m and 6 m, range from 0.07 bar to 0.4 bar and about 3 Nm^3 of gases were given off.

The decontaminating effect could not be measured. An area centered on the explosive charge and being about 5 cm wide, is affected by the shock wave. It is planned on winding detonating fuses up the G2 reactor's pipes in order to get a continuous result.

B.3. Definition of pyrotechnic devices

The optimized length of the hollow-shaped charges is one half circumference. The Nitroroc N which is a liquid explosive has given the best results.

3.10. UNDERWATER LASER CUTTING OF METAL STRUCTURES

Contractors: CEA-FAR, Radius
Contract No.: FI2D-0047
Work Period: January 1991 - December 1992
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 AND OBTAINED RESULTS

SUMMARY OF MAIN ISSUES

The experimental set-up for underwater cutting must correspond to the criteria of pression (10 meters water column) and distance from the cutting head to the laser source. To meet these boundary conditions, we have adopted the experiment across different floors of our building, as represented in fig. 1. The 500 W CO_2-laser is installed on the second floor. The experimentation vessel and control cabinet of the laser are located on ground level. The first tests have run with this configuration (originally, a 1.200 W laser was scheduled). In order to explore further the full range of plate thicknesses, a 5 kW laser will be operational at the beginning of 1992.

During the first year, we were mainly concerned by mechanical design problems : preparing the experimental site, designing the watertight cutting head as wel as the installation of these devices on the site in our facilities of CEA/STA Saclay. More specific, preliminary cutting test have been run under 0.5m water and have been compared with cuts realized under ambient conditions on thin plates of stainless steel.

PROGRESS AND RESULTS

1. B1,B2,B3. Design, realization and test of the cutting head.

The lay-out of the underwater cutting head installed on the experimentation vessel is given in fig. 2. The cutting head does not move transverse to its optical and mechanical axis. Movement of the steel samples is provided by a translation stage. The cutting head is designed using a modular optomechanical concept. This modularity makes maintenance more easy. The following specific elements have been imple-mented :

* Lens holder in a removable cassette.

The cutting head is provided with two cassette slots, corresponding to resp. 127 and 250 mm focal length. In this way, cleaning and inspection of the lenses is easy and the repositioning in the beam delivery system is reproducible. The optical axis is not affected by the re-installation of the lens, since its positioning relative to the head is fixed by two micrometers.

* Nozzle with assistance gas.

A system with interchangeable coaxial nozzles delivers the gas O_2 for the cutting operation.

* Pneumatic rotatory valve.

This valve serves to isolate the optical part of the cutting head from the surrounding water. When the system is not cutting, the valve is closed.

* Pneumatic axial translation.

In order to focus the beam on the work-piece.

* Manual translation of the nozzle tip.

In order to set the correct relative distance between the nozzle tip and the focal point of the lens.

* Surface follower

This contact surface follower is installed at the end of the cutting head and maintains a constant distance between the nozzle tip and the mechanical sample.

* The overall sealing has been studied to prevent infiltration of water in each of the modules of the cutting head.

After the head was designed and realized, immersion tests have proved both the proper water sealing and the ability to deliver high laser power to the sample.

2. B4. Experimentation vessel (0.4 m3).

In order to simulate an intervention in a water pool of 10m depth, it is necessary to put an external pressure of 1 bar on the water in the vessel. It proved impractical to provide for this pressure by making use of the oxigen cutting gas. On the other hand, the provision to take aerosol samples for further analysis, lead to the pressurization of the vessel through a water column of 10 meter.

The vessel is foreseen with a translation stage (for the cut operation), observation ports and water inlet and outlet connectors.

3. B5. Functional underwater cutting tests.

The explored parameter range for the plate thickness was set between 2 and 8 mm of stainless steel. The main objective was a comparison of the system performances with and without a 0.5 m water layer. At each considered plate thickness, the maximum cutting speed was measured, with the oxygen pressure set at 1 and 2 bars resp. The results are shown in fig. 3. We refer to /1/ where the definition of all parameters is discussed in detail.

Puis : laser power incident on the sample (Watt)
the value was 350 W for all cases.

Foc : lens focal length (mm)
Foc = 250 mm

▲ F : position of focus, relative to the workpiece (mm)
▲ F = 0

ø b : nozzle diameter (mm)
ø b = 1.2 mm

Pr : oxygen pressure (bar)
Pr = 1 & 2 bar

Deb : volumetric flow rate (l/min)
Deb = 25 & 40 l/min

dbp : distance between nozzle and workpiece (mm)
dbp = 1 mm

Vt : translation speed of the workpiece (m/min)

epais : workpiece thickness (mm)
epais = 2, 3, 4, 5, 6 & 8 mm (type 316)

4. Comparison of the cutting performances under water and under ambient, at 1 bar oxygen pressure and 25 l/min.

In fig. 4, the curves present the maximum cutting speed as a function of thickness. For the smaller thicknesses (2 and 3 mm), we can see that this speed is higher under water than in ambient atmosphere. This can be explained by noting that at small mass flow rate the confinement of the oxygen gas jet is better and contributes to a better coupling of the laser power in the kerf.

5. Comparison of the cutting performes under water and under ambient, at 2 bar oxygen pressure and 40 l/min.

Fig. 5 represents the results of the second set of experiments. The oxygen flow rate is sufficient to realize oxy-cutting. The jet confinement under water does not contribute to a higher cutting speed at small thickness. A maximum thickness of 6 mm under water was reached.

6. Visual aspects of the kerf.

Fig. 6 a and b show pictures of the kerfs for the case of cutting at 2 bar, resp. with and without the water layer. One notes the smaller roughness for the samples cut under water. This can be explained by the positive effect of the cooling capacity of the water in combination with the higher confinement of the gas jet.

7. References

/1/ O. MATSUMOTO, M. SUGIHARA, K. MATSUDA, K. MIYA : utting Technique for Reactor Internals by Laser Beam ; Proceedings of the International Topical Meeting on Nuclear and Hozardous Waste Management ; Conference SPECTRUM '90 30/09-04/10/1990 à Knoxville, Tennesse, USA

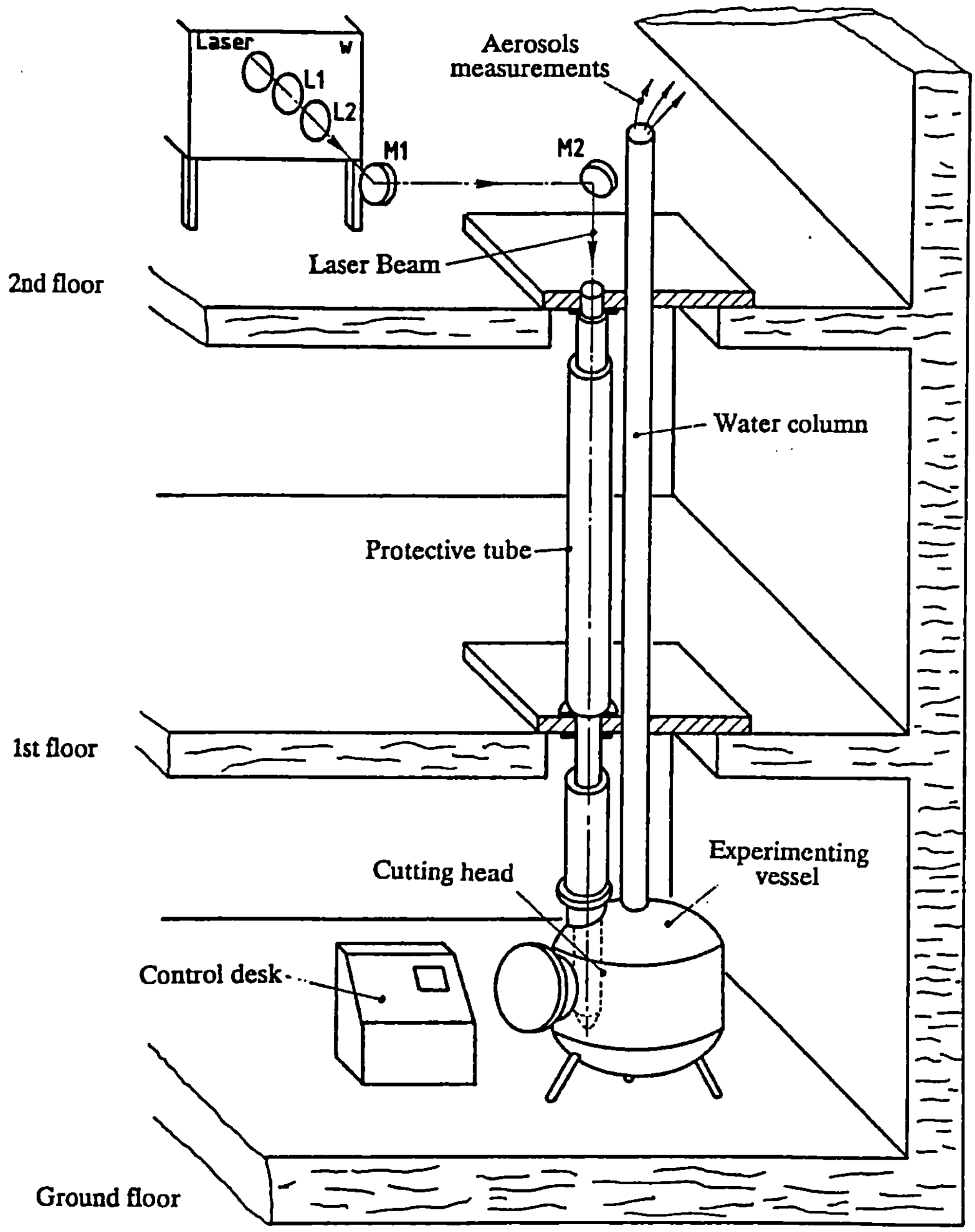


Fig. 1 : General presentation of the system

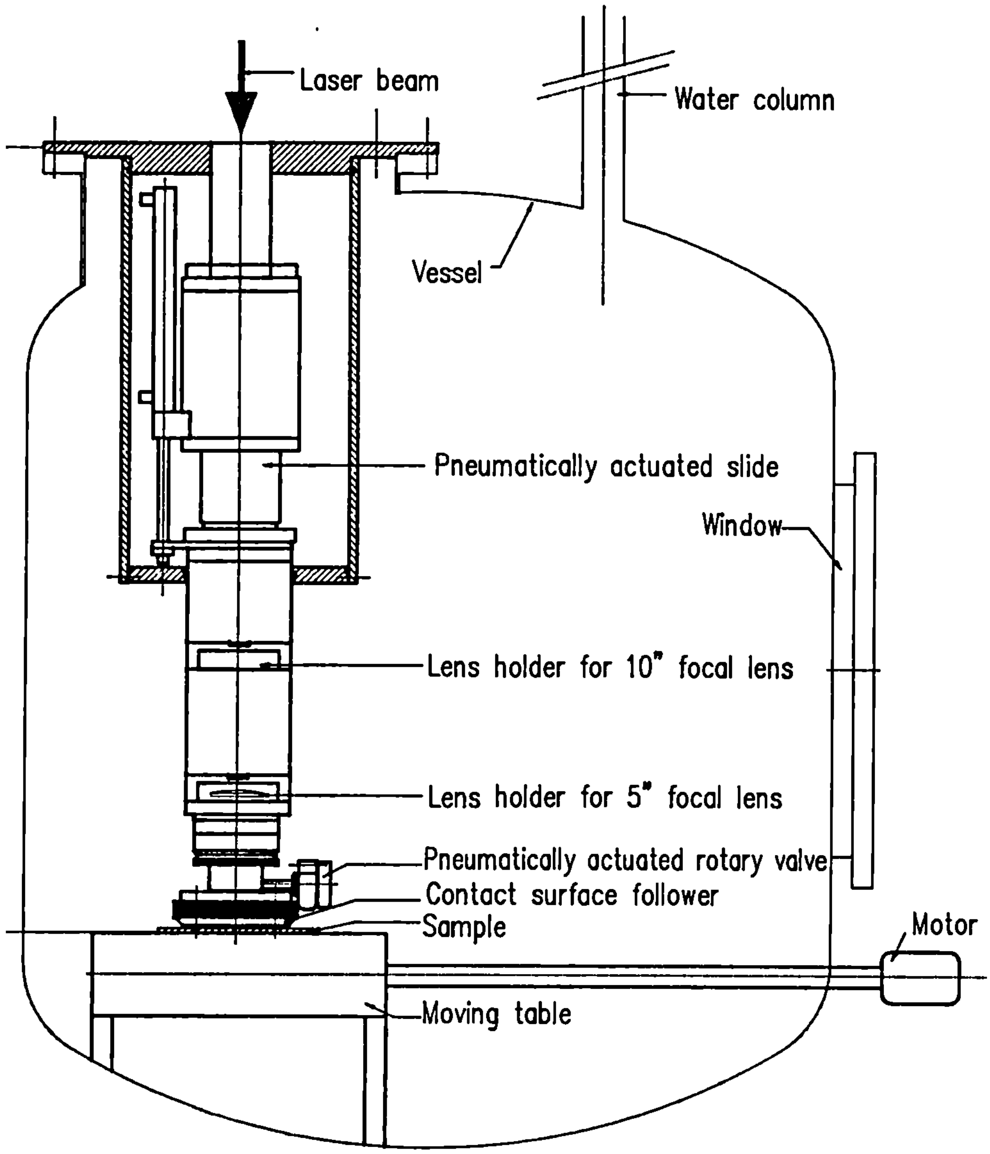


Fig. 2 : Underwater cutting head

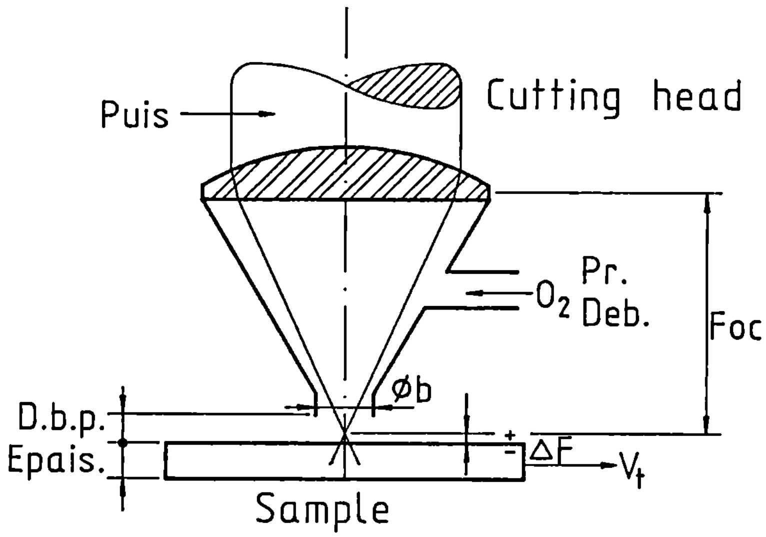


Fig. 3 : Cutting parameters

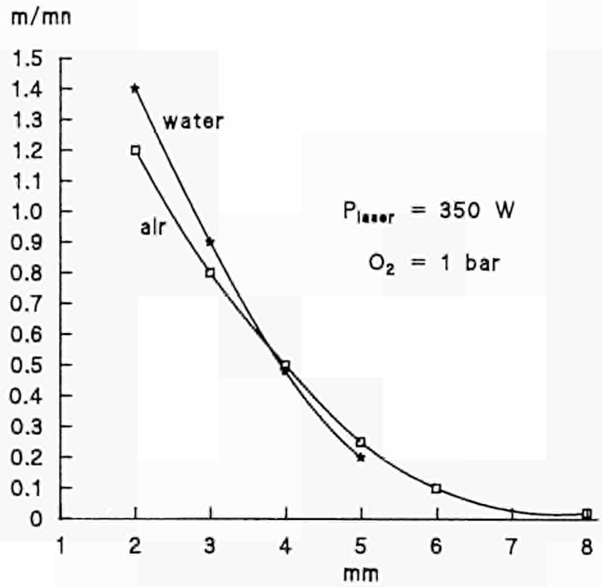


Fig. 4: cutting speed vs. thickness

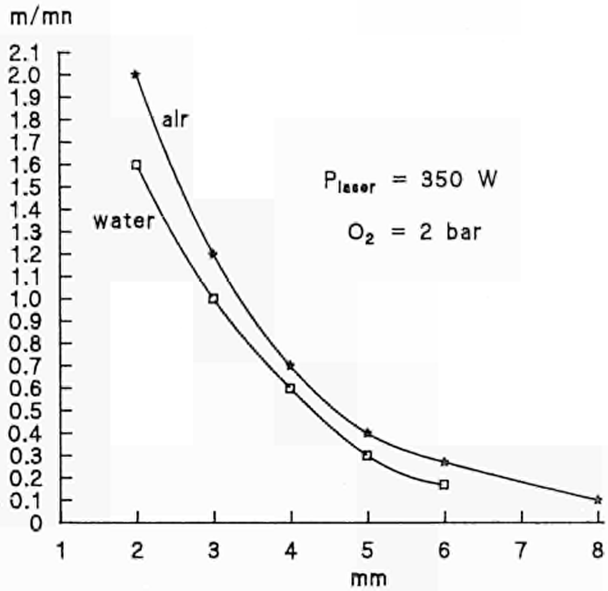
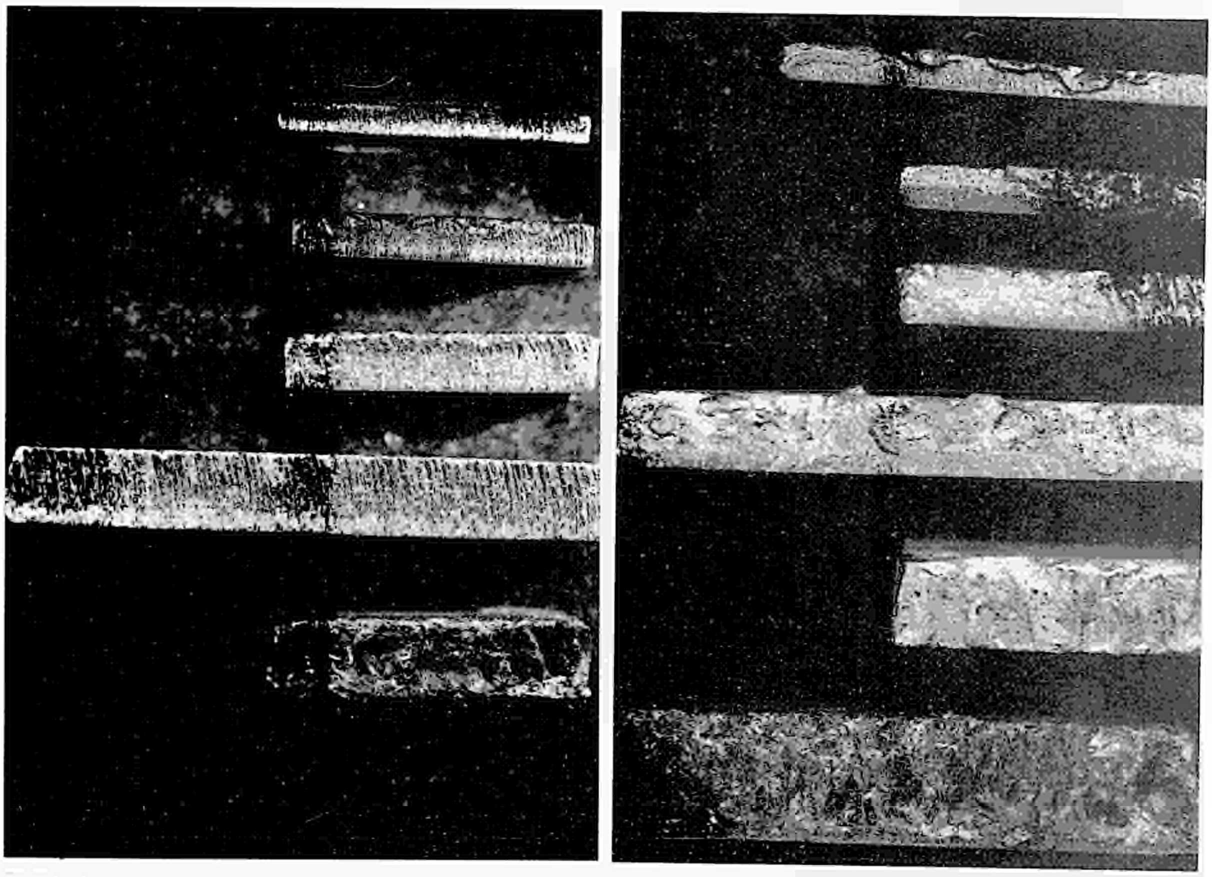


Fig. 5: cutting speed vs. thickness



a) under water b) in air

Fig. 6 : Visual aspects of the cut

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 - December 1992
Coordinator:	R S C 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 contract "Optimisation of Plasma Torch Electrode Design for Nuclear Dismantling Tasks" actually started in August 1991 and has since then progressed roughly according to the work programme. There has however been some programme slippage.

After literature survey and requirements study were completed (B.1. and B.2.), the identification of potential electrode materials is near completion, but final verification of material choices is to be confirmed by materials expert (B.3.)

For plasma arc process modelling, a useful and practical approach to this task has been defined and agreed by all parties. However, the start has been delayed pending completion of synergistic work on arc jets (B.4.).

For the testing of existing methods and electrodes, a database of customers experience with with Hafnium electrodes is being compiled by Goodwin Air Plasma (B.5.), whereas the manufacture and testing of new electrodes is held pending final completion of B.3. (B.6.).

Progress and results

Literature survey (B.1.)

An extensive literature search was carried out by Culham Laboratory in conjunction with the Welding Institute; a shortlist of candidate materials, to be narrowed down further following the requirements definition was prepared independently and then the results combined to give a joint selection. The outcome of this following consideration of the requirements study is given in B.3.

Requirements study and definition (B.2.)

The requirements definition was drawn up taking into account the particular needs of decommissioning. One early decision was to concentrate on air plasma rather than inert gas ones, which was based firmly on decommissioning experience at WAGR. This limited electrode choice to those which are not rapidly oxidised by air at the high plasma temperatures in a torch. A summary of the requirements definition for decommissioning purposes is given below.

1. The electrode and nozzle must be capable of operating with the transferred plasma arc.
2. The plasma gas will be air and possibly oxygen.
3. The developed system must be suitable for use with arc current up to 80 amps maximum with an air-cooled torch, and 300 amps maximum with a water-cooled torch.
4. Initial work will concentrate on electrode materials.
   The electrode material must:
   - be stable in air at the operating temperature, resistant to reaction with oxygen and nitrogen;
   - be resistant to thermal shock;
   - have a low rate of weight loss due to erosion at the operating temperature;
   - have a high electron work factor to facilitate good arc starting and maintenance.
5. Subsequent analysis will be made of alternative nozzle materials (including coatings) and torch design. The nozzle must:
   - be resistant to high temperature corrosion;
   - be resistant to damage from arcing;
   - be resistant to thermal shock;
   - be inadherent to spatter;
   - have a useable life greater than that of the electrode.

Identification of potential electrode materials (B.3.)

The electrode materials selected for further study taking into account the requirements definition and the results of the literature survey are as follows:

a/ Zirconium Hexaboride, ZrB_6, 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.

However, very recent discussions with materials specialists suggest that doped Platinium should be given further consideration. Furthermore, there are concerns that the volatile Ruthenium tetroxide, RuO_4, will rule Ruthenium out as a candidate material. A meeting is to be convened in the near future to decide the final candidate materials shortlist.

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

The limited budget available for this work does not permit full 3-D modelling of flow, electromagnetic and thermal effects in a plasma torch nozzle so an alternative study to give useful results has been agreed by the collaborators. This is based on the practical observation that electrode life is limited by its ability to strike up an arc. The approach to be followed is therefore:

- to carry out a literature search on arc striking mechanisms in plasma torches;
- to study the electromagnetic field around existing cathode design by analytical or computational techniques, taking into account fluid flow where possible;
- to examine specimens of failed electrodes and to establish common causes of failure;
- to apply the above findings to electrode design to try to extend operating time to failure.

This task is now overdue and is awaiting completion of synergistic work at Culham on arc jets and availability of experimental data from Goodwin Air Plasma and Arc Kinetics. Present target is to start the work in the third week of February.

Testing existing electrodes (B.5.)

A later test method for new electrodes is to be practical application in plasma cutting work at customers premises following laboratory or work tests to establish suitability. It is important to establish the performance of existing electrodes in practical applications, and Goodwin Air Plasma are undertaking this in collaboration with their customers.

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

The manufacture of new electrodes cannot start until at least B.3. is finally complete. However, a manufacturing route for the doped electrodes using specialist materials sections at Harwell Laboratory has been identified.

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

At the end of 1991, five research contracts relating to Area No. 4 were at the stage of execution.

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:	G RETTIG, 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

Within the working period a representative charge of tritium containing material was found at nuclear power plants and at a tritium laboratory. The waste consits of 13 tritium gas bottles and more than 1000 kg of dismantled tubes, valves and housing materials. Due to the fact that uncertainties occur with respect to the degree of activity, samples were sent to the Nuclear Research Center Jülich, in order to determine the level of activity. The determination of activity is yet in process. A test plan for measurement techniques was elaborated and the tritium elimination device was chosen.

Progress and results (B.1 and B.2)

Measurements at various labs showed that at a temperature of 150° C and 650° C tritium leaves the material (Figure 1). The H_2 content of the used scrap amounts to 5 - 10 ppm. It is expected that tritium will be spread homogeneously in the outgoing air. The lowest concentration for tritium detection in the air is 400 - 500 Bq / m^3. The following calculation gives an overview on the HT and HTO mass flow:

600 kg scrap with a specific activity of

100 Bq / g leads to a total activity of

$6 \cdot 10^7$ Bq

10 h time for a melting campaign at

1000 m^3 / h exhaust air flow rate leads to a spec. vol. activity of :

$Av = 6 \cdot 10^7$ Bq / 10 h$\cdot$1000 m^3 / h = 6000 Bq/m^3

As HTO **50 % = 3000 Bq/m^3**

As HT **50 % = 3000 Bq/m^3**

For the purpose of designing an appropriate by - pass measurement technique, the venting and filter system of the CARLA plant had been reviewed. Figure 2 depicts the flow rate for one adjustment of the filter elements.

Pre - Experimental set - up

Part of the tritium elimination device is an annealing muffle. Pretestings should give an overview on the furnace parameters. A test device was set up, including the annealing muffle, a cooling apparatus and a box where the samples were collected. The goal was to get some information on sample temperatures and the exhaust air.The annealing muffle was heated up to its maximum temperature, and the aspired exhaust air flow rate should be 2.5 m^3/h, which later was not obtained. Figure 3 depicts the experimental set - up. The achieved results are discussed in the following.

Annealing muffle

The heating device shows very good performance during the test phase. With an exhaust air flow rate of 1.2 m^3/h the oven reached the given temperature level nearly at the same time as without exhausting air. The heating of samples is possible, without any problems up to 1100° C.

Condenser

The dimension of the cooling apparatus was oversized. This leads to strong reduction of the air flow rate due to the increased air resistance. Further experiments will be carried out with another cooler.

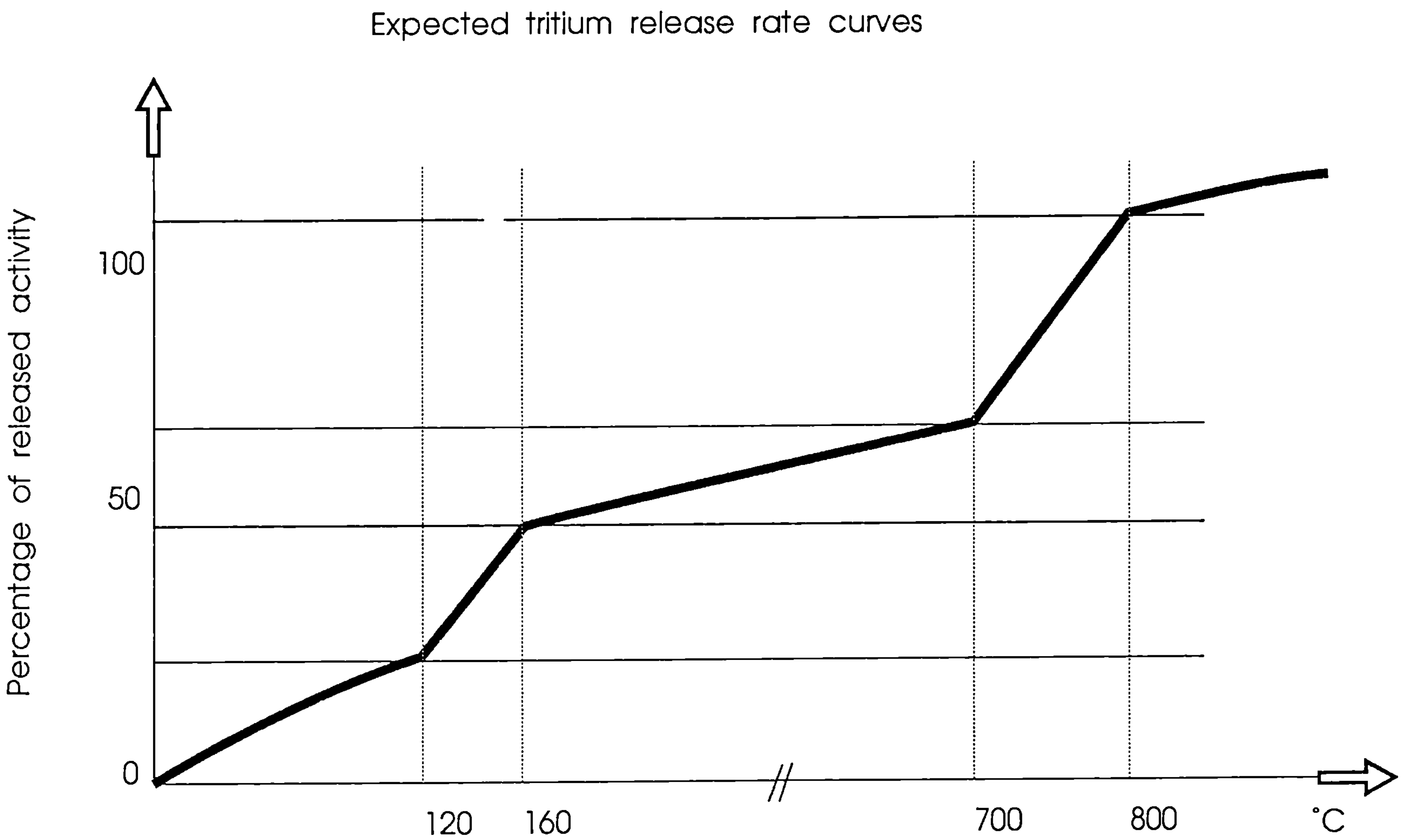
Ventilator

The ventilator was designed too small for the present cooler. With a new cooling apparatus, the expected flow rate of 2.5 m^3/h should be reached.

Conclusion

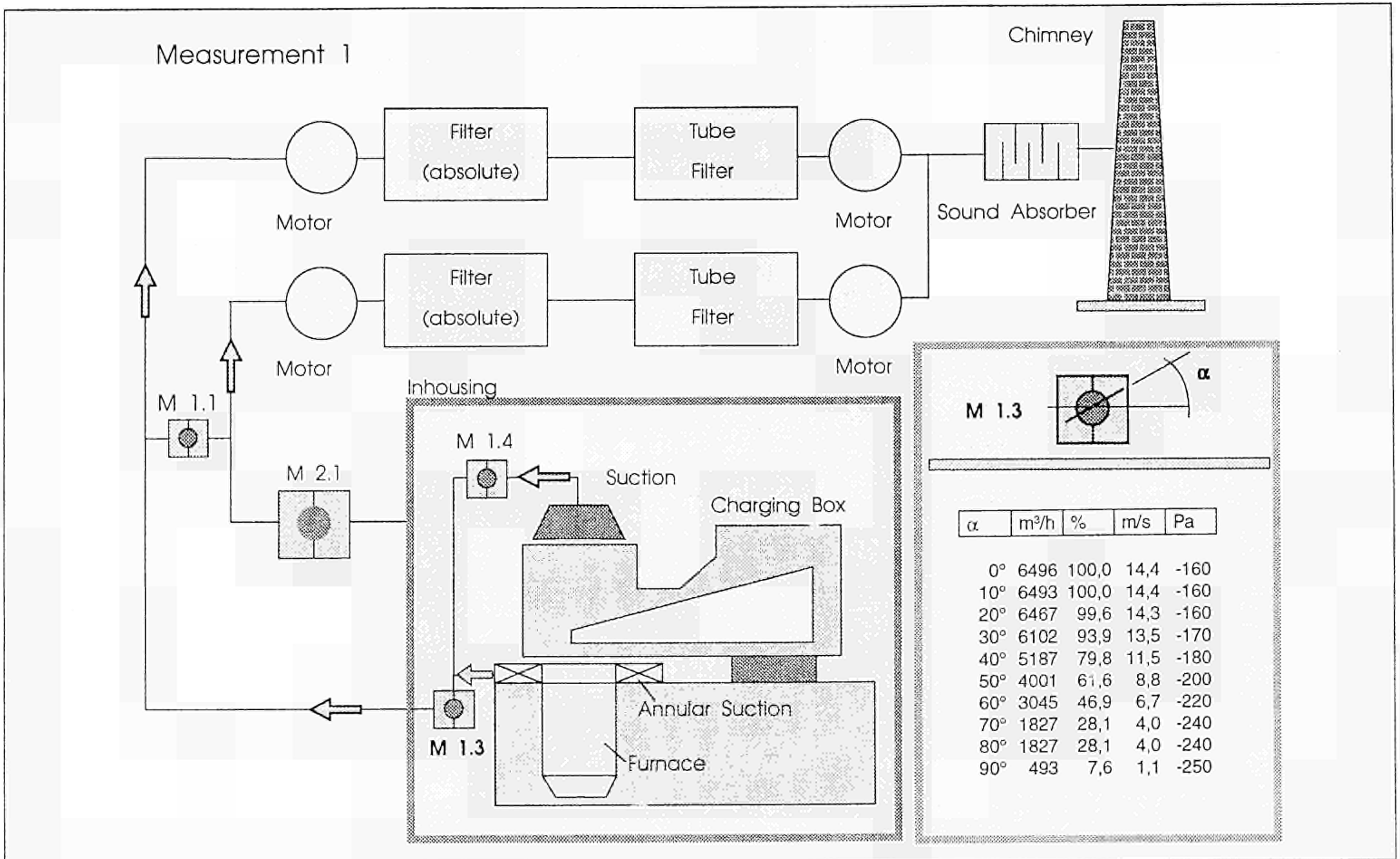
The entire experimental set - up is suited to heat samples up to a temperature of 1100° C and to analyse the released gases with respect to tritium.

Figure 4 gives an overview of the planned by - pass at the CARLA plant.



Melting of Tritium Containing Steel

Figure 1

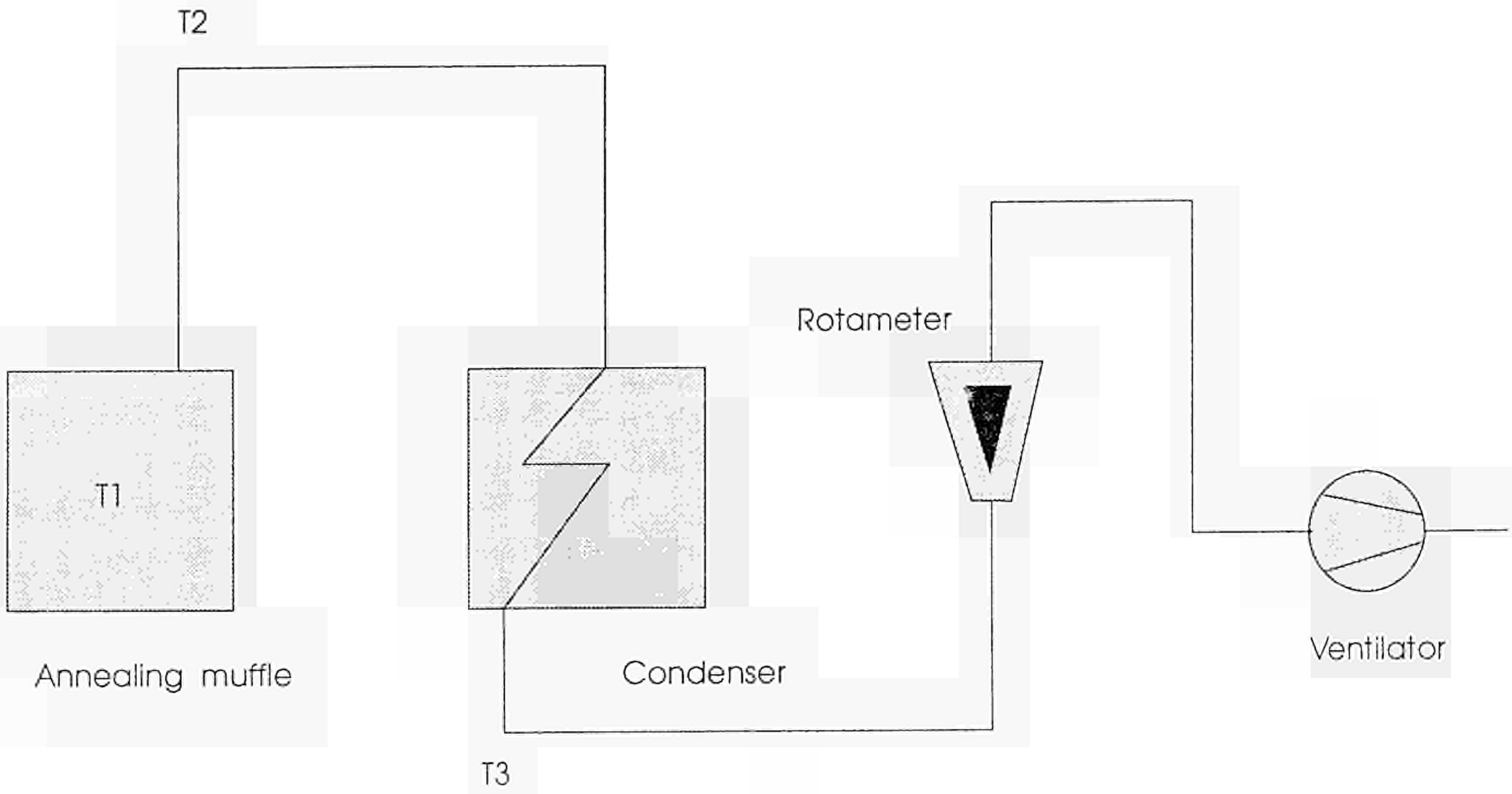


α	m³/h	%	m/s	Pa
0°	6496	100,0	14,4	-160
10°	6493	100,0	14,4	-160
20°	6467	99,6	14,3	-160
30°	6102	93,9	13,5	-170
40°	5187	79,8	11,5	-180
50°	4001	61,6	8,8	-200
60°	3045	46,9	6,7	-220
70°	1827	28,1	4,0	-240
80°	1827	28,1	4,0	-240
90°	493	7,6	1,1	-250

Industrial - scale melting of Tritium containing steel

Figure 2

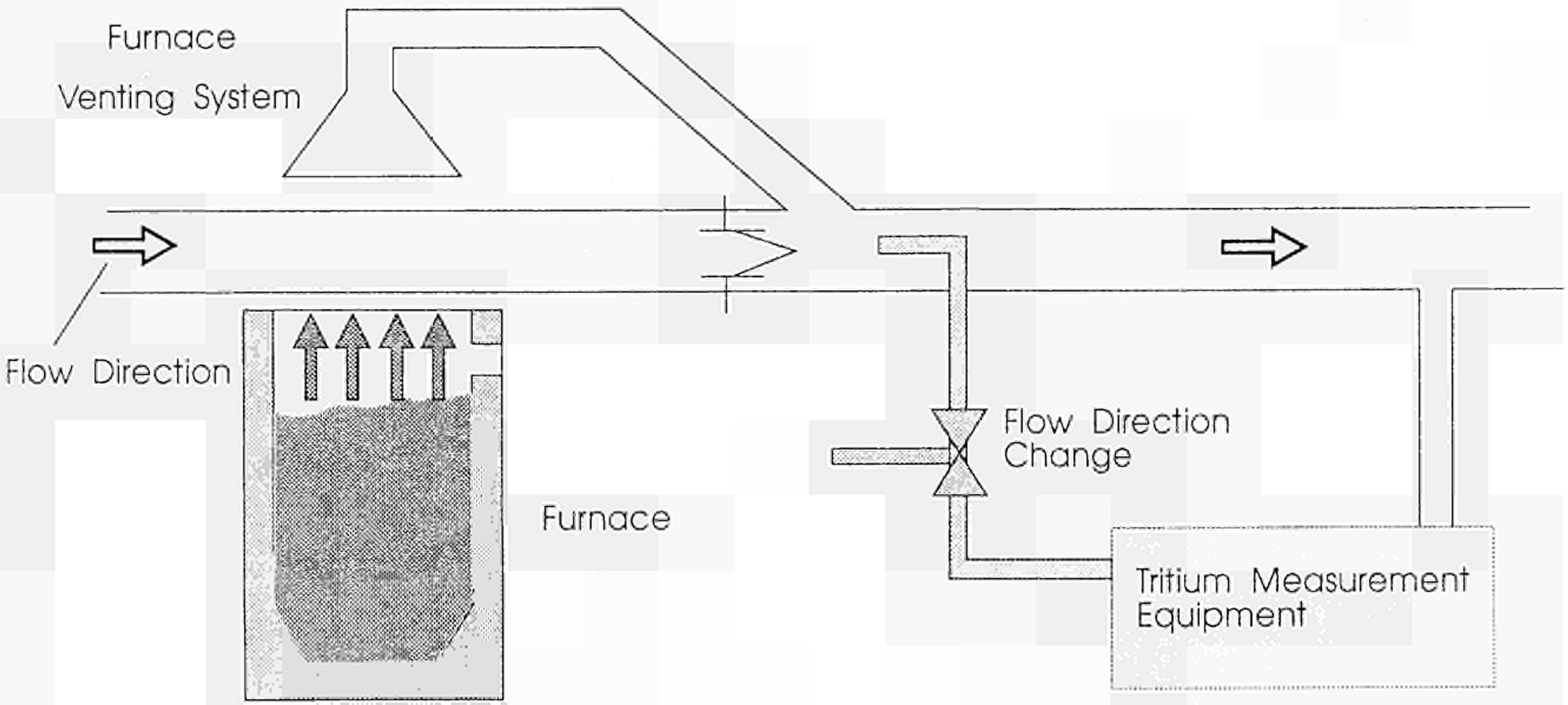
Experimental Set - Up



Melting of Tritium Containing Steel

Figure 3

Draft Design of the By-Pass for Tritium Activity Measurements at CARLA



Melting of Tritium Containing Steel

Figure 4

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 54 56

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 OBTAINED RESULTS

Summary of main issues

By separation of dense aggregate particles and porous cementstone a substantial volume reduction of contaminated concrete is achieved. This method can be applied to contaminated concrete where the contamination is concentrated in the cementstone and the aggregates are clean. It could be proved before that quartz aggregates are not contaminated and that their activation is of minor importance. This is also true for limestone concrete but not for baryte concretes. It was also found that separation of barytes is less effective because they will be crushed during milling.

For separation of mainly quartz and lime-stone concretes the separation pilot plant was engineered. The various components will be assembled in 1992 followed by tests.

Progress and results

1. Irradiation tests on concrete components (B.1)

In order to investigate if concrete will be activated, irradiation tests on its components were performed. The applied neutron fluence was 2.5×10^{23} n/m^3. The activity after 2 and 10 years decay was calculated from the results and are presented in table I. It can be seen that quartz and limestone are less active than cement, whereas the activity of baryte is higher. This means that separation is mainly effective for quartz concretes /1/.

Table I: Results of irradiation tests on concrete components

component	activity after 2 years decay (MBq/kg)	activity after 10 years decay (MBq/kg)
pc cement	88	33
pbfc cement	84	36
quartz gravel	5	1.8
limestone	27	9.4
barytes	122	51

2. Effect of aggregate type on separation (B.2)

In order to get more information about the development of a test installation for volume reduction of contaminated/activated concrete, standard specimens with different types of aggregates were subjected to a standard separation procedure (heating and milling). Three types of concretes with river gravel, limestone and barytes respectively were made and tested /2/.

All concretes were crushed, heated to 650 °C for 4 hours, milled for 4 hours and sieved over 1 mm. The results can be expressed by the separation efficiency (Ed):

$$Ed = As/Ao$$

As is the amount of material < 1 mm after separation and Ao represents the original amount < 1 mm of the mix. The main results are given in table II.

Table II: Separation efficiency of the given types of concretes

types of concrete	Ed mean	c.o.v.
gravel	77	3
limestone	72	5
barytes	142	3

c.o.v. = coefficient of variation (%)

It can be concluded that for the selected separation process, separation of river gravel and limestone concretes are similar.

Because of the relative soft baryte aggregates, crushing and milling of concretes based on these types, result in splitting of the aggregates into fine particles, which consequently reduce the separation of contaminated and non-contaminated parts. It is recommended to investigate if a higher (> 650 °C) temperature will facilitate the separation for these types of concretes, because in that case less milling energy has to be applied and pulverization of the aggregates is prevented.

3. Engineering of the test installation (B.3 and B.4)

The design of the test installation is based on separation by heating. From laboratory tests (subject B.1 and B.2) the components were selected and the test installation was composed /3/.

The installation contains the following process steps:
- crushing
- sieving over 1 mm
- heating
- cooling in air
- milling
- sieving over 1 mm
- conditioning of radwaste and re-use of clean sand and gravel.

These process steps were included in the design as shown in figure 1.

The small scale test installation has a capacity of 100 kg/day broken and contaminated concrete, to be output in two batches of 50 kg. Provisions are made for sampling the input material and output material for the different stages such as after crushing, sieving and milling. The installation will be compact, easy to transport and decontaminable.
All rotating and moving parts are protected and equipped with adequate safety features. Each component of the test installation is connected to the filtering system. After processing the dust will be removed through ventilation by keeping the component for some time at lower pressure than the atmospheric pressure. The lay-out of the small scale test installation is planned in such a way that important process variables can be measured and adjusted /4/.

Most components are standard material and are chosen for the following reasons:
- economical aspects
- availability of spare parts
- guarantee warranty and technical support
- decontaminability
- easy to accommodate.

The following components are selected and purchased:
- jawcrusher
- sieve unit
- furnace
- filter units.

All the components were, if nessary, adapted for special process requirements as:
- type of experiment
- process control
- safety criteria.

For the experiments it is possible to vary parameters such as:
- input parameters:
  * concrete composition (quartz, lime-stone and baryte concrete)
- process parameters:
  * optimization of sieving, heating, milling and ventilation
  * process cycles (time)
  * production amount (material flow rate)
- output parameters:
  * the volume reduction factor in each step of the process
  * efficiency (cleanness processed material).

4. Prospects

In the year 1992 experiments ("cold tests")* will be performed with the installation at KEMA-laboratory. In this experimental stage the following subjects will be emphasized:
- investigation of process and concrete variables
- optimization of the separation process
- optimization of ventilation system
- process control and safety requirements.

Information from VAK will be used concerning the specific conditions and materials used at the nuclear power plant. By the end of 1992 it is expected to transport and install the installation at Kahl or Gundremmingen.

In the mean time a preliminary study will be started on:
- the effects of coatings on the concrete separation process
- an estimation of costs and benefits of a concrete separation plant.

References

/1/ ECN 1991, (Nieuwendijk, B.J.T.). Report on the irradiation of concrete. Report number NP-BT-91-051/0.

/2/ KEMA 1991a, (Hulst, L.P.D.M. van and Cornelissen, H.A.W.). Volume reduction of contaminated concrete. The effect of aggregate type. Report number 10140-TFO 91-3003.

/3/ KEMA, 1991b (Peeze Binkhorst, I.A.G.M.). Test installation for separation of contaminated/activated concrete. Progress report 2. Basics for the design of the installation. Report number 10140-CBP 91-448.

/4/ KEMA, 1992 (Peeze Binkhorst, I.A.G.M.). Test installation for separation of contaminated/activated concrete. Progress report 3. Irradiation tests and final design. Report number 10140-CBP 91-852.

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* Experiments with inactive material

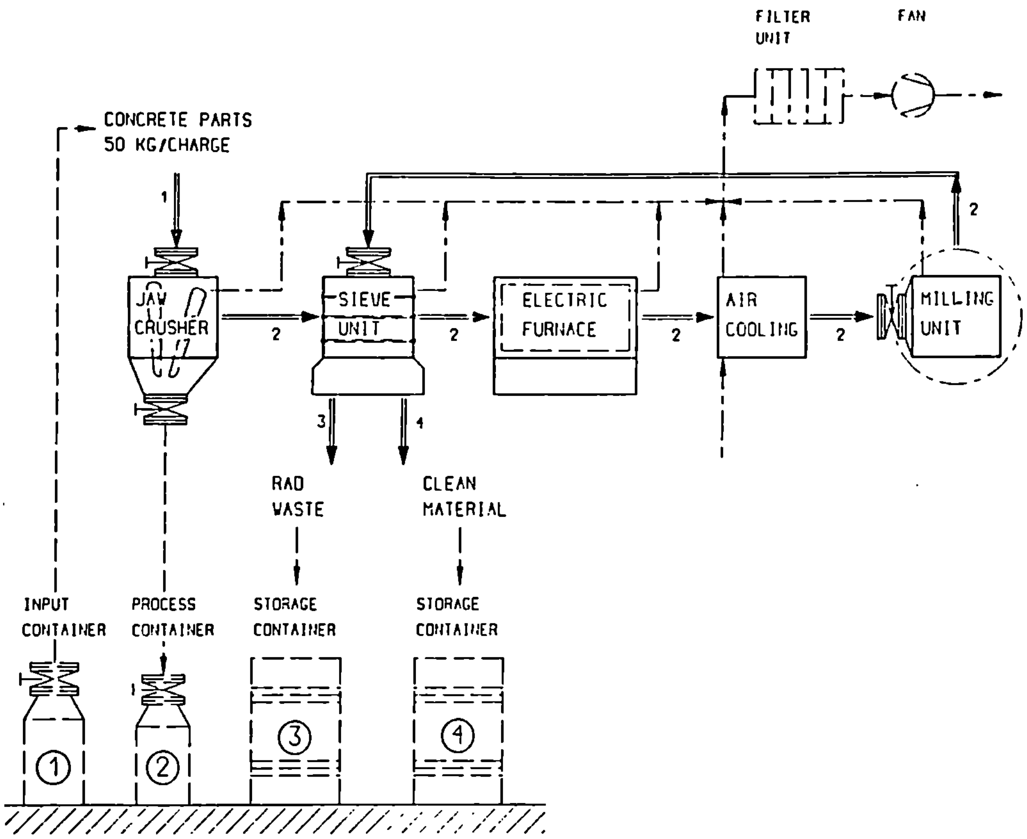


Figure 1: Schematic view of the test installation

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 Obtained Results

Summary of main issues

Graphite has been characterized with respect to organic impurities, surface characteristics and radionuclides concentration.

The radioactive characterization of graphite sleeves from Vandellos-I shows the presence of H-3, C-14 and Co-60 and small amounts of Cs-137, Cs-134 and Eu-154. The presence of the different radionuclides in the sleeves is mainly due to their contamination in the swimming pool of the reactor.

Several straight and cyclic hydrocarbon chains have been detected on inactive graphite samples. It could explain the presence of tritium in the radioactive graphite fixed in the hydrocarbons. This radionuclide can probably be desorbed by passing a wet air stream. Water adsorption and desorption kinetic in graphite has been studied in order to optimize the process.

The metallization of graphite surface by electroless using seventy different methods has been studied, nine of them showing the highest efficiency have been selected. Characterization of metallized surface has been done by electronic and optic microscopy and other technics.

Leaching and corrosion tests on active and inactive graphite samples have been started.

Progress and results

B.1.Removal and/or fixation of radionuclides

The preliminary studies of nuclide extraction from inactive samples have shown the presence of hydrocarbons in graphite, which have been separated within a wide interval of temperatures. The chemical analysis of CO_2 cooler from Vandellos-I reactor have also shown the presence of light hydrocarbons.

The analysis of organic impurities on inactive graphite samples have been realized by four procedures:

a) Thermogravimetry
b) Infrared analysis
c) Mass spectrometry
d) Gas chromatography

The radioactive characterization of graphite sleeves from Vandellos-I nuclear fuel mainly shows the presence of H-3, C-14 and Co-60 and small amounts of Cs-134, Cs-137 and Eu-154 (Figure 1). The radioactivity distribution is uniform along the sleeves but some points with higher concentration have been found. The presence of tritium is a function of hydrocarbon content in the inactive graphite and also if radioactive graphite has been handled under dry or wet conditions. Different straight and cyclic hidrocarbon chains have been detected which could fixe the tritium. This radionuclide can probably be desorbed by passing a wet air stream. Water adsorption and desorption kinetic in graphite has been studied in order to optimize the process (Figure 3 and 4).

The radioactivity content of five sleeves, with different irradiation time and postirradiation handling, has been studied. One of the sleeves has not been dipped in the swimming pool and the other four have stayed in the swimming pool during different time periods (Figure 2).

The conclusion of these experiments is that the radioactivity of the sleeves is mainly due to their contamination in the swimming pool, althougt the same radionuclides have also been found in the dry sleeve. The different concentrations of tritium in the studied sleeves seems to fit with our theory that tritium in fixed in the hydrocarbons.

B.2. Metallic coating of graphite by ionic deposition

Characterization techniques of graphite surface for metallized or original samples have been implemented. Electronic and optic microscopy and rugosimetry have shown to be the most suitable techniques. Flexion test has been used to study the mechanical behavior of the metallic coat.

The best results for the sensitization of the surfaces have been achieved using Ni, Sn/Pd and Ag. The use of oxidants has not given good results.

Seventy different electroless methods have been tested for the metallization with copper, nine showing the highest efficiency have been selected (Tables I and II).

The characterization by X ray of metallized surfaces shows the crystalline structure of the coating. The metallic film adherence on the graphite surface as well as its thickness have been studied using metallografic techniques.

No metallization of large original pores has been detected. It is due to surface tension of the metallic solution on the pores. The use of moistening reagents before metallization will be studied.

B.3. Leaching experiments with metallized specimen

Leaching and corrosion studies of graphite have been started. Metallized and original inactive samples are being studied by corrosion tests. Also original radioactive samples are being leached with deionized water.

TABLE 1

Selected copper plating procedures for graphite R

SPECIMEN	SURFACE ACTIVATION*	PLATING BATH**	RESULTS		
	Metal Salt	Additives and conc.	Time (h)	Amount of deposit mg/cm^2	Amount of deposit mg
72	$AgNO_3$/Borax	PEG 15 ml/l	12	15	196
73		NH_4VO_3 20 ppm	24	11	140
74		KCN 30 ppm	24	14	175
75	Sn/Pd	PEG 15 ml/l	12	14	181
76		NH_4VO_3 20 ppm	24	13	167
77		KCN 30 ppm	24	11	148
78	$NiCl_2$	PEG 15 ml/l	12	16	203
79		NH_4VO_3 20 ppm	12	12	153
80		KCN 30 ppm	12	16	199

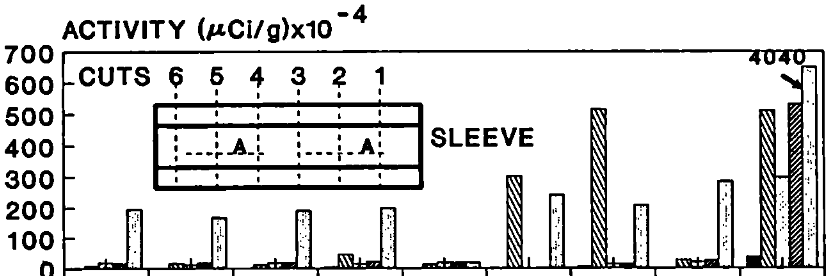
* PEG = Glicol polietilen

TABLE 2

Amount of Cu^{2+} ions extractables with water from copper coated graphite R

SPECIMEN	Cu^{2+} (gr/gr of graphite)x10^6		
	t= 9 days	t= 38 days	TOTAL
72	6.4	7.3	13.7
73	8.5	4.1	12.5
74	8.4	4.0	12.4
75	11.7	4.0	15.7
76	8.9	4.3	13.3
77	10.6	4.9	15.5
78	6.9	8.5	15.3
79	4.6	4.1	8.6
80	5.5	7.1	12.6

RADIOACTIVE ANALYSIS
SAMPLES OF VANDELLOS GRAPHITE
SLEEVE Nº 1

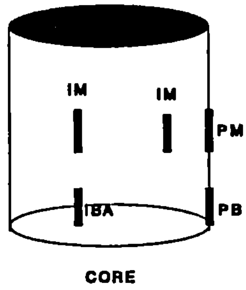


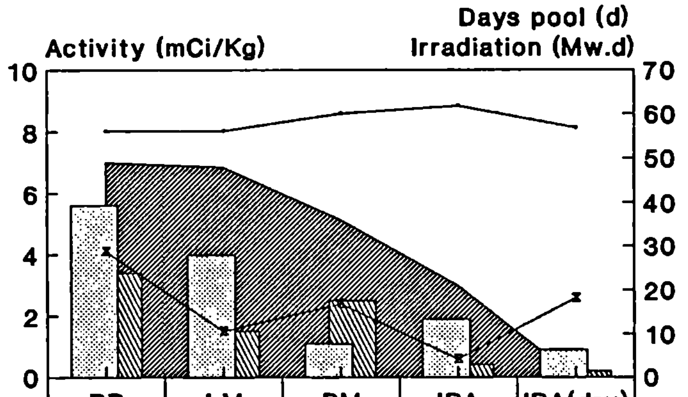
	1	2	3	4	5	1A	5A	6A	SURFACE
Cs-134				3,8			3,9		39
Cs-137	8,7	18,2	12,2	46,4	13,4	301	514	27,6	510
Mn-54	18,6	12,3	19,2	16,1	19,5		15,2	20,9	298
Zn-65	16,6	18,8	19,2	22,8	18,8		14,9	28,8	532
Co-60	194	168	191	200	20,2	240	208	285	650
C-14	2700								
H-3	7800								

FIGURE 1

GRAPHITE SLEEVES
VANDELLOS 1 POWER PLANT

SLEEVES IN CORE





	PB	LM	PM	IBA	IBA(dry)
BETA	4,1	1,5	2,4	0,6	2,6
GAMMA	3,4	1,5	2,5	0,4	0,2
ALFA x 10^{-3}	5,6	3,99	1,1	1,9	0,9
DAYS INTO POOL	49	48	36	21	0
IRRADIATION	56,13	56,07	60,12	61,79	56,82

FIGURE 2

ADSORPTION OF WATER STEAM
INFLUENCE OF PIECE SIZE
GRAPHITE R

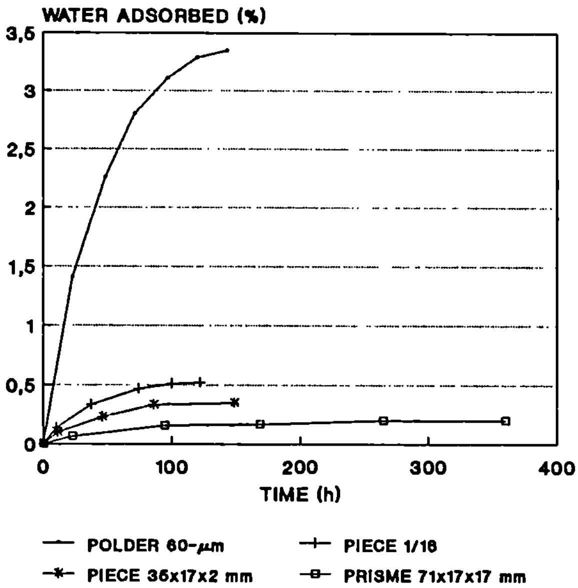


FIGURE 3

ADSORPTION OF WATER STEAM
INFLUENCE RELATION VOLUME/SURFACE

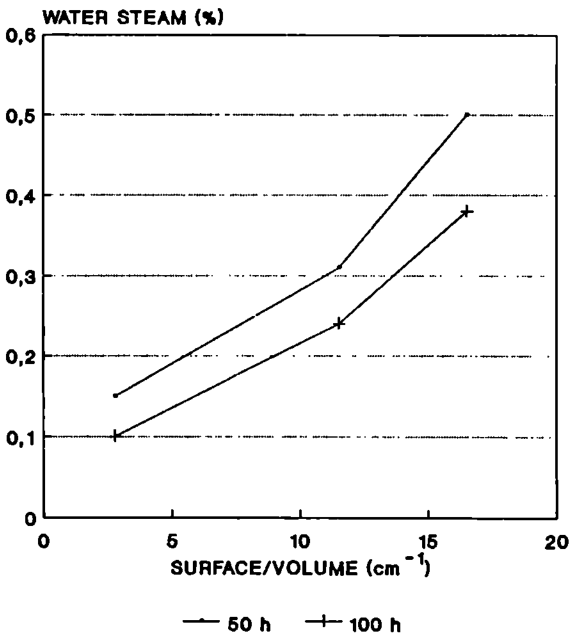


FIGURE 4

4.4. RECYCLING OF ACTIVATED/CONTAMINATED REINFORCEMENT METAL IN CONCRETE

Contractors: Bureau A+
Contract No.: FI2D-0021
Work Period: September 1990 - February 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

The main contaminated steel types in nuclear installations are mild steel and stainless steel. The amounts of contaminated steel, set free during decommissioning of nuclear installations, depends on several factors such as type of reactor, time after shutdown, operation time etc. The best way to recycle this material seems to melt the steel scrap first in order to gain both activity reduction and a controllable product and then process the molten steel into granules.

During the concrete research program three kinds of steel shapes were tested, fibres (long and short), granules and steelscrap. As a first conclusion it can be said that it is possible to add these different types and shapes of steel to concrete in order to get a good result for workability and strength requirements.

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.
Futhermore 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) is set up.

The literature revealed that the main contaminated steel types are stainless steel and mild steel which are mainly contaminated by ^{60}Co and ^{137}Cs. The amounts are depending on factors such as type of reactor, time after shutdown, operation time etc. The amounts of contaminated steel can increase up to several thousands of tons per type.

In the second semester of 91 also some information was gathered on melting and shredding techniques regarding process costs and the way fibres and granules can be produced. This information will be used for rough cost estimations on recycling of contaminated steel in either the form of fibres, granules or small scrap in concrete. From the first results, a tentative conclusion could be that the best option is to melt the steel scrap in order to gain both activity reduction and a controllable product and then process the molten steel into granules.

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

The research on metal-concrete-composites was separated into three parts.

2.1 Preliminary research

In the first part a preliminary concrete recipe study was carried out with the following parameters:
- 2 types of metal which are selected in part B1: Marketing study.
- 3 different metal shapes such as granules, fibres and steelscrap.
- the concrete quality used in this study will be B45 acccording to the Dutch concrete standard. This quality is used in heavy constructions.

2.2. Optimisation-process

Considering the types of concrete to be produced in the test program, the following requirements can be made:
- The concrete must have a compressive strength larger than $50 N/mm^2$.
- The workability must be equal to that needed in common practise of

concete production and application.
- The amount of steel (as total of all shapes) must be as high as possible.

First small tests were carried out in order to optimize the workability by adjusting the mixture composition for each of the combinations of steel and shape. Then preliminary tests were conducted to see whether the produced concrete according to these recipes, would meet the requirement for the compressive strength. Except the short fibres all test mixtures showed about the same strength. The lower strength of the short fibres mixture is possibly caused by either the somewhat higher workability and/or the less good mixing properties (Table I).

2.3 Test program

A next step of the research program was making and testing samples according to each individual recipe. Based on the results of the preliminary research and the optimisation-process, measurements were carried out on some basic mechanical properties of concrete, made according to the above mentioned parameters. For each of the selected recipes following properties were determined:

a) compressive strength (Table II).
b) tensile bending strength (Table III).
c) modulus of elasticity (Table IV).

3. Conclusions

The conclusions which already can be drawn in this phase of the test program is that it is possible to add different types and shapes of steel both from the point of view of workability and strength requirements.
The test results from the period September 1991 - December 1991 are included in table II and III. The recent information concerning the modulus of elasticity was not processed during the above mentioned period.

Table I: Results of the preliminary tests

Steel type	Shape	Compressive strength (N/mm^2) 3 days σ	σmean	7 days σ	σmean
Blanc (no steel)	-	20.2	20.7	29.5	30.3
		21.0		30.5	
		20.8		31.0	
Stainless steel	fibres (long)	16.6	19.2	31.5	30.3
		20.8		29.5	
		20.2		30.0	
	granules	25.1	23.4	31.5	31.1
		22.0		31.5	
		23.0		30.8	
Mild steel	fibres (short)	11.4	10.9	18.0	20.7
		10.3		22.5	
		11.1		21.5	
	granules	24.1	24.2	31.8	32.7
		24.5		34.1	
		24.0		32.2	
Miscellaneous	steel scrap	31.0	29.9	31.0	31.7
		33.3		35.2	
		22.7		28.0	

Table II: Results of the test program

Compressive strength (N/mm^2)												
days	no steel		mild steel				stainl. steel				miscell.	
	blanc		fibres short		granules		fibres long		granules		steel scrap	
	σ	σmean	σ	σmean	σ	σmean	σ	σmean	σ	σmean	σ	σmean
1	4.6	5.3	9.7	9.0	20.4	18.8	10.1	10.4	16.2	16.5	12.1	13.0
	4.2		8.7		17.4		12.4		14.8		14.3	
	7.0		8.6		18.6		8.8		18.5		12.7	
3	20.2	21.0	22.9	23.0	32.3	30.9	24.8	26.1	26.2	27.1	33.3	29.7
	20.2		25.4		33.4		27.6		27.8		30.8	
	22.7		20.6		27.0		25.8		27.3		24.9	
7	32.5	31.8	27.4	29.0	42.0	39.0	31.7	33.5	40.1	39.4	34.8	32.5
	34.5		34.9		42.9		37.7		40.3		32.4	
	28.5		24.8		32.2		31.2		37.9		30.4	
28	40.0	40.0	32.1	33.3	58.1	53.0	49.6	47.6	47.5	45.9	41.4	40.8
	40.2		30.8		55.7		47.4		45.5		46.2	
	39.7		33.6		49.4		42.6		43.1		34.7	
	41.4		34.1		55.8		51.1		46.3		42.5	
	42.4		34.4		55.6		48.3		49.4		41.1	
	36.4		34.6		43.5		46.4		43.5		38.9.	
56	46.1	44.6	35.9	35.3	58.7	54.4	50.5	49.7	50.4	49.4	46.3	44.4
	45.5		36.0		57.3		50.8		48.2		46.9	
	42.2		34.0		47.2		47.9		49.6		40.0	

Table III: Results of the test program

Tensile bending strength (N/mm^2)												
days	no steel		mild steel				stainl. steel				miscell.	
	blanc		fibres short		granules		fibres long		granules		steel scrap	
	σ	σmean	σ	σmean	σ	σmean	σ	σmean	σ	σmean	σ	σmean
28	4.2	4.8	5.3	4.6	4.7	5.3	8.5	8.1	5.0	4.7	4.9	4.6
	5.1		5.1		5.3		7.6		4.5		4.2	
	5.1		3.3		5.9		8.2		4 6		4.7	

Table IV: Results of the test program

Modulus of elasticity (kN/mm^2)								
days	no steel		mild steel		stainl. steel			
	blanc		fibres short		fibres long		granules	
	σ	σmean	σ	σmean	σ	σmean	σ	σmean
28	31.4	32.2	28.8	28.7	35.6	35.1	34.3	34.6
	32.9		28.6		34.5		34.9	

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 - 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 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 obtained results

Arrangement between CIEMAT, Madrid / Siemens-SG (B.1.)

The cooperation between CIEMAT and Siemens-SG was arranged. The aim of CIEMAT is to obtain a volume reduction of aluminium scrap by melting and the aim of Siemens-SG is to obtain a decontamination of scrap by melting.

Installation of an inductively heatend furnace (B.2.)

In industrial scale, contaminated aluminium scrap will be molten by an inductively heated furnace.

Therefore, in order to meet the same melting conditions, an inductively heated furnace was installed to perform melting experiments in a laboratory scale.

The technical data and the dimensions of the installed melting plant are specified in table 2. In figure 1 a schematic view of the plant is given and figure 2 shows the process flow diagram for the melting experiments.

Procurement of representative contaminated Al- and Cu-samples (B.3.)

We obtained a container with 160 kg of radioactive contaminated aluminium scrap from different nuclear power plants. The scrap was cut into pieces with outer dimensions smaller than 5 cm. So it can be used in the graphite crucibles with an inner diameter of 10 cm for the melting experiments.

Treatment of Cu (B.4.)

Siemens supported the tests of SG by analysing samples of a Cu-test-melt for lowest α -, β- and γ -activity. The results showed activities below the limit values for unrestricted reuse.

For example: cast copper block No 90 0269
nuclides: Co 60, Cs 137, Fe 55, Ni 63
Surface contamination

α < 0.05 Bq/cm^2
β/γ < 0.2 Bq/cm^2
β + electron emitter < 5 Bq/cm^2

Surface dose rate: 0.17 μ Sv/h
mass specific activity

α - activity	<	0.012 Bq/g
ß - activity		1.0 Bq
γ - activity Co 60		0.07 Bq/g
γ - activity Cs 137	<	0.17 Bq/g

Investigations on metal coating seperation and γ-nuclide distribution (B.4.1)

Representative charges of contaminated copper were ordered and its radiological documentation elaborated. Detected nuclides are Cs 137,Co 60 and K40. Because copper is usually covered by other materials, e.g. by plastics, glimmer or rubberlike materials, these casings had to be removed in order to avoid complications during melting (off gas from plastics). Several tests had been carried out. Manual procedures are not acceptable with respect to cost and time for an industrial scale process. The next procedure was to freeze the material by liquid nitrogen. The process itself seems to be successful, but again the operating expenses were not acceptable due to the high consumption of nitrogen. A more economical solution was a general purpose chopper device, able to cut, in a first step, a large variety of cables (including strong motorwindings). In a second step a rotating chopper cuts the copper / coating elements(and thereby separates them)into small grains. A vibrator together with tube filters assure the final fine separation. The entire material can be strored in a 20 ft container

Laboratory-Scale Melting Experiments with Al (B.5.)

Preliminary experiments

On a preliminary experiment with a customary resistance furnace a small portion of aluminium scrap was molten together with slag former NaCl, KCl and CaF_2. The contaminated scrap was procured from a nuclear power plant.

The γ -spectrometric examination showed mainly Co 60 and Ag 110 m as contaminating nuclides. After melting a decontamination factor of 30 resulted for Co 60 and a factor of 3.5 for Ag 110 m only.

Table 1: Conditions of melting of aluminium scrap in the preliminary test

Table 2: Technical data of melting plant

Fig. 1: Systematic view of the experimental plant

Fig. 2: Process flow-diagram of melting and activity measurements

Table 1: **Conditions of melting of aluminium scrap in the preliminary test**

Sample	aluminium with radioactive surface contamination
Slag (weight per cent related to aluminium)	12 % NaCl 12 % KCl 1.2 % CaF_2
Weight	1255 g aluminium 314 g salt-slag
Crucible	corund
Melting temperature	730°C
Melting time	50 min.
Activity of aluminium before melting	89 Bq Co60/g Al 33 Bq Ag110m/g Al
after melting	3 Bq Co60m/g Al 9 Bq Ag110m/g Al
Decontamination factor	30 with Co60 3.5 with Ag110m

Table 2: **Technical datas of the melting plant**

Electrical power supply	500/250 V 197/294 A 73.3 KVA 10 KHz
Furnace inner diameter	600 mm
volume	212 l
purging gasflow	2700 l/h max
Crucible inner diameter	100 mm
volume	1.5 l
material	graphite

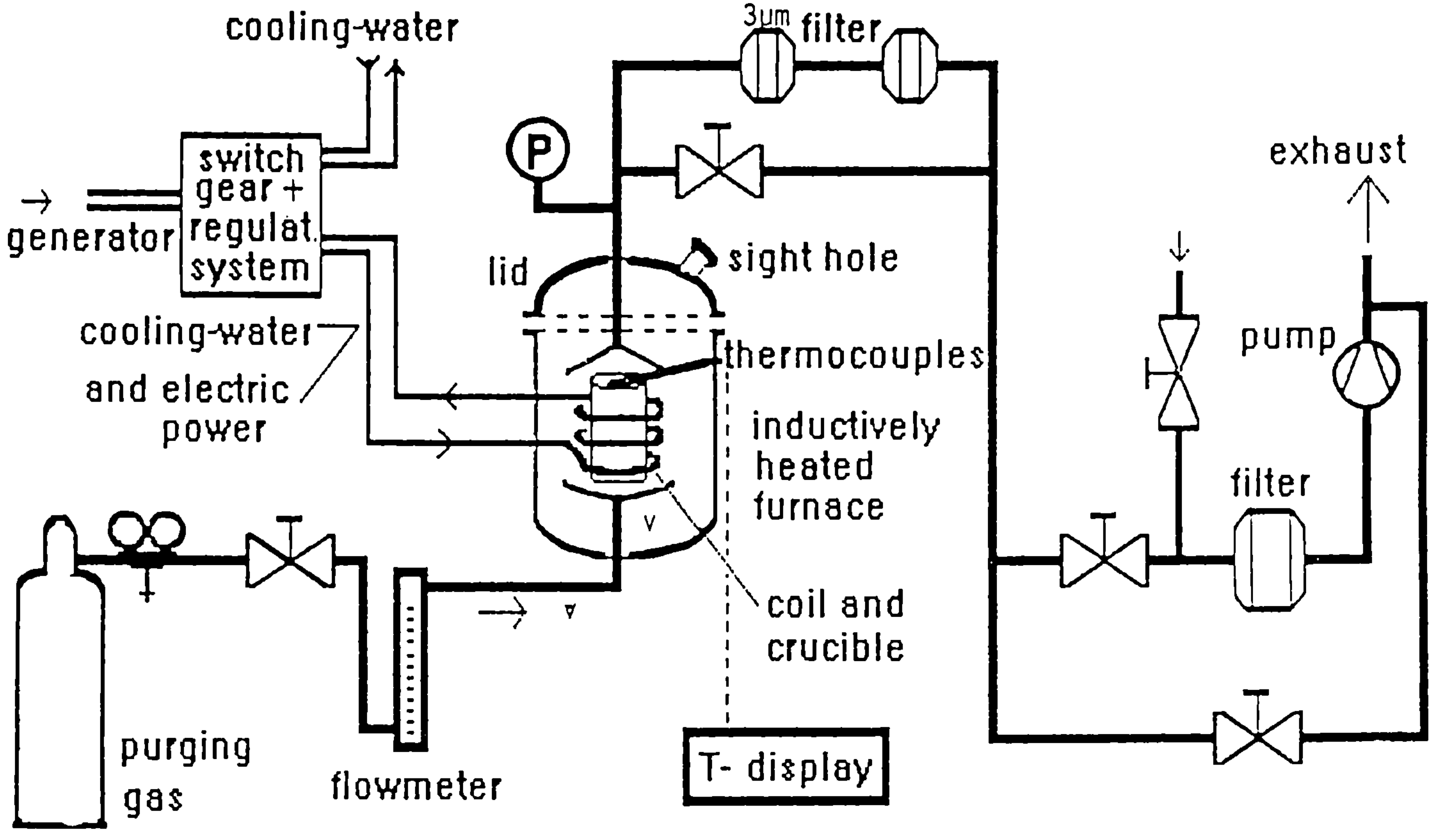
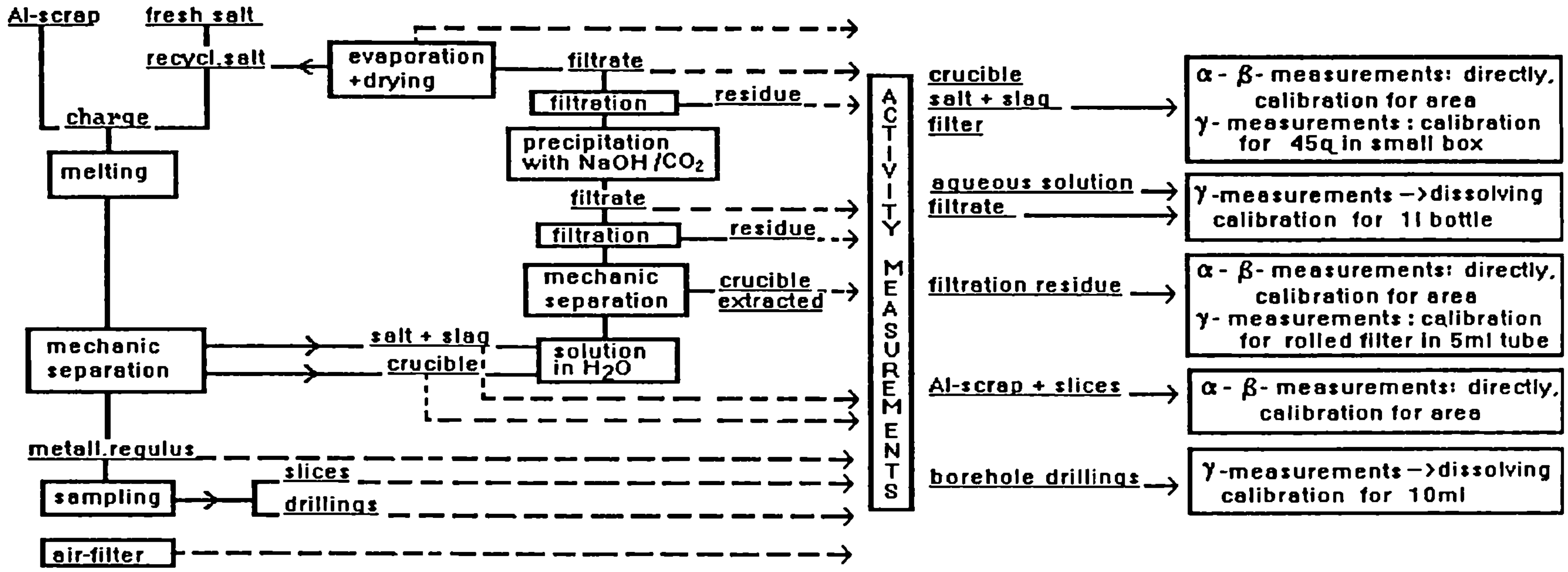


Fig. 1: **experimental plant**

v 200 l
v̇ 2700 l/h (maximum)



FLOWDIAGRAM: MELTING AND ACTIVITY MEASUREMENTS

Fig 2

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

At the end of 1991, six research contracts relating to Area No. 5 were at the stage of execution.

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 50

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 1991 concerned the design of the robotic system; the system consists in the MASCOT IV servomanipulator installed inside a containment box. The detailed design of the robotic system has been completed.

The detailed design of the mock-up of the Dissolution Cell of the EUREX plant has also been completed.

Because of the unavailability of the Technological Hall, the tests will be carried out in room G 43 of the ITREC plant, needing the cell mock-up design to be reduced to 4 metres high instead of 5 metres.

Progress and results

1. Design and construction of a mock-up of the Dissolution Cell of the EUREX plant (B.1.)

When the experiment was planned, it was to take place in the Technological Hall of the Trisaia Centre; this Hall was suitable for the simulated dismantling of the Dissolution Cell of the EUREX Plant because it is provided with a 6 m deep trench (where the cell mock-up could be positioned) and it is equipped with a bridge crane of sufficient lifting capacity (5000 daN).

The unavailability of this room made it necessary to find another location where the tests could be carried out.

The site chosen is the room G43 of the ITREC plant at Trisaia Centre; the room is used for entrance and exit of material to and from the plant and it must thereforebe possible to clear it when necessary. As the room is provided with a 4.3 m high door, the mock-up cannot be higher than 4 metres (the depth of the present cell is 5 metres).

The tests maintain their significance, even if the cell dummy is 1 m shorter because:
- a 1 m high sleeve is added to the cell dummy to ensure a depth of 5 metres;
- the mock-up of the equipment is in full scale, in particular the parts that must be cut, such as the pipes and the supports; the only piece that cannot be simulated in full scale is the dissolver, which is 5.3 m high; it will nevertheless be dismantled according to the procedure foreseen for the present dissolver: after cutting the pipes and the supports connected to the upper part of the dissolver, it will be cut into two; after the upper part has been taken away, the lower part will be dismantled in the same manner.

As previously mentioned, the design of the cell mock-up has been carried out; the layout of the equipment and of the piping being like that of the actual cell.

The cell mock-up is at present being constructed at the workshop of the Trisaia Centre.

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 communicates with the cell by means of an opening closed by a double lid system.

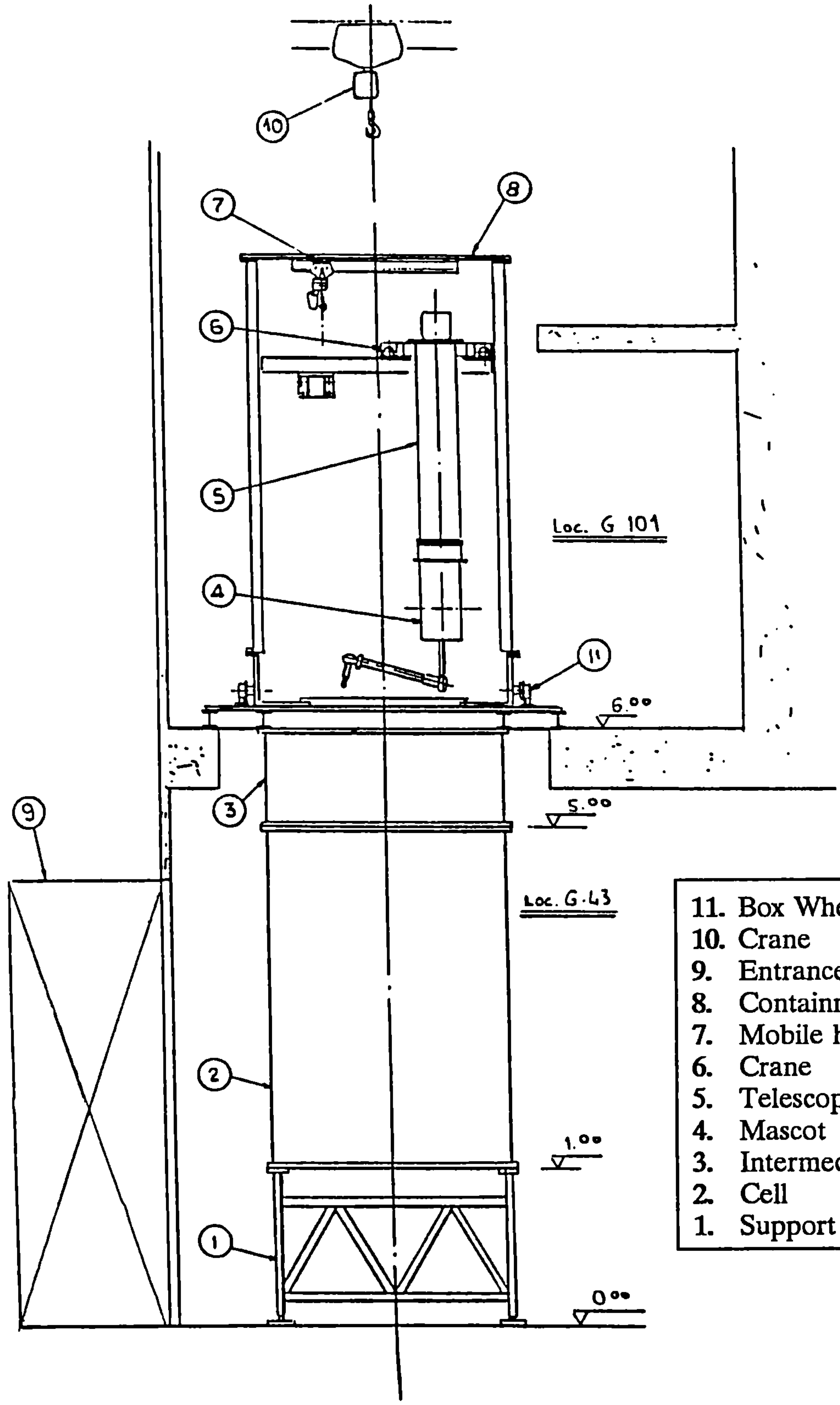
A telescopic tube, mounted on a carriage, supports the MASCOT; its elongation is about 5 metres.

The detailed design of the system has been completed; its construction will be carried out in the first half of 1992.

The transfer to an area other than the Technological Hall caused the design of the robotic system also to be modified. The bridge crane installed in the room G101, the room above the G43, has a lifting capacity (2000 daN) insufficient for the robotic system. The mobility of the box will be assured by four wheels moving along guide-rails; when the box is positioned in axis with the cell, it is lowered by means of manual jacks until the gasket is pressed.

For the overall drawing of the robotic system, see Figures 1 and 2.

The containment box is 2.1 m long, 1.7 wide and 4.5 m high; its dimensions have been chosen according to the dimensions of the MASCOT and of the telescopic tube.
An opening is present at the bottom of the box; its dimensions (1.6 x 1.1 metres) have been chosen according to the EUREX process cells.
The opening is closed by a double lid system. Two magnets assure the coupling of the two lids when it is open; four pins jam the cell lid when the box is taken away.
Four pneumatic cylinders assure the lifting of the two coupled lids to align them to the fixed guide-rails; afterwards another pneumatic cylinder assures the horizontal translation of the lids into a lateral appendix present in the box.
A mobile crane for moving the MASCOT is placed inside the containment box; the MASCOT is supported by a telescopic column; it consists of five square sections; bearing present in the sections assure the relative sliding; lifting is assured by a chain block.
A cable duct is installed outside the column; it consists of several sections connected by means of hinges allowing the cables to follow the elongation of the column.



11. Box Wheels
10. Crane
9. Entrance door
8. Containment box
7. Mobile hoist (500 kg)
6. Crane
5. Telescopic column
4. Mascot
3. Intermediate structure
2. Cell
1. Support

FIG. 1 - Overall drawing

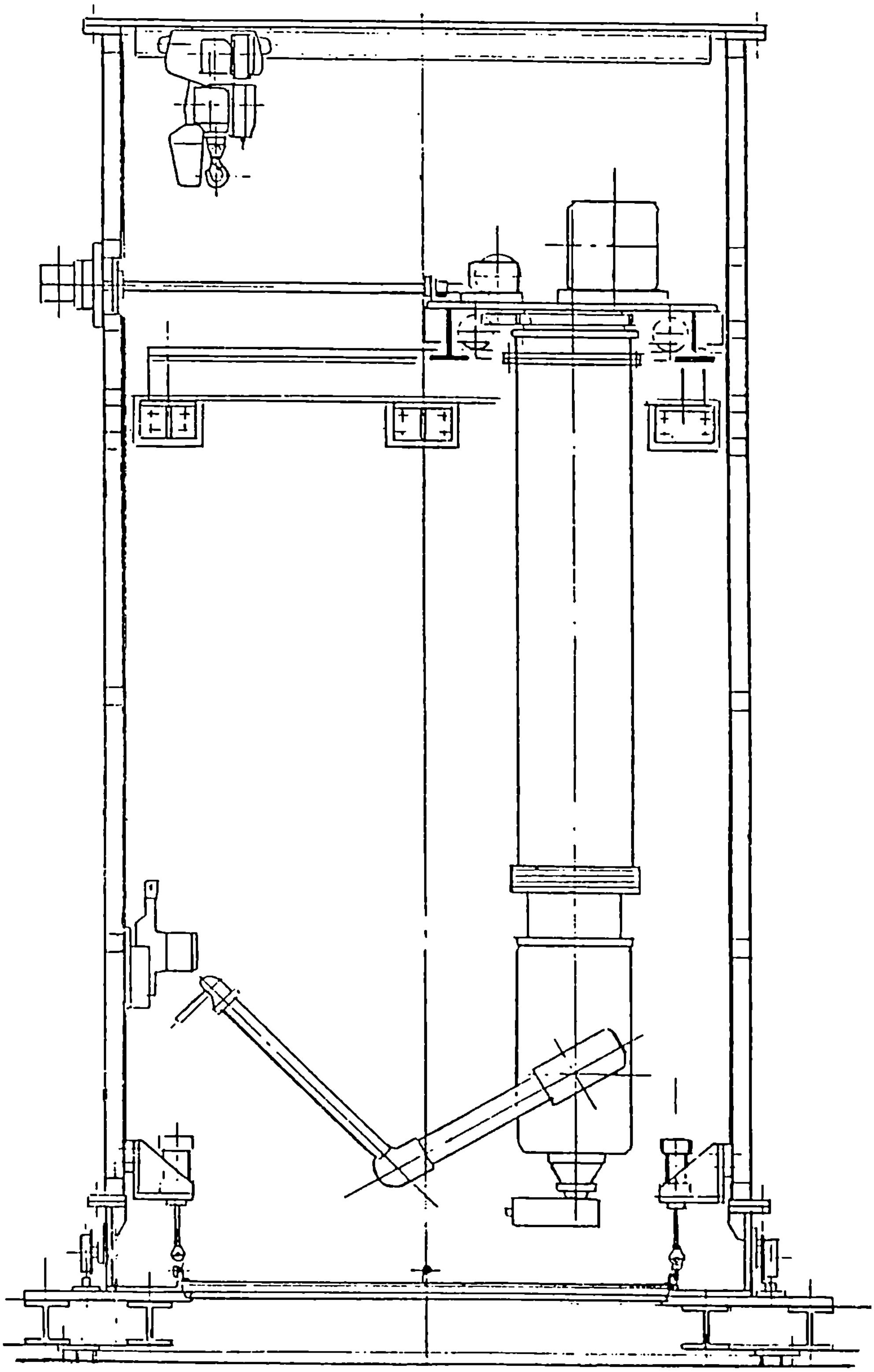


Figure 2 - The robotic system in the containment box.

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 work and obtained results

Summary

Starting from operational procedures and environmental conditions, and after an economical and technical evaluation, detailed design of a jib crane supporting a telescopic mast on a horizontal trolley has been developed.

The jib crane can be rotated over the crucible open and the telescopic mast can be introduced into the furnace, allowing the slag removal by a special grab and other operations such as the iron sampling and temperature measurement.
A special bellow will contain the dusts and fumes spreading in the Carla hall.

Progress and results

1. System requirements, basic operations and interfaces (B1)

The system is designed to meet the following functional requirements:

- to respect the existing lay-out constraints in terms of dimensions and general arrangement (charging system, crane, suction system, etc.)
- to perform slag removal (up to three times each charge) 200 Kg and 250 mm thickness each time)
- to allow melting bath temperature measurement (up to four times a cycle by inserting the sensors 200 mm inside the melting bath for 5 - 10 sec.). Four sensors are necessary per charge
- to check melting bath composition by taking 50 mm diameter x 8 mm thickness samples to allow automatic transportation to the laboratories (an average of three samples within a campaign)
- to perform grab polishing and coating
- to perform the above operation without the presence of personnel, thus avoiding their radiation exposure, in particular through inhalation of radionuclides, and reducing the contamination of the surroundings of the furnace.

The system environmental conditions are:

Pressure difference

Housing/Hall: - 10 Pa

Manipulator/Housing: - 200 Pa

Temperatures

Melting bath: max 1750 °C

Housing	min	+ 2°C	surrounding	min	+ 2 °C
	operation	+ 30°C	the manipulator	operation	+ 100 °C
	max	+ 70°C		max	+ 160 °C

Manipulator payload (200 Kg slag + 200 Kg gripper)

2. System definition (B.2.1)

Basic concept for the definition of the system is that a remotely operated manipulation during the operations of slag removal, sampling and temperature measurement, is necessary to prevent personnel from working in a radioactive environment, for considerably long periods.

In fact it is required to remove about 200 Kg of slag each campaign and, if this task had to be performed manually, it would require a direct exposure of the people working close to the crucible.

The same problem would exist for the several temperature measurements and samplings of the melting bath.

The single-purpose solution can perform all the operations required but does not meet lay out constraints due to the limitation of space available, not allowing the presence of the additional trolley with respect to the chosen solution.

Costs of single-purpose solution are a bit higher than these of the multi-purpose one.

3. The multi-purpose device "jib crane" (B.2.2) consists of a jib crane, mounted on the crucible operating floor, and able to cover a 240° rotation angle to allow positioning of the trolley above all the working positions (see sketch 1).

The crane structure is made of welded steel beams supporting the rails for the trolley, which can move horizontally for a 2.8 m stroke.

A 2.1 m stroke vertical telescopic mast is mounted on the trolley, the mast is driven by an electric hoist and is able to rotate for a 360° angle.

A pneumatic actuated Sommer WW180 automatic changing tool device is fixed to the bottom end of the telescopic mast and allows a quick connection of electric and pneumatic supply lines for tool operation.

The trolley is provided with a containment and exhaust system for fumes and dusts coming out of the furnace during open lid working phases.

Such system basically consists of two parts: the upper part joining the telescopic mast to the exhaust system and the lower bellow, in such a way that dusts and fumes spreading is prevented.

When the telescopic mast moves downwards, the bellow descends in the same way, almost touching the crucible, to convey the fumes to the exhaust system.

The grab rest position is located in correspondence of the jib crane vertical pillar, the stand is provided with a hammering device to detach the slag which can stick to the grab.

Tools

1) A special pneumatic grab with two hemicylindrical jaws will be able to pick up the slag (about 200 Kg) from the whole furnace surface.
To remove the slag from the edge of the furnace, a helical blade will be fixed at the periphery of the jaws.
The telescopic mast will rotate (about 180°) so that the slag will be moved to the centre of the furnace.
To join the grab to the crane a pneumatic hook will be used.

2) A special suction tool will be used to sample the molten iron. It will be joined to the mast by a pneumatic hook.

3) Thermocouples will be used for precise temperature measurements; a steel rod will be connected to the automatic flange and inserted into the molten iron by operating the vertical mast. The automatic interface will therefore be able to transfer the thermocouple electric signals.

The control system architecture is based on a Siemens S100 PLC. Main PLC task is to supply interface between plant and operator according to the control algorithms implemented. The operator talks to the PLC by means of a control console provided with a plant synoptic. Two kinds of operation are available, manual and automatic sequences controlled and monitored by the PLC. Several facilities are available to the operator to perform on-line monitoring.

4. <u>Designed of the defined system components</u> (B.3)

The whole project has been divided into the following main subassemblies

- grab and mast flange rotation
- mast lifting device and trolley
- jib crane structure and runways
- jib crane rotation and hammer device.

All the parts and the components of the system have been chosen considering the dimensions and interfaces.

During the detailed design of the system the following problems have been examinated:

- Definition of the constructive design of a grab able to remove the slag at the edge of the crucible, developing a special mock-up of the crucible and the grab.
- Tests of the grab functional performances will be developed at the beginning of next year
- To simplify the design avoiding the necessity of the whole mast rotation, the lower flange has been motorized using a DC motor inserted inside the lower part of the telescopic mast in such a way that it is cooled by the cold air flow passing through the mast
- Dimensional problems concerning the pneumatic tubes lay-out inside the mast have been overcome using miniature electrovalves and tubes
- To improve the vibration capability of the hammer device an AC motor will drive two eccentrics
- To avoid slipping of the trolley wheels on the rails, a pinion-rack driving system has been chosen thus assuring precise positioning.

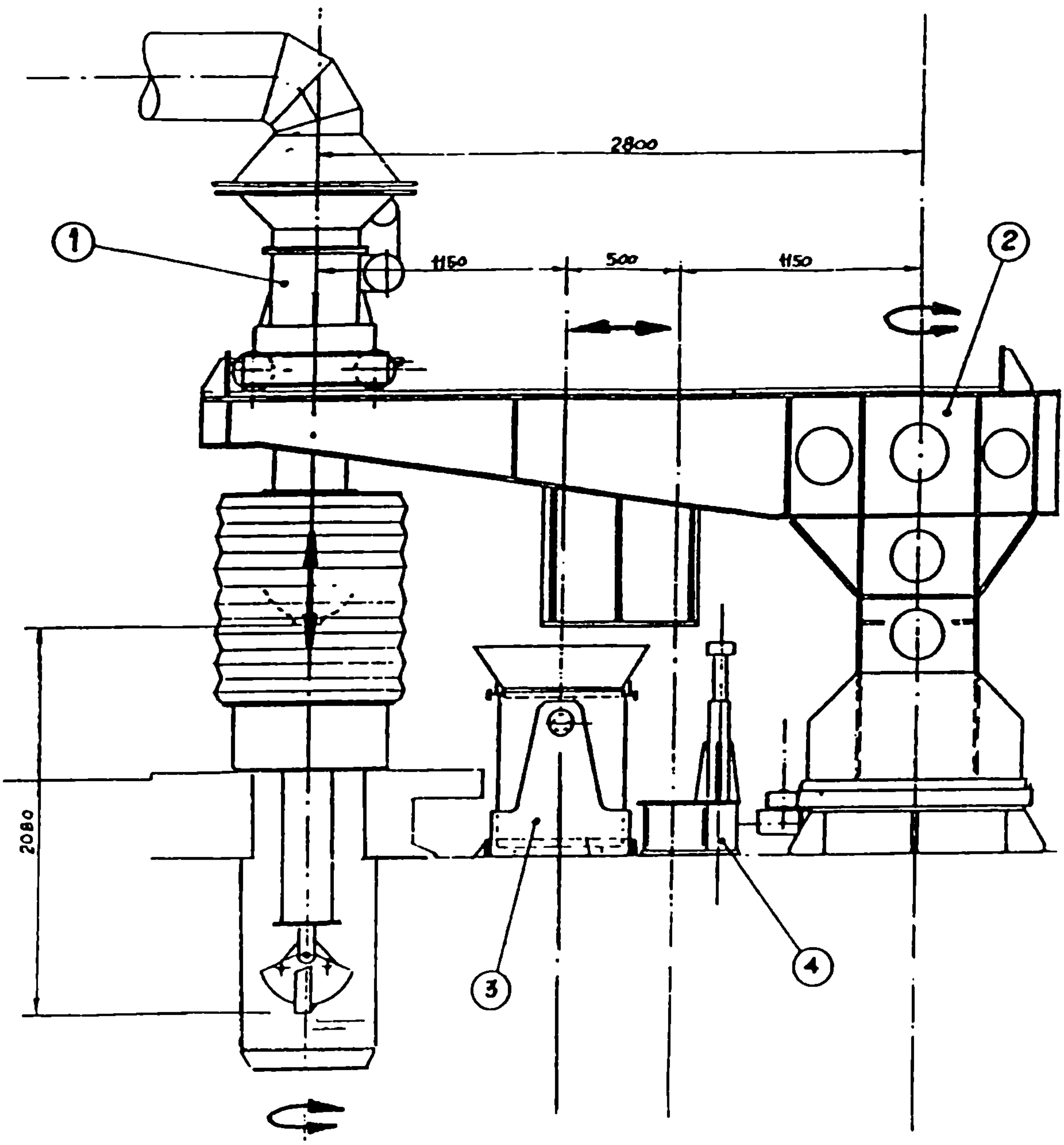


FIGURE 1 - JIB CRANE

1 TROLLEY
2 JIB CRANE
3 SLAG BARREL
4 TOOLS

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. Active decommissioning trials in the appropriate active area
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. Economic analysis of TDMSRS and its radiological impact on work force and working area
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

The control system (B1) has been extended to meet the requirements of the three demonstrators and the size reduction (B5) of five gloveboxes has been demonstrated. Work packages B1 and B5 are, therefore, complete. A new electropolishing head unit (EHU) (B2) has been assembled and interfaces to the robotic system constructed and tested. Irradiation tests for the EHU have been planned and will start in 1992. Scanning software development to deploy the EHU in a decontamination demonstration (B3) is under way. A kinematic analysis to define the clearance monitoring (B4) requirement has also started.

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 degree-of-freedom input device to provide 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 standard communications interface. A Cartesian mini-Master Arm (CARMA) provides, via the TRC, unilateral or bilateral control of NEATER.

2. Control system extension (B1)

To speed up the development of application software for the three demonstrators, two software modules were designed and coded. One module was developed to facilitate the rapid changing of system parameters to meet the different requirements of each application. The other module provides system diagnostics, an improved programmer interface and a library of i.o. device drivers to ensure that i.o. access is common to each application.

The interface between the TRC and the electropolishing head unit controller has been defined and the electropolishing operating sequence devised. Commands from the telerobotic controller to the electropolishing head unit controller are: initialise/release head, anode attach/detach, decontaminate and emergency stop and in the reverse direction: decontamination completed and system error.

The electropolishing head unit controller i.o. has been constructed to meet the interface requirements. Some modifications to the electropolishing unit were also necessary. The functionality of the system was successfully tested using a simulated TRC.

3. Electropolishing head unit (B2)

A new electropolishing head unit has been designed and manufactured. As a result of previous operational experience the double seal vacuum attachment system of the old head unit has been replaced by four vacuum cups leaving a single electrolyte seal.

A requirements analysis has been performed on the sensing needs of the decontamination task. A range of sensors were considered but the work was focused on the position sensing requirement. Special attention was paid to the task of selecting sensors with sufficient tolerance to meet the ionising radiation specification. A list of such sensors /1/ has been established. The results show that inductive sensors are suitable for proximity sensing and some types have high levels of radiation tolerance. Radiation hard ultrasonic and capacitive sensors were identified as suitable candidates for long range and collision avoidance sensing. The proposed sensors will be tested on the Puma 762 robot located at CEN/SCK, Mol. As the Puma 762 robot is kinematically similar to the NEATER robot, the results will be directly transferable to the surface decontamination task (B3).

All electropolishing head materials and sub-systems have been assessed for their stability in the presence of nitric acid and ionising radiation. These include thermoplastics, elastomers, cables and connectors. The electropolishing head unit and integrated sensors will be irradiated to verify that the radiation tolerance of the complete head unit is the same as that predicted by the radiation assessment of the individual head materials.

Two irradiation tests have been planned: one for the electropolishing head (B2) and one for a typical cutting tool used in B5. These tests will be performed in the BR2 reactor gamma irradiation facility at Mol. It uses a Co-60 source with characteristics: 40 krad/hr dose rate, 1 to 10 Mrad total dose and a facility for on-line monitoring of key parameters. The testing facility has been adapted to accommodate the targets; irradiations are scheduled for execution in 1992.

4. Surface decontamination (B3)

A Requirements Analysis for the surface decontamination software was completed and a specification issued in 1990. The basic scanning software has been coded and successfully tested. The design of software:

- to interface to monitor/decontamination tools,
- to log radiation monitor position and output,
- to display the arising radiation map.

has started and is scheduled for completion early in 1992. Once the software has been coded, tested and integrated with the hardware, an inactive demonstration of robotic electropolishing of a flat surface will take place.

5. Clearance monitoring (B4)

The aim here is to develop a robotic system to scan a radiation monitoring head over large concrete surfaces to confirm or otherwise the removal of active concrete. To extend the operating area of the robot, a compliant monitoring head will be deployed at the end of a light weight pole attached to the robot end point. A kinematic analysis of the problem has revealed that the vertical reach of the system is proportional to the compliance angle of the monitoring head. The compliance angle is measured between the surface normal and the pole. Computer simulations show that the maximum reach occurs at a compliance angle of 55 degrees.

6. <u>Glovebox size reduction</u> · (B5)

Tool mounts and cutting techniques have been developed to eliminate robot stalling arising from tool misalignment and to optimise cutting performance. Extensive cutting trials on a wide range of tools has resulted in a definition of the tool set required to size-reduce plutonium contaminated gloveboxes. The proposed set comprises: reciprocating saw, bandsaw, hole saw, jaw gripper, vacuum gripper.

The reciprocating saw was selected as a sensible compromise between the low speed jigsaw and the high speed nibbler. Although the nibbler cuts through 5mm mild steel plate at 1m/s, it does have a large footprint and is rather susceptible to misalignment error. A large footprint negates the high speed cutting performance of the nibbler by introducing the need to cut a large rectangular hole with a jigsaw before nibbling can commence. The reciprocating saw also has the advantage of being relatively untroubled by any internal pipework that might be attached to the cutting surface. Cutting speeds of up to 150mm/min, were achieved during extensive cutting trials. No blade breakages occurred during these trials and blade wear was minimal.

A hole saw with a centre mounted pilot drill has been developed to cut access holes for the reciprocating saw. The pilot drill prevents the hole saw from skidding across the metal plate surface and makes alignment of the hole saw less critical. Hole saw feed rate is controlled by a spring/damper system mounted along the axis of the pilot drill. The tool cuts 38mm diameter holes in 5mm mild steel plate in approximately 90 seconds. Hole saw life is now in excess of 70 holes.

A seven station tool change rack, to hold the tool set defined above plus two spare tools of similar size to the reciprocating saw, has been built and tested. It is based on the proprietry tool change mechanism supplied by Stäubli. Repeatability of the tool change procedure was not affected by: aggressive cutting, robot stalling, robot calibration and interruption of power to the robot between tool change operations.

A glovebox size reduction demonstration facility (Figure 1) has been built. It comprises:

- a slideway mounted NEATER
- a lift-rotate-tilt cutting table (LRT)
- a conveyor belt to transport waste from the cutting area to the waste packing station
- remote viewing system including a stereo camera
- tool change station
- tool set
- input device for teleoperator mode

Five glovesboxes have been size-reduced using the following strategy. Cutting lines are marked on all external surfaces before mounting the box on the LRT table. It is attached to the table via four remotely operated vices. A grid of holes is drilled in each glovebox panel. These provide access for the reciprocating saw which cuts between the holes, removing plates of suitable size for the 200l waste drum. Metal supports are cut into pieces of suitable length by a bandsaw. The adopted cutting strategy avoids prematurely weakening the box and thereby making sawing operations unduly difficult. Debris collected on the floor of the glovebox is transferred by the robot to the conveyor belt for transmission to the waste packing station. The size reduction process is then applied to the floor pan. A glove port is used to pack the waste manually into a 200l drum. Another glove port is dedicated to the tasks of changing saw blades and drill bits.

As all the gloveboxes are different, most of the decommissioning tasks are performed telerobotically through twin joysticks or a force reflecting input device. Repetitive tasks, such as tool change, waste transfer to the conveyer belt and presentation of the tool to the blade change port, are performed in robotic mode. Changes between teleoperator and robotic mode are initiated through the TRC's menu driven touch screen.

The main conclusions drawn from the glovebox decommissioning demonstration are:

- operators, who normally decommission gloveboxes manually, liked the telerobotic method,
- there was no need for manual intervention to free jammed tools etc,
- operators found the stereo cctv system to be helpful but preferred to view it in the seated position,
- the size-reduction strategy worked,
- tool alignment times are still too long,
- the hand held input device would benefit from a harness to reduce operator fatigue.

References
/1/ DECRETON, M, Electropolishing Head - Requirements Analysis and Selection of Sensors, CEN/SCK Note TEL: G4002/91-05 (1991).

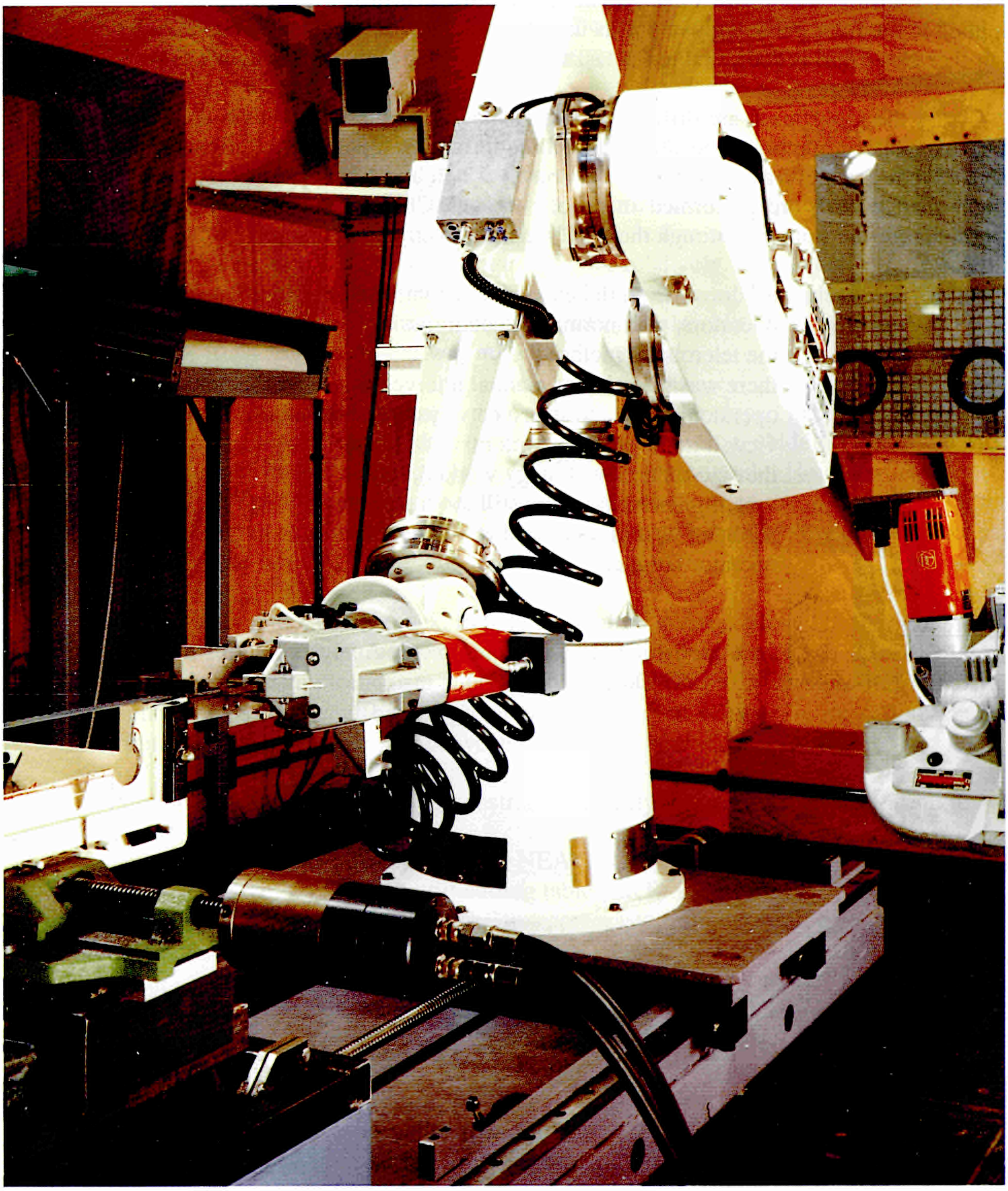


FIGURE 1 - GLOVEBOX DECOMMISSIONING

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 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 specification for the vehicle (B.1.) has been completed. The vehicle is currently being manufactured (B.2.). It is due for delivery to AEA Technology in March 1992. The dry test facility has been constructed (B.3.). Work is proceeding in preparing the test facility for the vehicle development programme (B.4.1.). Use of the vehicle for fuel recovery operations (B.5.) is forecast for February 1993.

Progress and results

1. Specification of the underwater vehicle, to allow manufacturers to tender. (B.1.)

This work package has been completed. The vehicle and manipulator will be capable of working at a depth of 6m of water. The reach of the manipulator will be 1.5m and the manipulator will have the ability to position fuel on the back of the vehicle.

The lift capacity of the manipulator arm will be 30Kg at full extension. The gripper for the manipulator will have force control variable between 0 and 500N to enable delicate objects to be retrieved

Access to the Water Duct is restricted. The vehicle will be capable of being deployed through a hole measuring 1.8m x 2.7m maximum.

The vehicle umbilical cable has been designed so that it can be used for recovery of the vehicle if it should break down in the water duct.

2. 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 is programmed for delivery to Windscale site in March 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 production vehicle and adapt it to work in the water duct.

The manufacture of the vehicle is on schedule and within the forecast contract cost.

3. 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 facility is a steel frame with plywood panels affixed. It is now envisaged that a separate wet test facility will be built at a later date for the programme of wet development.

4. Adaptation and testing in the mock-up of the vehicle/system on its ability to perform the various decommissioning tasks.(B.4.)

4.1 Dry testing of the complete system.(B.4.1.)

Work is in progress in preparation for the vehicle development in the Dry test facility. A number of areas are being addressed:

(1) Camera systems: An overview camera mounted on the rear of the vehicle will monitor all vehicle operations. An additional camera will be mounted on the manipulator to give a better view of the gripper during operations.

(2) 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. Suitable devices for the measurement of these forces are currently being designed.

(3) 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 is being designed to measure the gripper force.

(4) Vehicle detection system: Investigations are in progress to ascertain if suitable sonar devices are available to enable the distance that the vehicle has travelled along the water duct to be measured. This will assist in the record keeping of fuel element locations when they are recovered.

(5) Simulation of water duct silt: A simulation of the water duct silt is being prepared. this will be used in the wet test facility to study dispersion effects of the vehicle and problems with ingress of contamination.

4.2 Wet testing of the complete system.(B.4.2.)

The design and manufacture of the wet test facility has not yet started. The facility is required to be available by June 1992.

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. Adaptation of tools and experiments

B.3.1. Setting up of a representative test mock-up (KfK)

B.3.2. Tool integration and testing (All)

B.4. Data evaluation

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 1991, the components and the coupling systems were designed concerning the defined stations. The Test bed layout was discussed in more detail. All components are ordered and under manufacture now. Test site preparation has started.

Progress and results

KfK activities: All specification work was solved as the tool attachment and tool exchange systems, the couplings, the enhancement of hydraulic circuits and replacement of arm 5 by a wrist.

All partners agreed on coupling components and wiring and cabling modifications as well as special components such as cooling water supply etc. The concrete plates have been ready since June 1991.

KAH activities: KAH took care of all design; constructional and optimization work between the partners, beside their own components construction and manufacture.

Tool adapter plates; tool store and replacement equipment is under manufacture now (Figures 1 and 2).

AEA-Harwell: The microwave device is under manufacture and will be delivered begin 1992. Places for mechanical and electronic equipment are settled on the test site, as well as the place for demonstration of the component. In order to avoid high leakage of microwave, the demonstration place was found near the bridge, where leakage is at least partly absorbed by surrounding soil walls.

The official licensing of this device is due to field tests in open air with stepwise increasing power on place.

BAJ: The contamination measurement array is under manufacturing now. In parallel, new detector types are being tested at BAJ, so as to replace the chosen ones if better.

The contamination test array was settled in implementing radiation capsules on one of the test plates in an unknown pattern to BAJ in order to avoid licensing trouble by using free field tracing.

<u>Test bed</u>: Test bed parameters are set up in more detail for assessment of the different tools. During tool integration and testing, the ultimate parameters will be fixed (Figure 3).

WORK PROGRAMME TIMETABLE:

- The work schedule for B.1.A. up to B.1.11. has been solved, as foreseen in 1991.
- The work of B.2.1. to B.2.14. is ongoing and will end within the next 3 months.
- The representative test mock-up is being constructed (B.3.1.), as well as the microwave shielding, which will be set up by early 1992.

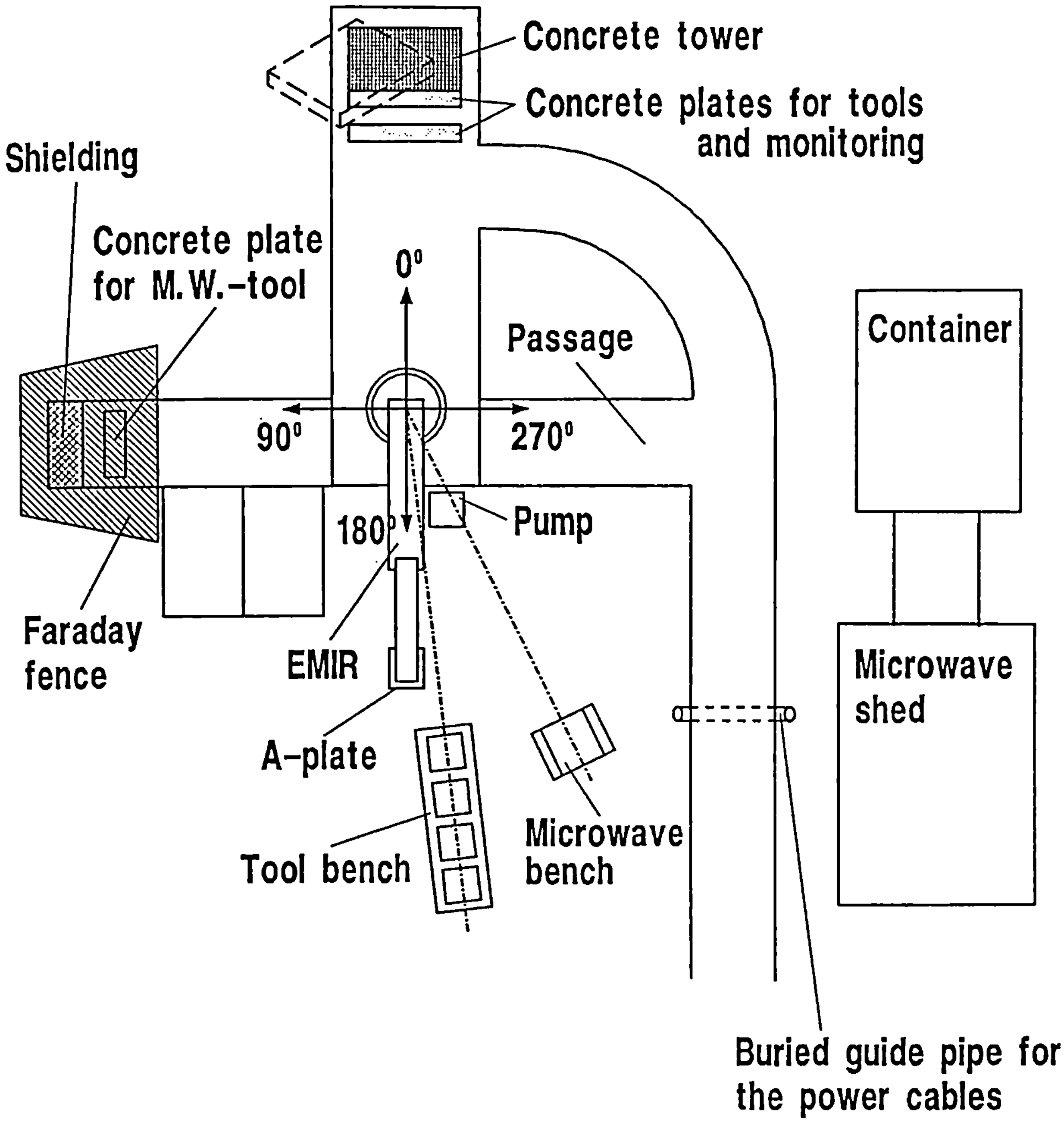


Fig.1: Layout of EMIR site and testbeds

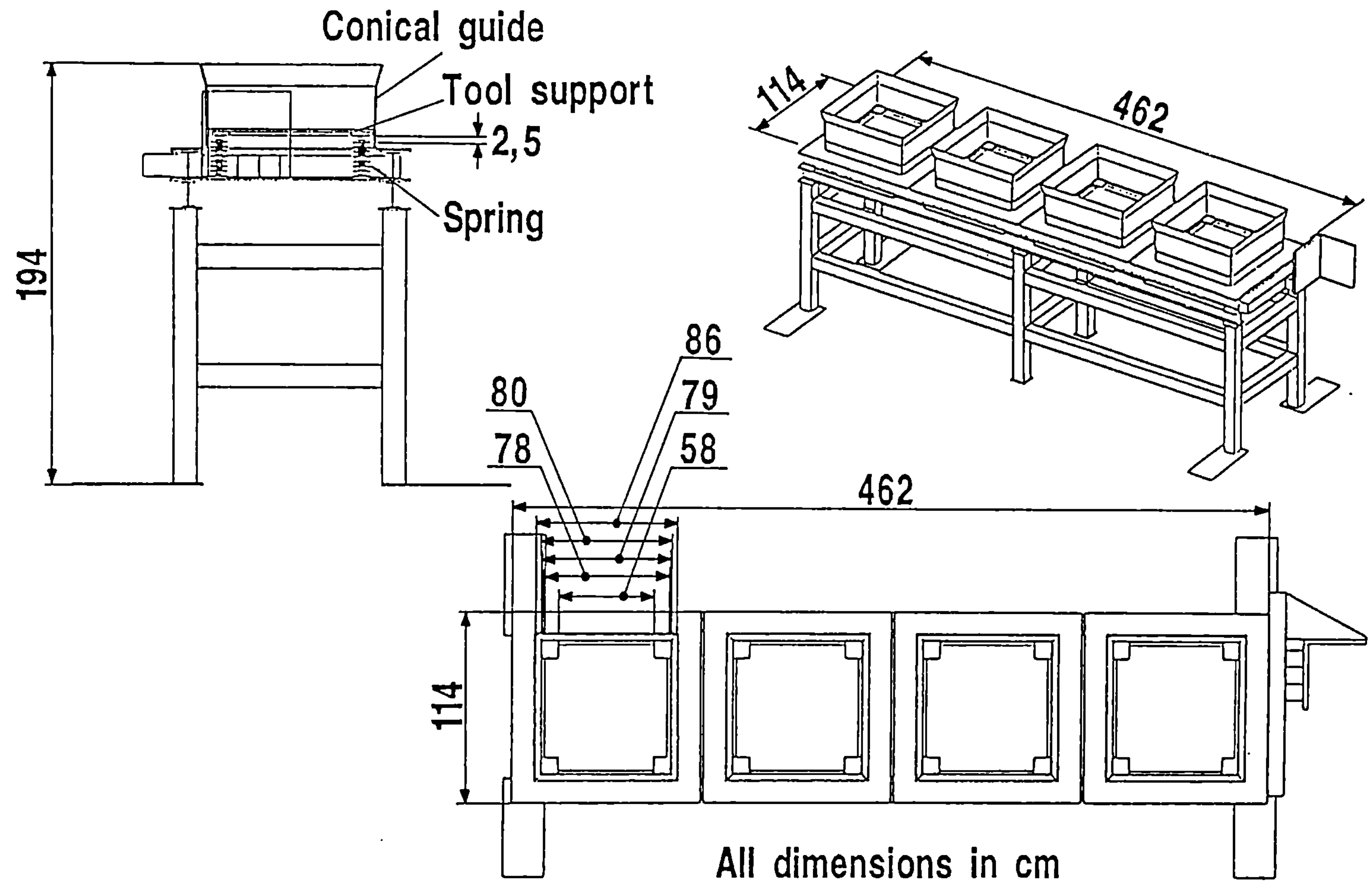
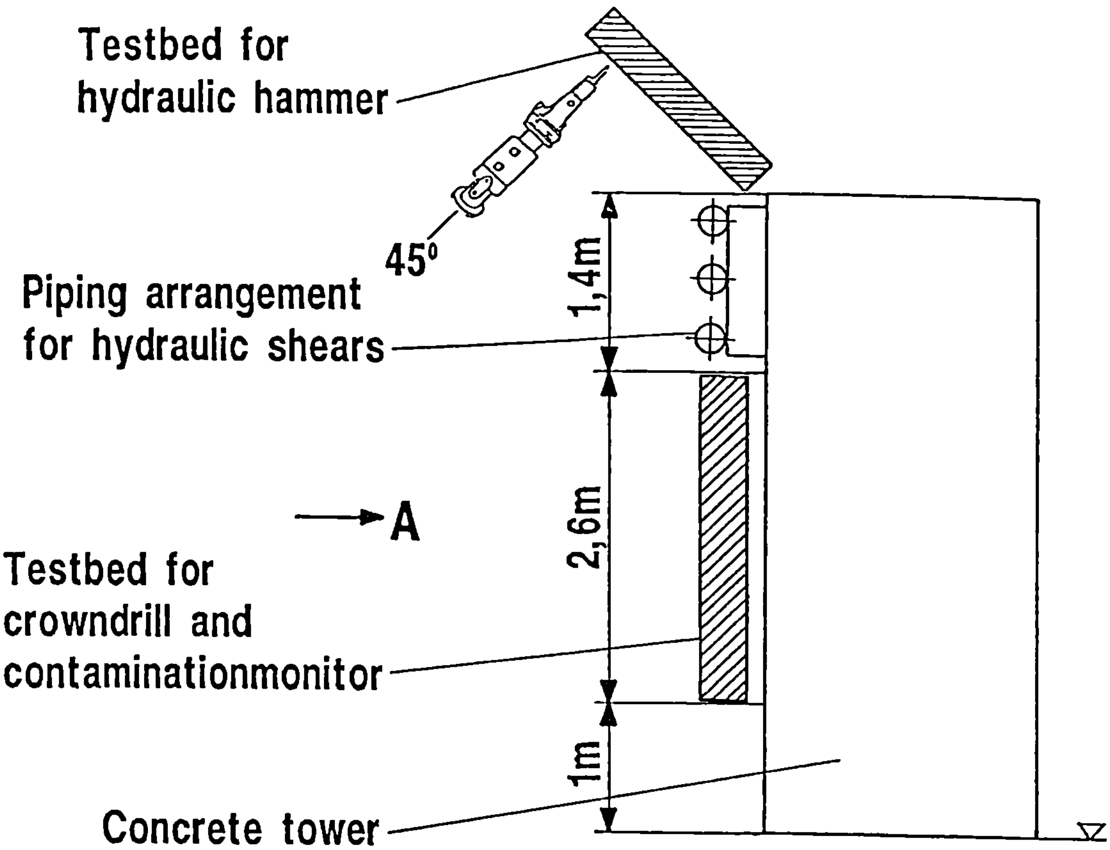


Fig. 2: Tool bench



Testbeds mounted on the concrete tower

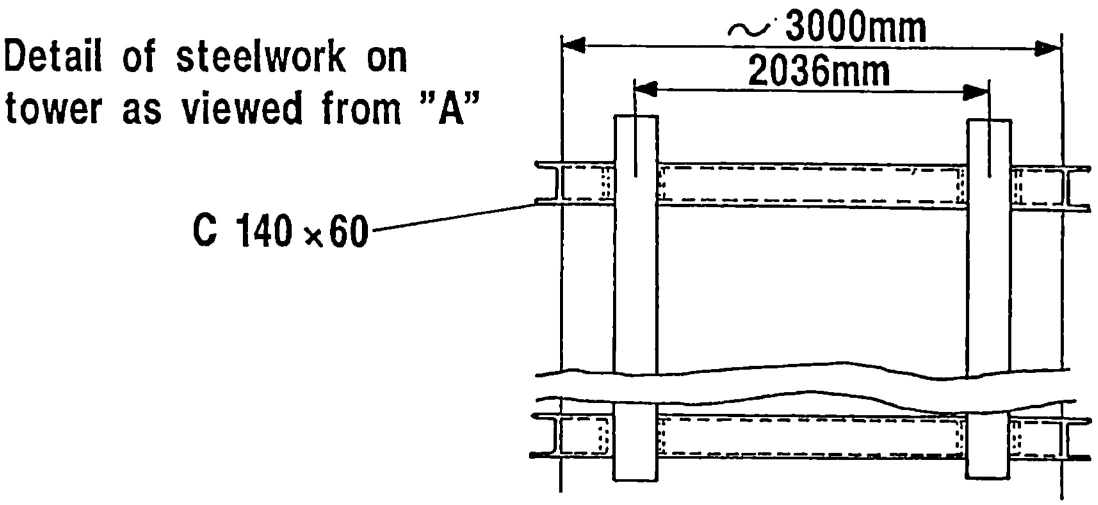


Fig.3: Disposition of testbeds on the concrete tower

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: G CLEMENT, CEA/DTA/UR, CEN/FAR
Phone: 33/1/46 54 91 16 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 OBTAINED

Summary of main issues

Underwater qualification of the RD500 manipulator and of an ultrasonic imaging system, to carry out dismantling tasks, is envisaged.

In particular, the identification of general tasks to be performed underwater, taking into account the hypothesis presently made for the dismantling of internal structures of PWR reactors and the French RAPSODIE reactor, has been performed.

Three dismantling tools have then been chosen, which permit to perform most of these tasks and to qualify the robot, together with three representative mock-ups to carry out the qualification tests. These mock-ups and tools are the following:

- a mock-up of lower internal structures of a PWR vessel and an electro-erosion tool,
- a mock-up of the thermal baffles of the outlet sodium pipes of RAPSODIE and a plasma torch,
- a mock-up of the bolted connection of the grid on the RAPSODIE vessel and grinding tool.

The ultrasonic imaging system, aiming at the provision of an image when the classical vision is disturbed by environmental conditions due to the tools operation (dazzles, bubbles...), has been specified.

The tools and mock-ups have also been specified, as well as the qualification site, in a pool located at NANTES (France).

The RD500 improvements, which are necessary to carry out the underwater tests (pressurization equipment, waterproof umbilical cable...) have been defined.

Finally, a preliminary qualification procedure document has been established.

Progress and results

1. Identification of relevant underwater tasks (B.1.)

The relevant underwater tasks have been identified considering the current hypothesis made regarding the dismantling of PWR reactors and of the French RAPSODIE fast reactor. In the second case, it seems advantageous to perform the in-vessel operations under a water shielding protection, after filling up of the vessel, as in a PWR reactor, naturally in water. This being the case, it is easy to identify some generic tasks in both cases, which are the following:

- cutting tasks, i.e. cutting of small tubes, eventually concentric (small) tubes, cutting of bolts, assembling pins, crossbars, ... and cutting of steel sheet-pieces of various thicknesses;
- unscrewing of screwed parts; when it is not possible to cut as in the previous case (for example bolt head lost in the piece that it locks), it is necessary to unscrew the assembly to ensure its dismantling. In the case of the so-called "baffle assembly" of the lower internal components of a French PWR, bolts of this type, in great number (about 1000 for a baffle assembly), are secured by a welded key, so that it is necessary to remove this key or to destroy the bolt head in its hole. This operation is usually performed with an electro-erosion technique.
- handling tasks, including the welding of lifting eyes or clamps.

2. Selection of appropriate tooling systems (B.1.2.)

Three tooling equipments are selected to carry out the qualification tests with the robot, on three representative mock-ups, in non-active environment. The choice of these tools is made considering the criteria of number of potential applications of each tool and the qualification of RD500, that the tools will permit to perform.

Following these criteria, the three tools are the following:

- an electro-erosion tool, because of the large number of bolts to be unscrewed in a PWR reactor, as already mentioned above, and because this tool will permit to qualify the characteristics of accuracy and resolution of the robot.
- a plasma torch because it enables to cut number of components, tubes or sheets of various thicknesses, and because it will qualify the semi-automatic control modes of the robot: trajectory control and stability at low speed, notably.

- a grinding disc because it is a polyvalent tool, and because it will allow to qualify other semi-automatic control modes, particularly the locking of degrees of freedom (for example to perform a straight grinding operation) and the force control during operation.

The electro-erosion process consists in metal machining by an electrical discharge, in a dielectric fluid, between the workpiece and an electrode. The electrode is made of graphite. The metal fragments released by the process are evacuated by a fluid circulation around the workpiece, provided by a pump of the electro-erosion tool. Presently, this tool has dimensions 330x80x250 mm and a weight of about 25 kg. Figure 1 and Figure 2 show the process and a simplified sketch of the tool.

The plasma torch and the grinding tool are more classical and easily available; two tools, available at Fontenay-aux-Roses, will be used.

3. Definition of test mock-ups (B.1.3.)

Three representative mock-ups of components or internal structures of systems to be dismantled have been chosen, to carry out the qualification tests with the robot and the tooling equipments defined above.

For the electro-erosion tool, the choice of operation to perform enacts practically the mock-up, of which only the dimensions, support must be defined. In fact, the choice of representative parts of the "baffle assembly" is obvious. The bolts are equal to the real ones (material, dimensions), and are mounted on "replaceable" parts, to be changed easily on the mock-up.

For the other tools, mock-ups of representative parts of RAPSODIE have been chosen:

- for the grinding tool, the connection of the grid on the vessel, made by assembling pins, placed regularly around the grid; the access conditions, laterally with the limitation of the vessel surface, as well as the real dimensions and materials, are reproduced.
- for the plasma torch, the thermal baffles of the outlet sodium pipes of the reactor are considered; these baffles are made of four sheets, of thicknesses from 5 up to 10 mm, separated by water slides. As well as for the previous mock-up, real dimensions and materials are reproduced.

4. Specification of the testing environment (B.1.4.)

The definition and specification of the testing environment in the pool where the qualification tests will be performed, are carried out. The equipments to be considered are the following:

- the robot RD500, master out of the pool and slave in the pool, with all cabinets, umbilical cable,
- three mock-ups, previously defined,
- three tools,
- the vision equipment including the classical and the ultrasonic systems.

A layout of the pool is proposed (particularly with the robot and three mock-ups simultaneously). This layout is preliminary and could be reviewed (separate tests with each of the three pairs mock-up and tool).

5. Specification of RD500 adaptations (B.1.5.)

The RD500 has been conceived to operate in underwater conditions: nevertheless, this must be checked and refurbishments have to be made. In this way of looking at things, the following adaptations have been defined:

- assessment and refurbishment, if necessary, of the watertightness of the arm,
- installation of a pressurization equipment (air governor, manometer, escape valve...),
- realization of a watertight umbilical,
- adaptation of the present connector (not watertight; it must be included into a tight box with the extremity of the new umbilical);
- tests after installation, to verify the tightness (in air, then under water).

6. Specification of vision systems (B.1.6.)

The specification of vision needs concerns both classical vision and ultrasonic vision.

For the first one, there is no particular difficulty and what is presently defined can be modified if necessary. Three cameras have been identified: a camera to give a general insight of the scene, and two cameras mounted on the robot, at the level of the shoulder and on the arm.

For the ultrasonic vision, the following work has been achieved:

- identification of possible applications of the ultrasonic vision, in the case of dismantling. For the plasma torch, the inspection of the plasma cut has been foreseen. We know that the plasma cutting may not be 100% effective, certain parts being not fully separated. Therefore, ultrasonic measurements could be operated during movement when moving backward the robot (after cutting), along the plasma cutting track. This approach is foreseen because of the production of bubbles and temperature gradients during the cutting process, which could strongly influence the quality of the measurements. Nevertheless, on-line inspection will also be investigated.

For the electro-erosion technique, ultrasonic measurements will be investigated as means to control the positioning of the tool head, which must be very accurate.

For the grinding tool, no particular application of the system has been foreseen:

- a pilot experiment has been carried out, using an acoustic sensor available at TNO, on a model of a bolt mounted on an aluminium plate. The images obtained (see Figure 3) show a good resemblance with the optical image, but they are the best obtainable, and time involved in acquiring this data significantly exceeds the requirements of the ultrasonic imaging system. Optimization is then still necessary.
- the specification of the ultrasonic imaging system has been carried out, in terms of resolution, data acquisition geometry (this refers to the design of the sensor arrays), environmental conditions and imaging performance.

7. Drafting of a qualification procedure document (B.1.7.)

The preliminary qualification procedure document is based on the realization of three steps of tests:

a) preliminary tests of each tooling equipment, and of the RD500. These tests aim at the verification of the correct operation of each tool and the assessment of the main machining parameters (for the tools). For the RD500, the tests will mainly allow the verification of the watertightness of the arm and of the umbilical: they include the pressurization of the arm up to 0.2 bars and tests to verify the good functioning of the regulator and of the escape valve.
b) integration tests, i.e. tests with the robot and each tool. They are envisaged in a first pool located at Fontenay (pool "MINERVE"). They aim at verifying the operation in real environment (robot, tool, mock-up) and at realizing the first operating tests in these conditions.
c) qualification tests performed in the ATEA pool (subsidiary of FRAMATOME) in Nantes; the scope is to carry out a set of qualification tests with each tool, following a detailed procedure, so as to permit the specification of an industrial dismantling system for underwater operation based on the use of the RD500 robot.

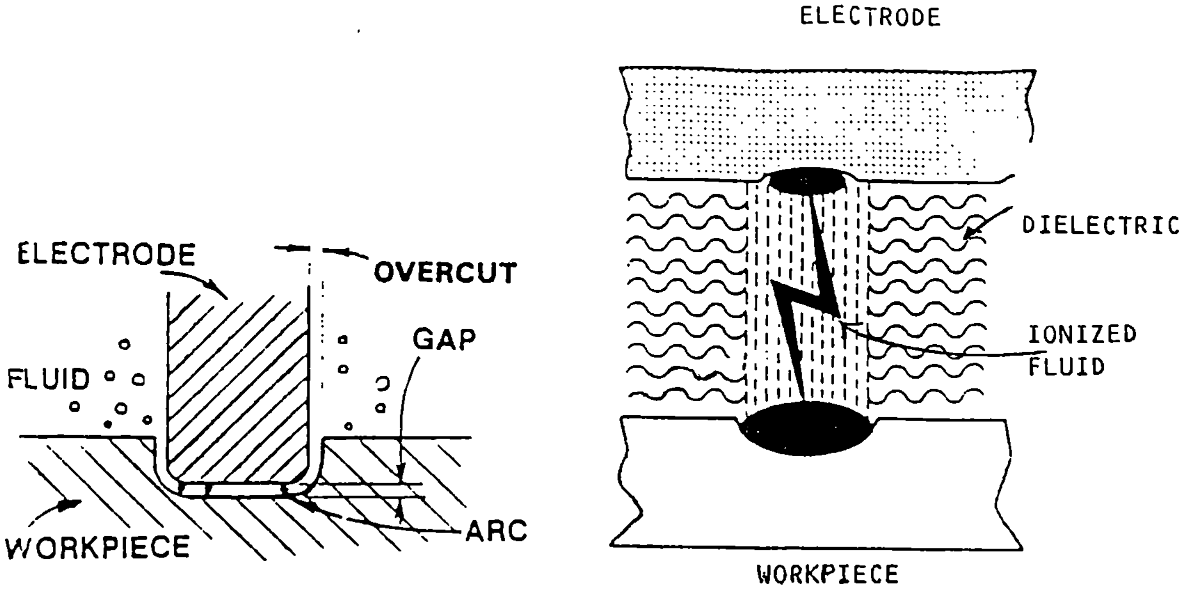


Figure 1 : Process of electro-erosion technique

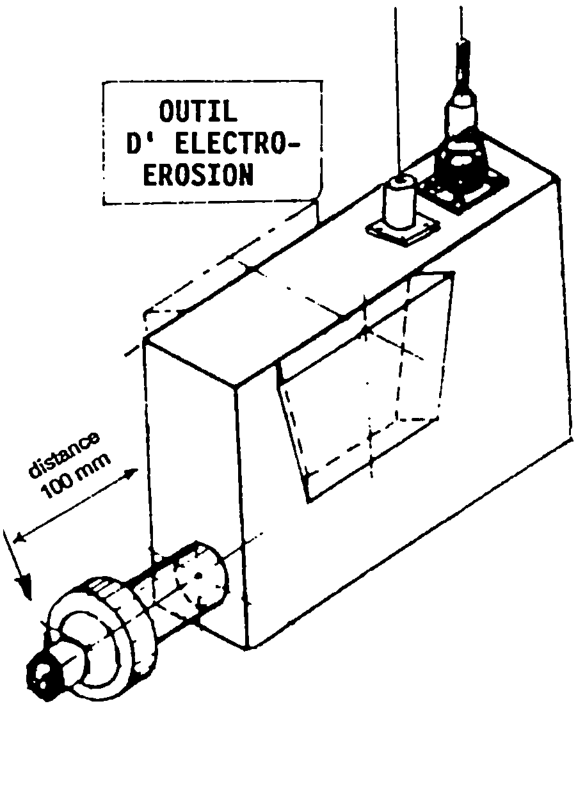


Figure 2 : Sketch of the electro-erosion tool

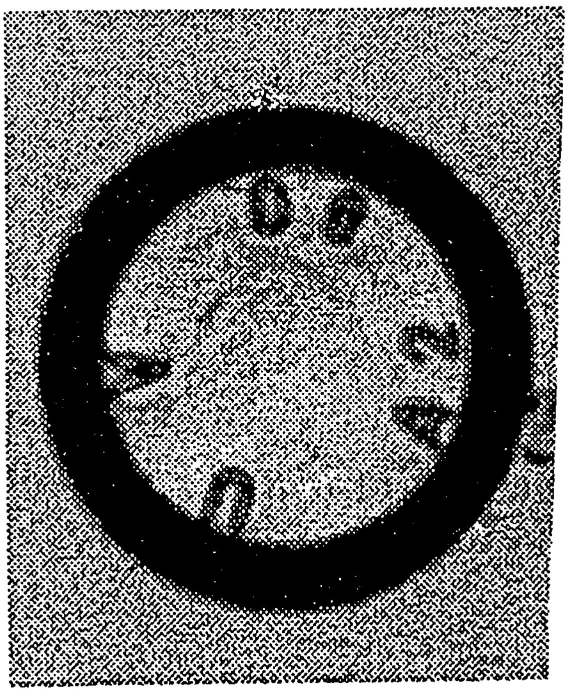


Figure 3 : Ultrasonic image of a bolt in its ϕ 30 mm hole

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

At the end of 1991, nine research contracts relating to Area No. 6 were at the stage of execution.

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 - September 1992
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 obtained results

Summary of main issues

Part A:

Substantial progress has been made on Task B.1A (development of a methodology for evaluating risks); however, some the work will not be finalised until Task B.8A (demonstration of methodology) is underway. Task B.2A (identification of waste stream) has been completed, and Tasks B.3A (definition of waste inventory) and B.4A (definition of waste management options) are close to completion, subject to further data requirements as other tasks proceed. Work on Task B.5A (estimation of financial costs) is underway, but cost estimates have not been finalised. Work on Task B.6A (estimate of radiological impact), has not yet begun, but is expected to be complete by June 1992.

Part B:

Much progress has been made on all except the final task. Task B.1B (definition of decommissioning phases) is complete. Task B.2B (identification of decommissioning techniques) is on-going; new data on dismantling and decontamination methods and their associated resuspension factors will be collected as they become available. Task B.3B (identification of risk assessment procedures),Task B.4B (evaluation of risk assessment procedures), and Task B.5B (risk aggregation) are nearly complete. Work is well advanced on Task B.6B (methodology application). Task B.7B (final evaluation) has not been started; completion of this task will occur towards the end of the project when all other tasks are complete.

Progress and Results

Study A:

1. Development of a methodolgy for evaluating risks (Task B.1A)
Decommissioning waste management decisions require consideration of radiological risks to workers and the public, and of risks to individuals and society as a whole. An existing NRPB methodology for calculating these risks /1/ has been updated to include risk representation by multi-attribute decision analysis and also the new recommendations of the ICRP /2/. Although substantial progress has been made on this task, it cannot be considered finished as the methodology may require updating to address particular issues that may arise in the example.

2. Identification of waste stream (Task B.2A)
A number of waste streams will arise from the decommissioning of WAGR (Windscale Advanced Gas-cooled Reactor), and these include concrete from the bioshield, graphite from the moderator and reflectors, the reactor pressure vessel and its insulation, the neutron shield and the thermal shield.

Graphite has been selected as the stream most suitable for use as an example, after consideration of such factors as availability of good quality data and alternative waste management strategies. There are 285 tonnes of active graphite within WAGR, where it is the major constituent of the moderator, the reflectors and the neutron shield. The moderators and the reflectors fall into the category of ILW (Intermediate Level waste), whereas the neutron shield is partly ILW and LLW (Low

Level Waste). This gives the option of disposal entirely as ILW or segragation into ILW and LLW for disposal.

3. Definition of waste inventory (Task B.3A)
Estimates of the activity of the graphite components have been made using the computer code ANISN /3/. These suggest that the beta/gamma activity is due principally to C-14,H-3 and Co-60, and this is supported by radiochemical analysis of samples taken from the moderator. A complete list of the isotopes considered in the ANISN calculations is given in Table I.

Although a number of graphite samples have been analysed previously, these have been taken from various reactor channels, subject to various neutron fluxes, and at different times and by different organisations. It is therefore difficult to quantify errors and compare methods, and thus activity estimates are approximate. Work will shortly be carried out to standardise sources and calibration methods in order to arrive at better estimates.

4. Definition of waste management options (Task B.4A)
Due to the imminent start of WAGR remote dismantling (March 1993), there are no longer any alternative feasible management options available for the assay, packaging and storage of waste.

Development of the chosen option is progressing well, with all major plant built and installed, and minor equipment undergoing trials. In summary, a representative proportion of the graphite bricks segregated from the reactor will be analysed by gamma spectrometer to determine the waste category (ILW or LLW). Two sets of cores will be taken from these, one for further analsysis at a dedicated laboratory and the other for retention in an archive store. ILW will be grouted into containers, and stored until an underground repository is ready, and LLW will buried at Drigg land burial site.

Waste management options which are not open for WAGR for historical reasons, such as long term deferral of dismantling, may also be studied in Task B.8A, in order to show particular features of the methodology.

5. Estimation of final cost (Task B.5A)
Work on this task is currently underway. Preliminary estimates of the costs of graphite waste management options have been made, but are not yet finalised.

Study B:
6. Definition of decommissioning phases (Task B.1B)
The IAEA definition of decommissioning phases (Stage 1: storage with surveillance, Stage 2: restricted site release, Stage 3: unrestricted site release) is the most commonly used definition. However, except for general discussion these categories are rather vague, and may not adequately distinguish between the state of various plant.

Suggested scheme for risk assessment purposes is to modify IAEA Stage 1 definition into 3 distinct phases; Stage 1a: removal of working materials, Stage 1b: decontamination down to levels for surveilllance, and Stage 1c: decontamination down to levels for alternative use.

It is recognised that there may be occasions where the definition of additional decommissioning stages is helpful.

7. Identification of decommissioning techniques (Task B.2B)
General decontamination and dismantling techniques are described and discussed, including chemical and mechanical decontamination methods, and concrete and steel removal and cutting methods.
All are shown to generate contaminated aerosols and wastes to varying degrees when applied. Of principal concern to the risk assessment of decommissioning operations is the aerosol generating capacity ('resuspension factor') of the methods. It is the dispersion of contaminated aerosols through the plant ventilation system, which gives rise to radiological risks to on-site workers and the general public. Data for resuspension factors are currently being collected.

8. Identification of risk assessment procedures (Task B.3B)
In order to assess the total radiological risk of a particular decommissioning plan, risks from potential accident conditions and faults must be taken into account, as well as risks from normal operations.
Methods to identify possible accidents and faults with the potential to cause radiological risks are identified. Suitable methods include Hazard and Operability Studies (HAZOPS), Event Tree Analysis, Fault Tree Analysis, Cause/Consequence Analysis and Failure Modes and Effects Analysis (FMEA). Benefits and drawbacks are identified.

9. Evaluation of risk assessment procedures (Task B.4B)
Many of the hazard identification procedures described in working package B.3B can be quantified, allowing the frequency and consequences of faults/accidents to be calculated. Radiolgical risk can then be evaluated according to the formula

$$\text{Risk} = \sum \text{frequency x consequence}$$

A difficulty with this approach is the lack of accurate data. Plant records relating to accident frequencies and their consequences are scant because very little decommisissioning has yet been carried out, and in many instances approximations must be made which reduce the accuracy. However upper risk bounds can be derived and provided that approximations are consistent, meaningful comparison of risks is possible (Fig.1).

10. Risk aggregation methods (Task B.5B)
Figure 1 shows a suggested methodology for evaluating radiological risk in decommissioning. The radiological risk to representative groups in the community, ie decommissioning personnel, other on-site nuclear workers, and members of the general public, can be evaluated, and alternative decommissioning schemes compared.

11. Methodology application (Task B.6B)
Work is currently underway to apply the methodology (B.5B) to the decommissioning of eleven glove-boxes in a plutonium fuel fabrication laboratory (Laboratory F, B33, Windscale). The

glove-boxes are to be dismantled in a semi-portable structure called a Modular Containment System, using hand-held power tools. The operational sequence has been analysed using the HAZOP technique (B.3B), and the contamination control system analysed using FMEA (B.3B) to determine faults and accidents that could result in contamination resuspension and release. The risks associated with these hazards are currently being evaluated (B.4B).

References

/1/ Davis, JP,Barraclough,IM, Mobbs, SF, CEC Report EUR 12701 (1990)

/2/ ICRP, 1990 recommendations of the ICRP, ICRP Publication 60 (1990)

/3/ Oak Ridge National Laboratories Report K-1693 (1967)

Table I: Radionuclides considered in ANISN calculations of WAGR graphite

Nuclide	Half-life in years	Nuclide	Half-life in years
H-3	1.2×10^1	C-14	5.7×10^3
Cl-36	3.0×10^5	Ca-41	1.4×10^5
Fe-55	2.7×10^0	Co-60	5.3×10^0
Ni-59	7.5×10^0	Ni-63	9.6×10^1
Nb-94	2.0×10^4	Eu-154	8.8×10^0
Eu-155	5.0×10^0	------	---------

Methodology for Evaluating Radiological Risk in Decommissioning Non-Reactor Nuclear Plant

Milestone B.4B - Evaluation of Risk Assessment Procedures

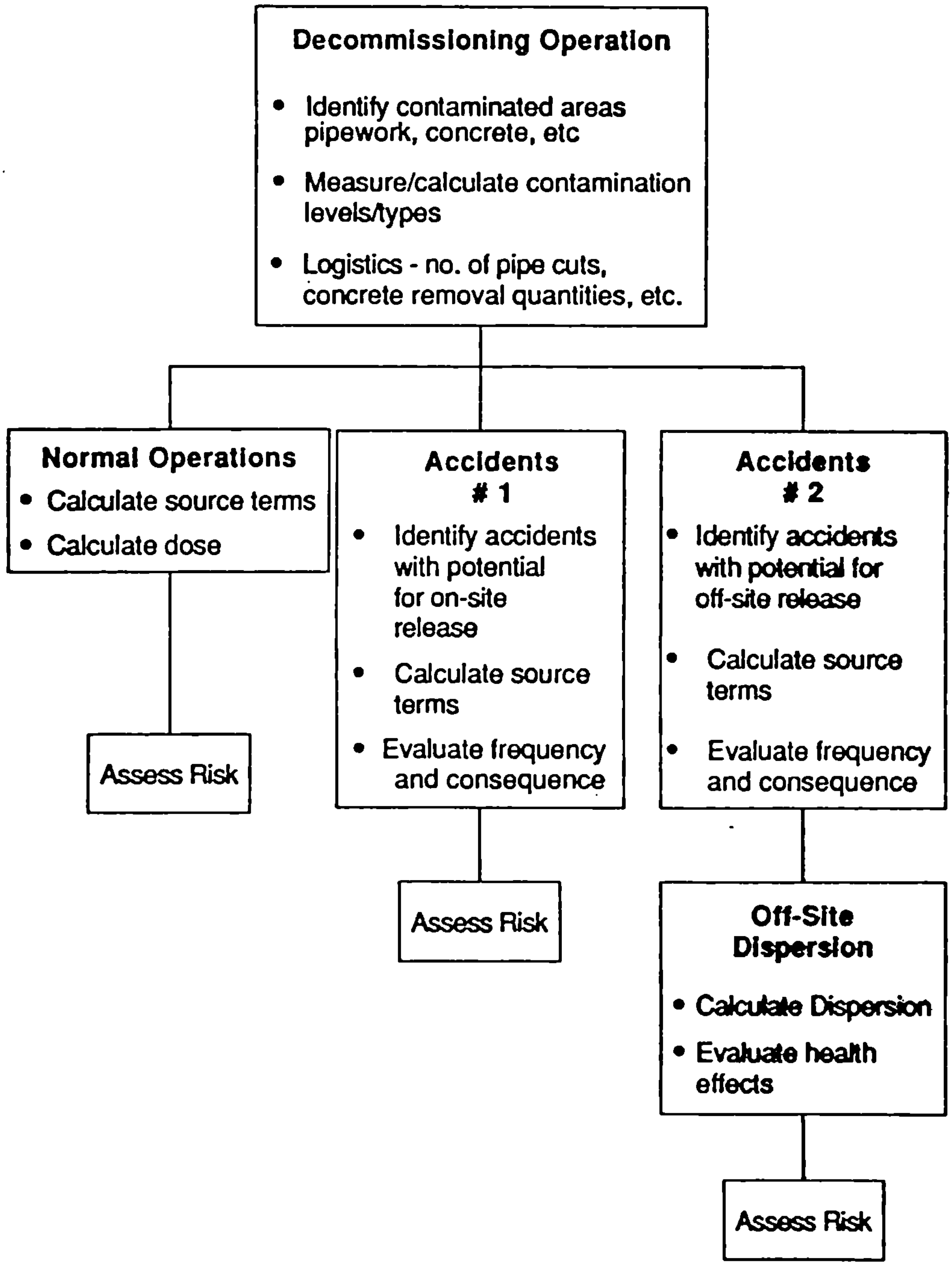


Figure 1

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 1993
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 OBTAINED RESULTS

The metal scrap from a nuclear installation is released to a scrapyard where it is sorted and cut to adequate size before melting. A number of methods are used to reduce the scrap size including shredding, shearing, power saw, grinding and torch cutting. During these processes the workers can be submitted to metal dust emission. It is the goal of this project to study the different steps of scrap conditionning, evaluate the associated nuisances due to dust emission from a potential residual surface activity on the scrap and quantify the potential radiation dose for a person handling the scrap.

Bibliographic survey (B1) An economical and technical overview of the scrap market was done, a scrapyard was visited and the most used cutting means were compared : shearing, the most commonly used, and grinding which is going to get a large importance because steriles are separated from scrap by this mean.

A bibliographic study on the different cutting techniques and the different associated dust emissions was done ; some documents are dealing with currently used mechanical and thermo-mechanical techniques, very few documents are dealing with cutting of surface contaminated metals ; two publications have been studied, the first concerning grinding and plasma torch cutting, the contaminants being cobalt, cesium and plutonium. This experiment shows the importance of the choice of the contaminating technique of the metal ; but the experimental conditions (cuttings were made in a cell with a volume of 0.3 m^3, the air renewal was varying from 133 times per hour before cutting to 33 times per hour after cutting) do not allow to extrapolate the results to working conditions in a scrapyard. The second publication does not precise what contaminant is studied (probably mixed, dominated by cobalt because samples originate from nuclear plant dismantling operations) and the experiment is made in a non ventilated room (again not representative of scrapyard working conditions). The most probable values of the resuspension factor, in this experiment, are in the following ranges : for sawing : 1.7 E-3 to 9.2 E-3 $Bq.m^{-3}$ /$Bq.cm^{-2}$; for torch cutting : 0.07 to 0.74 $Bq.m^{-3}/Bq.cm^{-2}$; for grinding : 4.4 to 5 $Bq.m^{-3}$ /$Bq.cm^{-2}$.

Regulatory situation (B1) The exemption of metal from nuclear facilities in Germany is regulated by the 1987 recommendation of the Commission for Radiological Protection (Strahlenschutzkommission, SSK). The unrestricted exemption of metal from a nuclear facility is allowed if :

1. The mass specific activity is less than 0.1 $Bq.g^{-1}$ and
2. The total surface activity is less than :
   0.5 $Bq.cm^{-2}$ for nuclides like Co-60 and Cs-137
   0.05 $Bq.cm^{-2}$ for alpha emitting nuclides.

In August 1991 a recommendation was made by the SSK for metal coming from the uranium mining works in the former GDR. This recommendation allows a maximum surface activity of 0.5 $Bq.cm^{-2}$ for alpha emitters if the metal is cut to size under radiological control, eliminating the cutting of contaminated metal by hand.

Execution of the experimental programme (B2) Taking into consideration the results of bibliographic studies, realization of representative experiments of scrapyard working conditions seem to be necessary. The different methods used for cutting metals in industry can be divided into two basic categories : mechanical means and thermal means. In this contract, we consider : as mechanical system, a shear and as thermal system, a cutting torch (oxyacetilene torch).

These two cutting means will be tested in a first experiment in inactive conditions. Aerosols of inactive cobalt and cesium are deposited on carbon steel and stainless steel samples. The samples are tubes of 1 m long, 2.1 cm external diametre and 3 mm thick. The aerosol is deposited by thermophoresis, the more realistic method compared to the deposition phenomenon in nuclear reactor circuits. Some carbon steel samples will be oxidized. Those different samples will then be cut by shear and by cutting torch. Information about the aerosol release from the metal and from the deposit on the metal surface is expected. In the following it will be possible to define the needs for an experiment in active conditions.

During the second semester the experiments of aerosol deposition began. Cesium chloride was deposited on stainless steel, on carbon steel and on oxidized carbon steel. Figures 1 to 3 show the evolution of deposit along the tubes for the different types of steel.

The cutting experiments have begun with a 20 tonnes-shear. Preliminary tests showed that the shear did not produce enough dust to work in a large cell ; the experiments will be continued in a glove box.

Another work performed within the report period was to establish and evaluate a first set of experiments for orientation purposes with respect to experimental conditions and the test equipment chosen.

The following conditions are selected :

- material :	austenitic steel, 4 mm, unpainted
- contamination source :	depleated UO_2-powder, contamination on both sides of the austenitic steel plate
- contamination degree :	variation of alpha-contamination in the range of 2 - 18 Bq/cm^2
- thermal treatment :	acetylene torch cutting
- number of tests :	15

Results achieved : It was found that the total release of activity in relation to the cutting length is a linear function of the surface contamination (see figure 4) :

$$F_S = C_{RA} \times O_K$$

alpha activity release (Bq/cm)	$(Bq.cm^{-1}/Bq.cm^{-2})$	(Bq/cm^2)

whereby C_{RA} is the release rate and O_K is the surface contamination. The release rate determined for 4 mm austenitic steel has the values of

$$C_{RA} \approx 2 \times 10^{-2} \quad (Bq.cm^{-1}/Bq.cm^{-2})$$

Measurements to determine the distribution of activity with respect to the different fractions of dust particles show results in the range of the detection limit of the measuring device. For future measurements, modifications of the experimental conditions are envisaged.

<u>Identification of risk (B3)</u> The following rough estimation for torch cutting gives an idea of the expected range in which the doses will fall. This estimation is based on preliminary data. The inhalation risk is characterized by the ratio of the activity concentration in the air (inhalable fraction of aerosol) in $Bq.m^{-3}$ to the surface contamination in $Bq.cm^{-2}$. This ratio is called the resuspension factor R, it can be estimated by :

$$R = (f_r \cdot \tau) / v$$

It is assumed that the aerosol moves upward from the treated surface with an effective velocity of $v = 0.5$ m/s. Aerosol dispersion is not taken into account, which is justified for the area close to the work piece where the worker is located. τ (m^2/s) is the rate of surface treatment (total surface of the material cut per time), typically 0.002 m^2/s for torch cutting of components to a length of $\approx$ 1m. It is derived from a mass treatment rate of 1 tonne/h and an average material thickness of 20 mm. f_r is the fraction of the total surface activity that is released as inhalable aerosol during cutting, assumed to be 1%. This does not apply to volatile radionuclides like Cs-137, which is significantly higher. The value for the release fraction f_r is derived from data obtained during the decommissioning of the BWR at Gundremmingen. The dose from torch cutting can be calculated using :

$$D_{inh} = f_D \cdot V_i \cdot R \cdot A \cdot t = 70\ \mu Sv,$$

where :

f_D = 140 μSv/Bq (dose conversion factor for Pu-239),

V_i = 1.2 m^3/h (breathing rate),

R = 0.4,

A = 0.05 Bq/cm^2 (surface contamination),

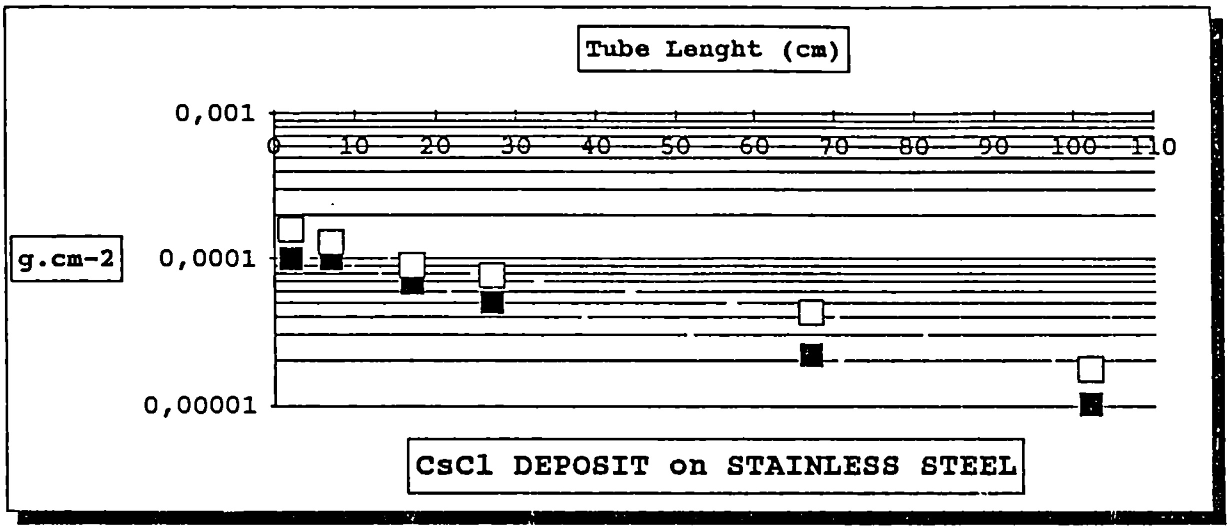
t = 20 h (exposure time).

Stochastic simulation (B5) The preliminary data was used in a stochastic model developed by Brenk Systemplanung to estimate the doses and the number of individuals exposed due to the processing of alpha contaminated scrap. The results are presented in table 1. The values in table 1 are to be interpreted as the average number of individuals exposed to a dose in the indicated range per simulation. Since many simulations were run, an average value of less than 1 is possible. For example a value of 0.1 means that in 1 of every 10 simulations a person received a dose in this range. The model demonstrates that the critical pathway is the manual preparation of scrap (small scrap yard).

Table 1 :

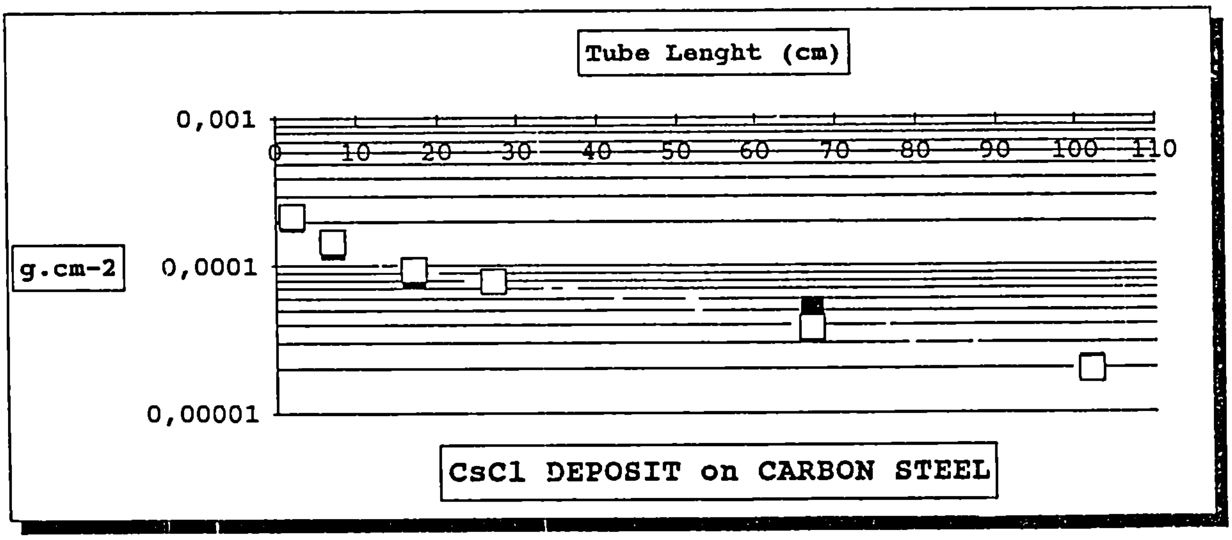
PATHWAY	1 - 10 μSv	10 - 100 μSv	> 100 μSv
small scrap yard	6	3	0.1
large scrap yard	0.01	-	-
foundry	0.4	0.01	-
slag use	0.3	0.01	-

10,000 simulations averaged ; 200 metric tonnes of metal released ; 0.05 Bq/cm^2 as exemption level ; 50% processed by hand (small scrap yard)



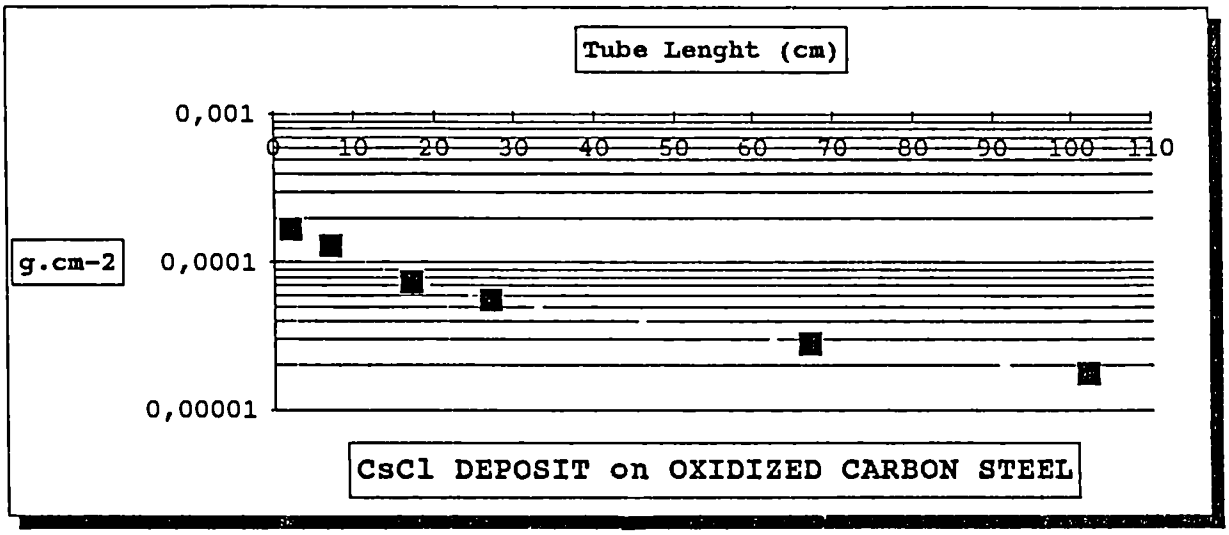
■: experiment 1 □: experiment 2

Figure 1



■: experiment 1 □: experiment 2

Figure 2



■: experiment 1 □: experiment 2

Figure 3

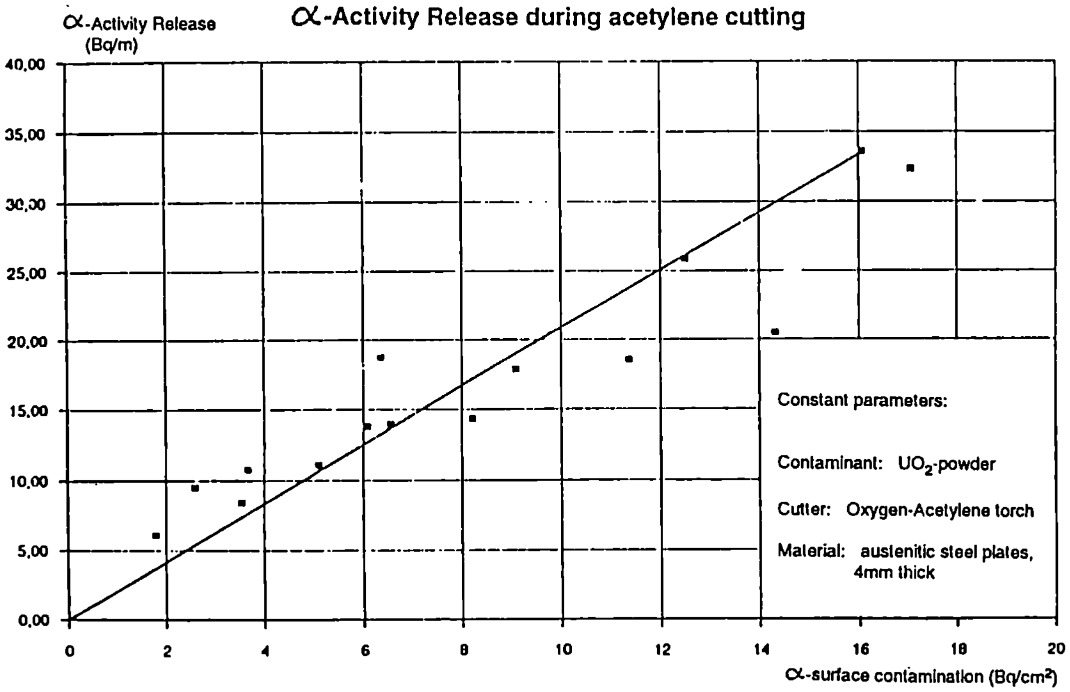


Figure 4

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

After calibration of the coaxial Ge-detector and low-level proportional counter specially for concrete and steel samples the determination of its activity will take place.

The measured samples (steel and concrete) were processed chemically and their content of iron was determined. The samples will then be processed for liquid scintillation counting to determine the contribution of beta-activity of Fe-55.

1. Acquisition of appropriate instrumentation (B.1.)

For the determination of gamma emitting nuclides in samples consisting of concrete and steel it was necessary to calibrate the applied Ge-detector. This spectrometer was calibrated using standard solutions from the Physikalisch Technische Bundesanstalt in Braunschweig with different nuclides and geometries. The above mentioned calibration procedure was necessary to take into account the different sizes and volumes of the samples.

The analysis of beta and gamma emitting isotopes was carried out with an end-window Geiger-Müller-detector and a gas-flow proportional counter.

2. Choice of samples and work procedure (B.2.)

The samples for determination of radionuclides by quick measuring methods were obtained from nuclear power plants during decommissioning.

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). The alpha and gamma surface activity was determined with the end-window Geiger-Müller-counter and the specific activity with the Ge-spectrometer (see fig. 1 and 2). The detection of alpha and beta emitting isotopes was carried out with the gas-flow proportional counter.

The determination of gamma emitters contained in the samples was carried out using long counting times to achieve a detection limit as low as possible. Inspite of low activity inventories of the samples, results with slight errors were achieved. Thirty samples were taken during dismantling of the Niederaichbach nuclear power plant.

The concrete samples were parts of the biological shield and adjoining areas, containing neutron induced activation products.

The determination of alpha-beta-gamma-activity was carried out similar to the samples of the Philippsburg nuclear power plant.

After the chemical separation by the Fachhochschule Gießen the Fe-55 and Ni-63 nuclides will be measured with a liquid scintillation counter at the TÜV laboratory in Stuttgart.

The liquid scintillation counter is currently being calibrated by using standard solutions from the Physikalische Technische Bundesanstalt in Braunschweig.

In addition, different counting cocktails for liquid scintillation must be applied.

3. <u>Laboratory work</u> (B.3.)

The chemical processing of the concrete and steel samples will be carried out at the radiochemical laboratory in the Fachhochschule Gießen.

The samples will be wetly decomposed, chemically processed and run through an anion-exchanger.

After separation of Fe-55 and Ni-63 the samples will be prepared for liquid scintillation by using different counting cocktails depending on different chemical mixtures.

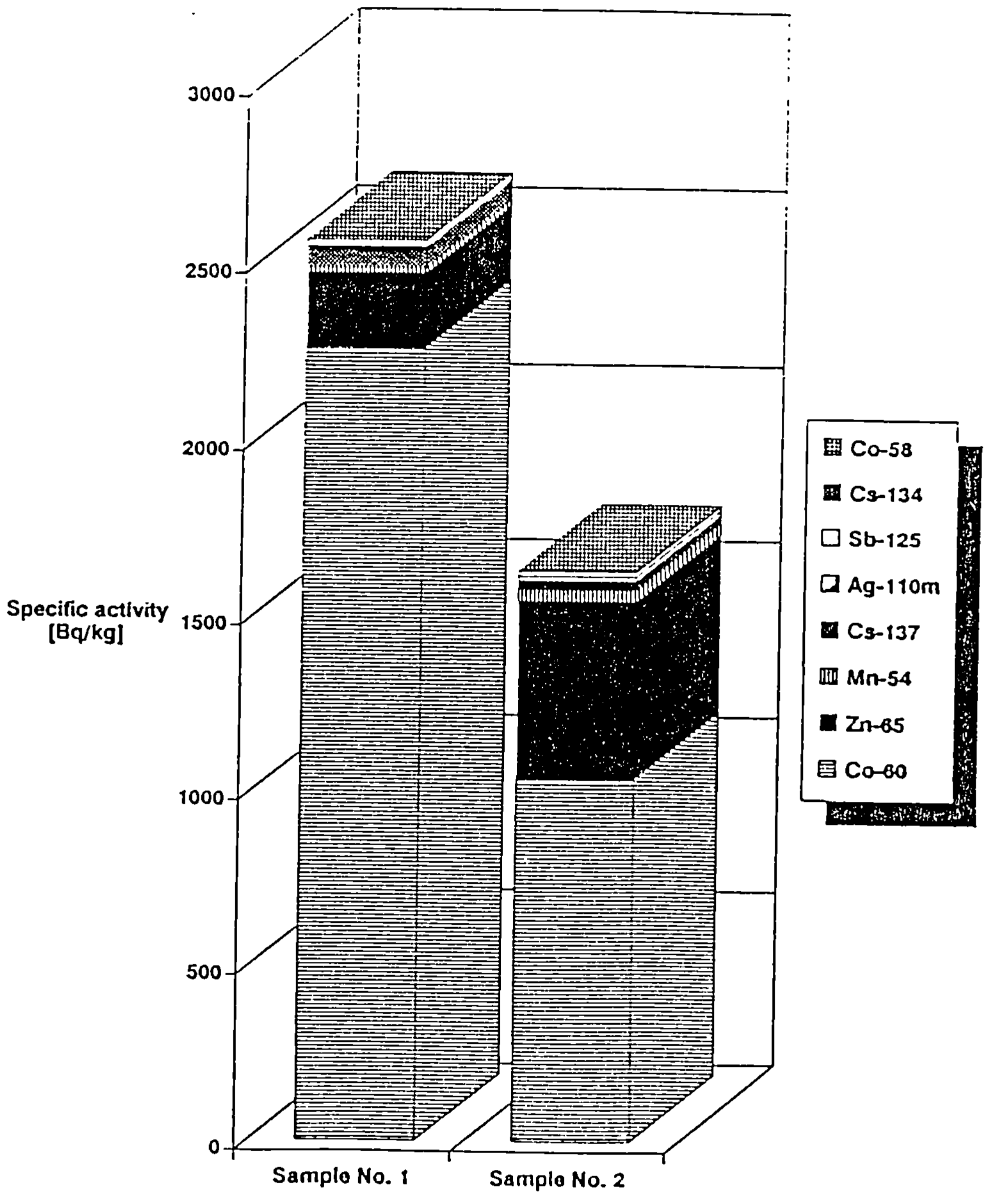


Figure 1: Pipe samples of the reactor cooling system of a boiling water reactor

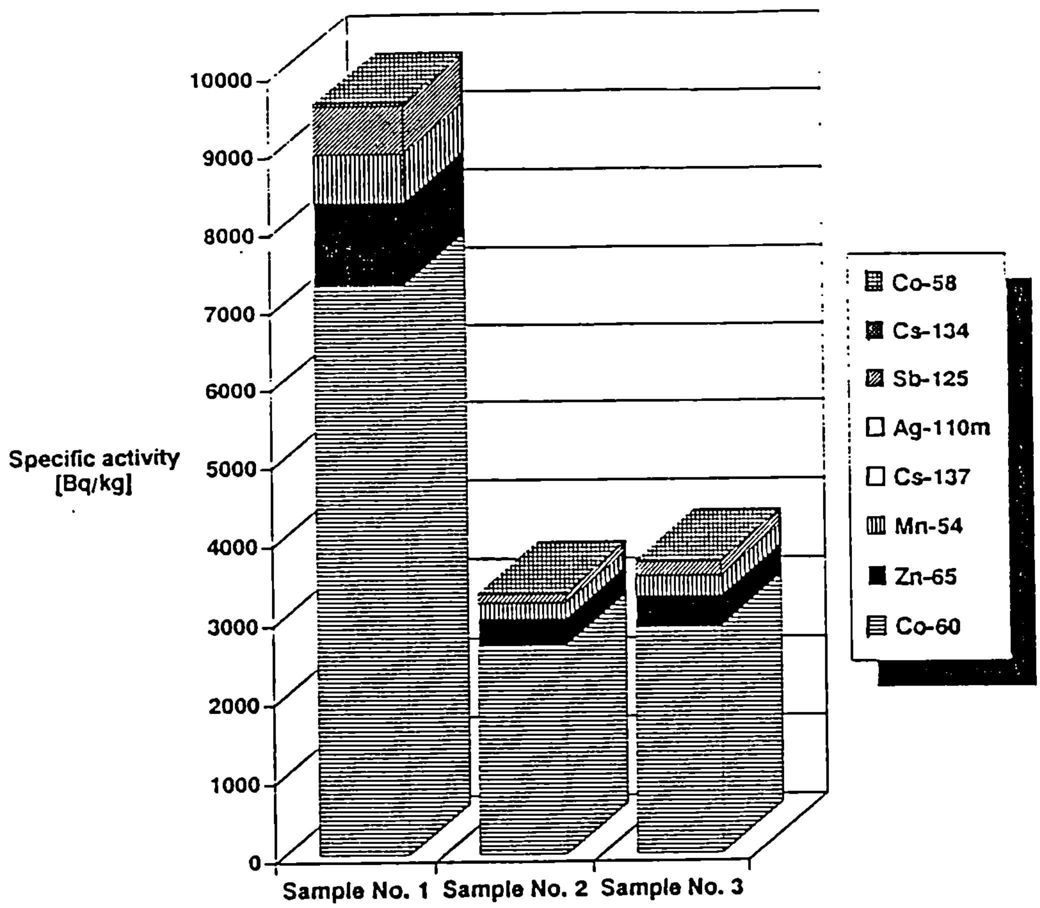


Figure 2: Concrete shielding material samples of a boiling water reactor

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 1551

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

The preparations and planning of the wash-out-tests were finished, the test stand in the nuclear power plant Gundremmingen, unit A (KRB-A) was completed. The tests have been started in the middle of August 1991. Measurements of the test materials and first results of the wash-out-tests were obtained.

The literature study concerning the wash-out behaviour of concrete was completed. A literature study on the composition of natural rainwater was carried out.

Twenty samples of aged concrete and 52 samples of freshly poured concrete were collected in the western part of the Federal Republic of Germany and measured by gammaspectroscopy in order to determine the natural radioactivity of concrete. The literature study, performed in parallel to those activities, examining the natural radioactivity of concrete, was completed.

Progress and results

1. Wash-out-tests (B.1.)

It is the aim of the wash-out-tests to evaluate the applicability of crushed, contaminated concrete as material for foundations, substructures of traffic routes or sound barrier walls from the radiological point of view. Limiting values for the maximum permissible contamination are to be derived.

The wash-out of radioactivity by rainwater is simulated by using concrete test specimen. In the period under review the planning of the test facility was completed and the test facility has been installed in the controlled area of the shut down power plant of Gundremmingen, unit A. In Figure 1 the test facility is shown.

For the wash-out-tests, rainwater, collected at the site of the plant, is used. A literature study showed that the pH-values of natural rainwater in Germany range between 4.5 and 5. Therefore the pH-value of the collected water was adjusted according to these values as well. It is the aim of the investigations to simulate the precipitation of 20 years in a time span of 20 months. The sprinkling is set for two 24 hour cycles a week. Between these sprinkling periods there are two respectively three days for natural drying of the material. The investigations are based on a precipitation of 1000 mm/a. The sprinkling intensity is 5 mm/h thus simulating a heavy shower. Samples from the water, seeping through the test specimen, are taken after each day of sprinkling. The fine concrete aggregates are filtered. These filtered aggregates and the collected seeped water are measured spectroscopically. The easily measurable nuclids Cs-137 and Co-60 are taken as leading nuclids in the measuring procedure.

For the year 1992 we intend to sum up a few sprinkler cycles to one measuring unit as soon as there are no significant changes in the measured values of the radioactivity. All the materials for the production of the test specimen, including the concrete rubble, were gammaspectroscopically tested on their content of radioactivity. Thus the content of radioactivity of every single test specimen is known. The radioactivity inventories of the five test specimen of the wash-out-tests are listed in Table I as far as Cs-137 and Co-60 is concerned. The sprinkler water used was measured as well to exclude a misrepresentation of the test results. These measurements resulted in a concentration of $2.93*10^{-4}$ Bq/g (equivalent to 0.29 Bq/l).

Meanwhile we have gathered the results of the measurements of the seeped water up to the 35th cycle and of the filtered concrete aggregates up to the 10th cycle. These results are shown in Figure 2. The occasionally strongly variing results can be explained by the fact that the values of the radioactivity are near the limits of detection. Sometimes only the limits of detection were balanced.

In 1992 the test will be carried on and first estimates of influence of wash-out-water on the environment will be made.

2. <u>Natural radioactivity of concrete</u> (B.2.)

Up to now 72 samples of concrete from conventional buildings have been collected. 20 of these samples are aged concrete, 52 freshly mixed concrete. Besides the samples are separated in different charges according to the location they have been taken from. (SD: Southern Germany, MD: Central Germany, ND: Northern Germany). The measured activities of the samples are shown in Figure 3. It can be seen that the values of the different samples are spread over a large range but there are no significant differences in the mean values of the charges. The overall mean values of the natural activity of concrete (in Figure 3 simply called mean value) are devided in the contribution of single nuclids or decay series. The obtained values are:

K-40	226.6 Bq/kg
Nuclids of Uranium decay series	14.8 Bq/kg
Nuclids of Thorium decay series	14.0 Bq/kg
U-235	1.2 Bq/kg

The literature study on the natural activity of concrete has been finished. It shows that the values of the natural activity of concrete is in the same range as the ones obtained by our own measurements, but in general higher values are given (up to a factor of 4).

3. <u>Development of methods for recycling concrete</u> (B.3.)

The literature study on recycling methods is not finished yet. It will be finished in 1992 so that its results will be given later.

Table I. Caesium- und Cobalt inventory of the test specimen

	ts 1	ts2	ts3	ts4	ts5
	activity in MBq				
Cs-137	1.9	2.2	1.9	2.2	2.2
Co-60	3.4	4.4	3.4	3.9	3.9

ts: test specimen

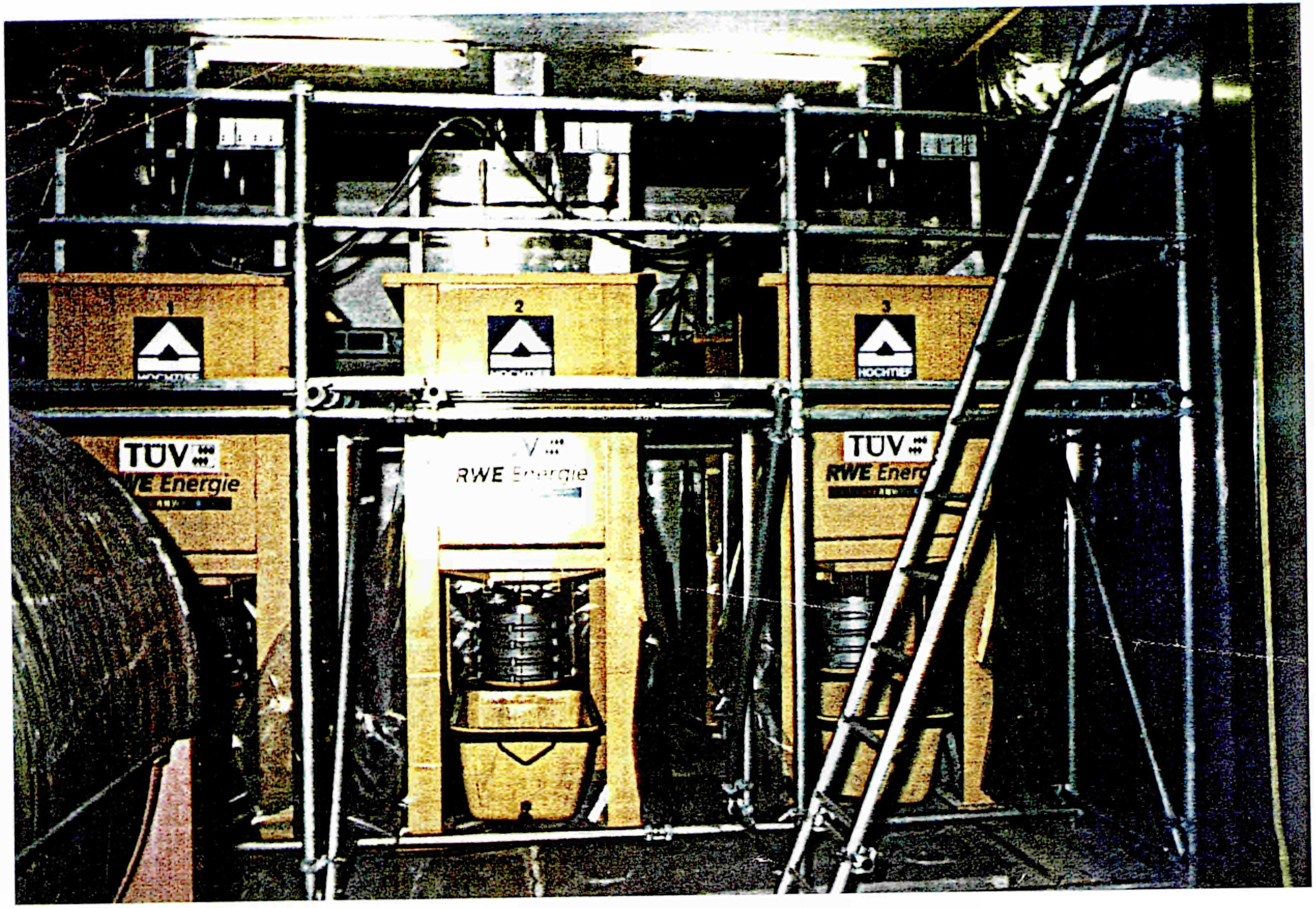


Figure 1: test facility

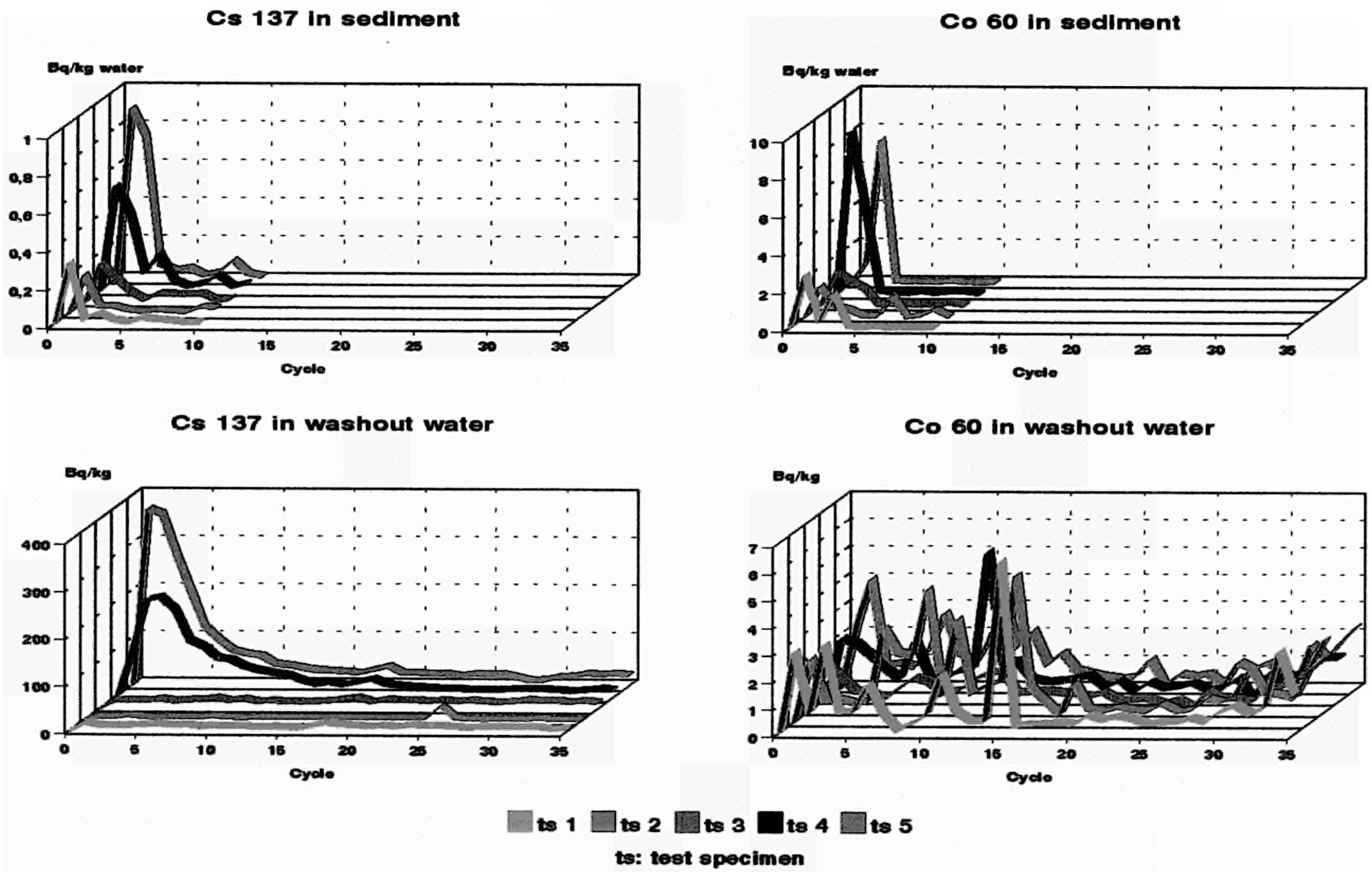


Figure 2: Washed-out activity

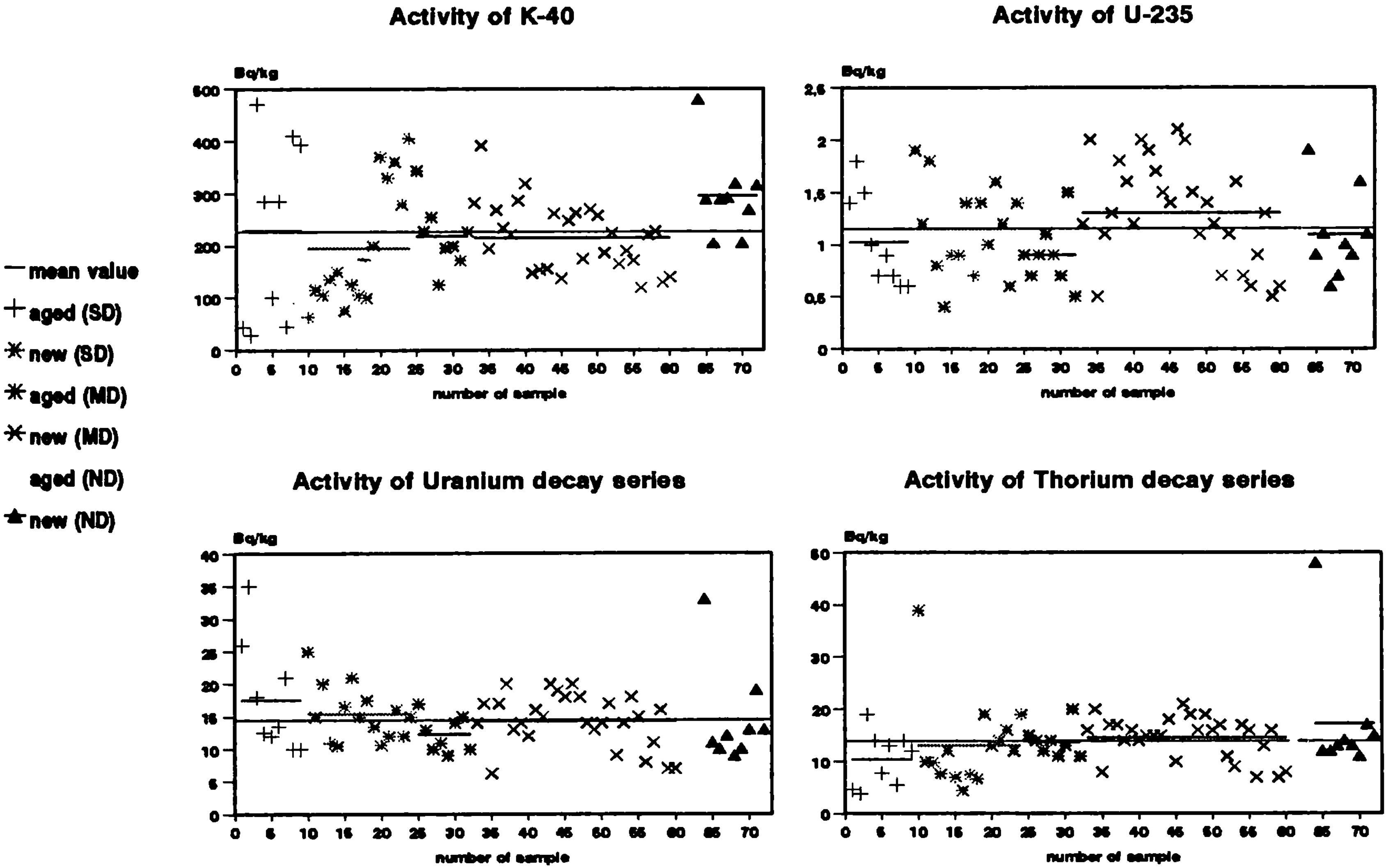


Figure 3: Natural activity of concrete

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) Estimation of contaminated and activated concrete quantities (B.1.1.).

A litterature review on the mass resulting from dismantling of nuclear installations, including nuclear power plants and other plants from the fuel cycle, were done.

In the case of a nuclear power plant (900 MWe), the whole concrete mass, ranges from 400 000 to 550 000 tonnes. Less than one percent will be concerned by exemption levels.

In the case of uranium milling plant, the mass of debris (bricks, plaster, soil) ranges from 2 000 to 16 000 tonnes.

2) Estimation of concrete quantities which are recycled or disposed off (B.1.3.).

The results of an inquiry on the different crushing stations in the suburbs of PARIS are :

- the capacity of the installations ranges from 225 000 to 360 000 tonnes
- 60 % of the installations have a device to limit dust emission (water spray or dust collection).

From these results we may estimate that about two years are needed for crushing the concrete arising from one nuclear power plant.

3) Dust measurements and analysis in different crushing stations (B.1.6).

The dust concentration ranges from :

- 4 to 81 $mg.m^{-3}$ outside nearby the crusher.
- 0.2 to 0.4 $mg.m^{-3}$ inside the cabin of engine.

The table I gives the values of the inhalable dust concentrations measured in different conditions.

4) Modelling of the different realistic exposure situations for critical groups (B.2.1).

In 1991 the study of the concrete crushing station was done.

The different exposures are :

For workers :

- Dust (inhalation, hand contamination, face contamination, ingestion, exposure to a deposit on soil, crusher maintenance).
- Concrete debris or aggregates (exposure to pile and during transport).
- Iron rebar (exposure to pile and transport, inhalation of iron dust during rebar cutting).

For public :

- Inhalation of dust around the concrete crushing station.
- Ingestion of vegetables from a garden located nearby the concrete crushing station.

Table II gives for the critical workers, the exposure parameters in the concrete crushing station.

Bold face = critical worker for the scenario.

5) 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 differents parameters were issued from observations of working practices (work duration,..) and standard of living (breathing rate, ..).

Table III gives the dose values for different critical pathways in a concrete crushing station.

The doses are calculated for the critical worker (maximum value of the table II)

The maximum doses calculated for $1 Bq.g^{-1}$ are :

- ^{60}Co	8.5μSv/y	for aggregates piles.
- ^{137}Cs	1.8μSv/y	for aggregates piles.
- ^{152}Eu	3.7μSv/y	for aggregates piles.
- ^{240}Pu	33.0μSv/y	for dust inhalation.

Table I: Inhalable dust concentrations.

	Inhalable dust concentrations			
station 1 impact double crushing	(surrounding)	0.03	$mg.m^{-3}$	AM
	without watering	55	$mg.m^{-3}$	AM
	" "	73	$mg.m^{-3}$	CH
	" "	81	$mg.m^{-3}$	AM
	watering	16	$mg.m^{-3}$	AM
	" "	56	$mg.m^{-3}$	CH
station 2	(surrounding)	0.3	$mg.m^{-3}$	CH
jaw	near the crusher	4	$mg.m^{-3}$	CH
	near the crusher	7	$mg.m^{-3}$	CH
after raining	inside the cabin			
	excavator	0.2	$mg.m^{-3}$	II
	concrete cuttings	0.4	$mg.m^{-3}$	II

AM = Andersen Mark II, CH = Collecting Head, II = Individual Impacter.

Table II: Critical workers and exposure parameters for concrete crushing.

station 1	dust Inhalation	piles external exposure	dust deposit on soil	maintenance
supervisor	**t = 20 %** **C_{inh}**	t = 90 % G = 0.4	t = 40 % G = 0.2	**t = 2.5 %**
labourer	t = 10 % C_{inh}	t = 60 % G = 0.3	**t = 25 %** **G = 1.0**	t = 1 %
crusher loader	t = 60 % C_{inh} / 100	**100 %** **G = 0.85**	non exposed	non exposed

t = percentage of the working duration (1860 h/a), G = exposure geometrical factor, $C_{inh} = 36\ mg.m^{-3}$

Table III: Doses (Sv/a) calculated for the crushing of concrete with a mass activity level of 1 Bq/g.

	dust Inhalation **supervisor**	piles external exposure **cusher loader**	dust deposit on soil **labourer**	maintenance **supervisor**
^{60}Co	9.6 E-09	8.5 E-06	1.2 E-06	3.3 E-09
^{137}Cs	2.0 E-09	1.8 E-06	3.0 E-07	7.3 E-10
^{152}Eu	1.4 E-08	3.7 E-06	5.7 E-07	3.4 E-09
^{240}Pu	3.3 E-05	3.7 E-10	2.7 E-06	6.6 E-06

6.6. THE CHARACTERISATION AND DETERMINATION OF RADIOACTIVE WASTE FROM DECOMMISSIONING

Contractors: AEA-Harwell
Contract No.: FI2D-0042
Work Period: January 1991 - December 1992
Coordinator: J W McMILLAN, AEA Technology, Harwell
Phone: 44/235/43 48 53 Fax: 44/235/43 45 22

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

Following discussions with the proposed collaborators, TUV Sudwest, Germany and other interested parties it became clear that the most urgent requirement in the area of radioactive waste characterisation was a method for the determination of ^{41}Ca and several other low energy beta emitting and difficult to measure nuclides. Consequently, the timetable, as shown in the Technical Annex to this contract, was rearranged to enable work to begin on the development of separation and measurement methods for these nuclides (item B3) ahead of item B2. Since these methods would also require developed counting methods (item B4) this work was also brought forward.

One of the influencing factors in making these changes was the acquisition of suitable material for analysis, namely a full length core cut from the Windscale AGR. Thus the work carried out during this reporting period has been concerned with the development of methods for the dissolution of concrete, procedures to separate chemically and radiochemically pure ^{41}Ca, ^{55}Fe, ^{63}Ni, ^{3}H and ^{14}C and methods for counting the separated products. The methods developed have been used successfully on concrete from the WAGR for which a good correlation has been found between these difficult to measure nuclides and the more easily measured gamma emitting nuclides ^{152}Eu, ^{60}Co, ^{133}Ba and ^{134}Cs.

Progress and Results

1. Rearrangement of the Timetable

Discussions with our collaborators, TUV Sudwest, Germany and other interested parties made clear the urgent need for methods for the separation and measurement of ^{41}Ca in reactor structural materials. It was also clear that this need extended to other low energy beta emitters, ^{3}H, ^{14}C, ^{63}Ni and electron capture nuclides ^{55}Fe and ^{59}Ni. Since suitable material on which to carry out this development was available, the decision was taken to rearrange the order of work shown in the Technical Annex of the contract. Counting methods were also required for this aspect of the programme so both sections B3 and B4 were brought forward ahead of section B2.

2. Acquisition of Samples (B1)

A full length (2.7m) core, diameter 10cm, cut from near the reactor core centre line of the bioshield of the Windscale Advanced Gas Cooled Reactor (WAGR) was acquired. A smaller sample, comprising a slice 2cm thick of a core cut from the Harwell reactor BEPO has also been acquired.

3. Measurement of ^{41}Ca, ^{55}Fe, ^{59}Ni, ^{63}Ni in Irradiated Concrete (B3)

Measurement of the nuclides ^{41}Ca, ^{55}Fe, ^{59}Ni and ^{63}Ni can only be achieved by counting the K-Xrays or the conversion or Auger electrons from suitably prepared sources of the purified elements containing these isotopes. Procedures were needed which would achieve total dissolution and enable separation of calcium, iron and nickel sufficiently pure for counting with no interference from other radionuclides.

3.1 Dissolution of Concrete (B3)

The total dissolution of the concrete sample was the necessary first step in the separation of the elements Ca, Fe and Ni. Preliminary experiments with inactive concrete showed that the complete dissolution could be achieved in two stages, namely digestion with hydrochloric acid (HCl) followed by fusion of the residue with sodium carbonate to convert the insoluble salts to soluble salts. The melt after washing with water to remove excess sodium carbonate was dissolved in HCl and the solution combined with that from the first stage.

3.2 Separation of Ca, Fe and Ni (B3.2, B3.4)

The procedure used to obtain highly purified calcium was developed using inactive concrete and radioactive tracers. It was based on a procedure used by Mabuchi et al /1/ to separate the radioactive components, including ^{41}Ca, from meteorite material. Steps were added to separate out the iron and nickel. The scheme is shown in block format in Figure 1. In order to achieve the high degree of purity needed for the counting, it was necessary to repeat some of the steps.

3.3 Recovery of ^{3}H and ^{14}C (B3.5)

The nuclides ^{3}H and ^{14}C cannot be recovered or measured using the scheme described above. For these nuclides a separate procedure has been developed. In this procedure, a sample of partially crushed (particle size about 1mm) sample is heated in a tube furnace at temperatures up to 950°C. Air which has been first dried and then rehumidifed with water, free of ^{3}H, is passed over the hot sample and carries away the released gases. These are fully oxidized before the water and carbon dioxide are trapped separately and measured by liquid scintillation counting.

An initial experimental programme using inactive concrete and material from the unirradiated end of the WAGR core showed the method to be satisfactory giving recoveries of tracer nuclides, added to the furnace tube, close to 100%.

3.4 Counting Methods (B4)

A number of experiments were carried out to enable decisions to be made about the most appropriate method for counting the purified sources of the individual nuclides. These experiments made use of tracer solutions of the nuclides ^{55}Fe, ^{63}Ni as well as of the γ-emitters which were known to be present in the irradiated concrete (^{152}Eu, ^{133}Ba, ^{60}Co, ^{134}Cs). Since no source of ^{41}Ca could be found, the tracer ^{45}Ca was used in some experiments and ^{55}Fe which has a decay scheme very similar to that of ^{41}Ca, was used in others as a substitute. It was concluded from these experiments that although X-ray counting, using lithium drifted silicon (Li(Si)) or high purity germanium (HpGe) might be satisfactory for ^{55}Fe, the much greater efficiency offered by liquid scintillation counting (LSC) far outweighed the higher resolution of X-ray counting. LSC, however, has relatively poor resolution and hence requires a greater degree of separation from possible interferences. This is particularly so in the case of ^{41}Ca for which the shape of the LSC spectrum was unknown. Counting of the separated products with an HpGe detector was used to confirm that complete removal of all other radionuclides was achieved before the LSC sources were prepared.

3.5 Measurements on WAGR Concrete

The procedures described in the preceding sections have been applied to samples taken from the WAGR core. Since the object of this work is to be able to infer the difficult-to-measure nuclides from the activities of those nuclides which can be measured easily by γ-spectrometry, these γ-emitters have also been measured.

The full length of the core was 2.7m. The fall-off in the activity level from the inner surface (closest to the reactor core) was such that the measurement of gamma emitters was not possible beyond 1.7m and those nuclides with the lowest specific activities were limited to much shorter distances into the core. However, the activity of ^{3}H was such that its distribution has been measured over the full length of the core. The results, expressed in terms of $Bq.g^{-1}$ concrete are given in Table 1 and shown graphically in Figure 2. A typical LSC spectrum of ^{41}Ca is shown in Figure 3. Although some variation is apparent in the first 100mm, there is, overall, a good correlation between the activities of the various nuclides measured.

In order for this conclusion to be valid it is also necessary to show that the chemical composition is uniform in the samples which were analysed. ICP/OES (inductively coupled plasma emission spectroscopy), is being used to measure the concentrations of the elements of interest but problems are being experienced with low concentration elements (Cobalt, Europium and Nickel) which are difficult to measure in the presence of other higher concentration elements such as iron. Although there are insufficient data at the time of reporting to enable any firm conclusion to be drawn, the variations observed so far in the chemical composition are not large, suggesting that the correlation is valid.

3.6 Measurements on WAGR Rebar

Some measurements have also been made of the radioactivity of the steel re-inforcing bar which is located at about 310mm in from the high activity end of the core. The results for ^{55}Fe, ^{60}Co and ^{63}Ni are included in Table 1. When the results are expressed in terms of becquerels per gram of target element (eg iron) the ^{55}Fe activity is consistent with the activity profile obtained for the concrete but the ^{60}Co activity is high relative to the concrete levels. As discussed above difficulties have been experienced in determining the cobalt content of the concrete and this is the most likely reason for this discrepancy.

3.7 Measurements on BEPO Concrete

One section of a core cut from the bioshield of the Harwell reactor BEPO was obtained. After non-destructive gamma measurements about half of this sample was ground up for analysis. This material differs from the WAGR material in that it has an high barium content. Samples were used for determinations of ^{3}H and ^{14}C but it soon became apparent that the procedures developed for WAGR material are unsuitable for BEPO material which is both more difficult to decompose and more retentive for ^{3}H, primarily because of the presence of barium and sulphate. Further development is needed to overcome these problems.

References
/1/ Mabuchi, H, Trobailem, J, Leger, C, Bibron, R and Biettmann, D. Geochimica et Cosmochimica Acta, 32, 949-963, (1968).

Table 1 Measured Activities in WAGR Concrete

Bqs per gram of material									
Dist * (mm)	10	55	110	290	530	1020	1690	2170	2680
Eu152	2151	2212	3533	927	44	2	0.66		
Eu154	257	231	262	67	3	0.1			
Eu155	9	6	7	2		0.005	0.06		
Ba133	17	13	23	4	0.2				
Cs134	40	29	36	6	0.3				
Co60	514	371	743	170	9	0.4	0.09		
Fe55	452	589	1134						
H3	28800		49400	9790	508	24	10	9	4
C14	20		28	6	0.5				
Ca41	87	130	88						
Steel @310mm									
Co60		3190							
Fe55		18000							
Ni63		157							

* Distance from core inner face

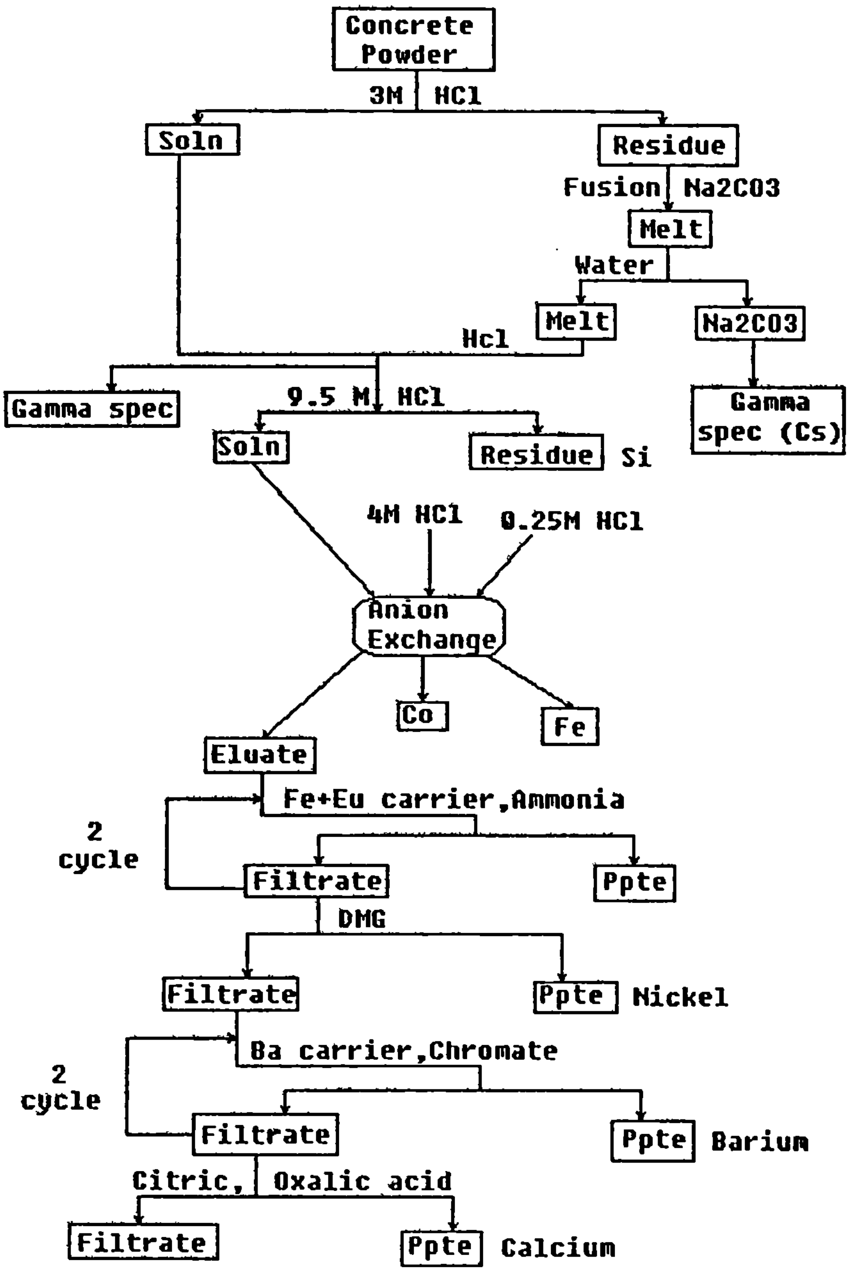


Figure 1. Block Diagram of Separation Scheme

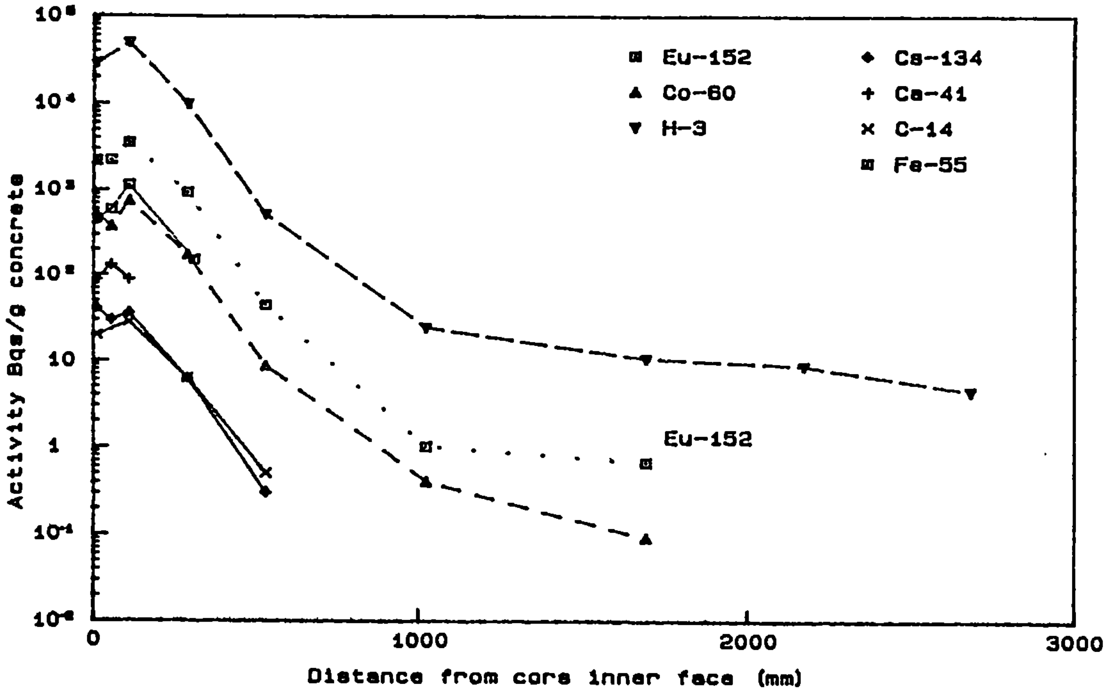


Figure 2. Activity Profiles in WAGR Concrete

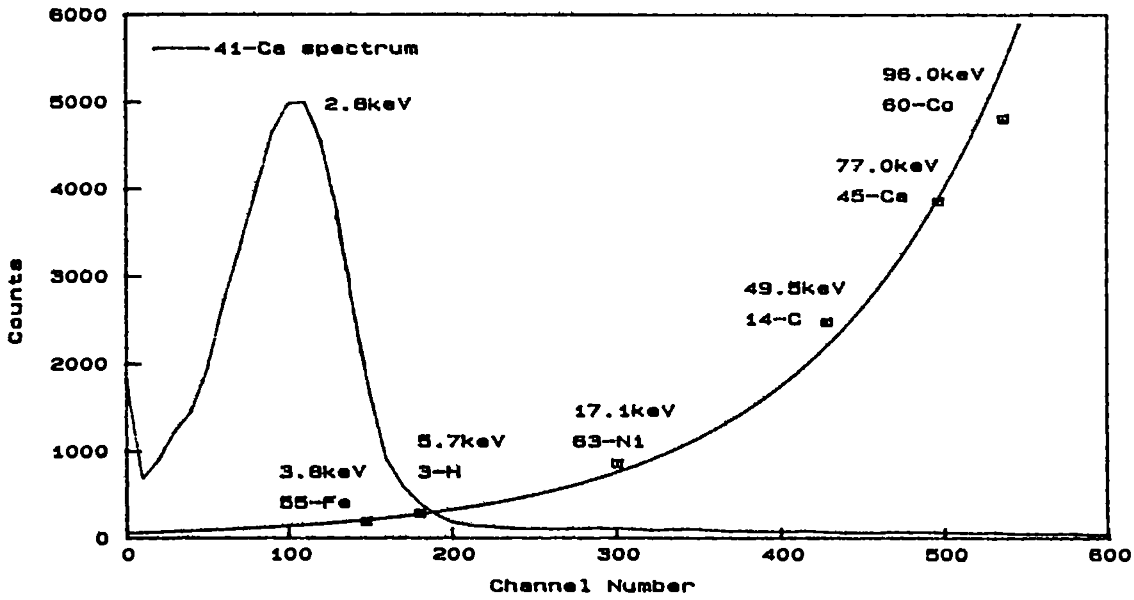


Figure 3. LSC Spectrum of 41-Ca

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 - September 1993
Coordinator:	J R COSTES, CEA/DCC/UDIN, Bagnols-sur-Cèze Phone: 33/66 79 13 Fax: 33/66 79 64 32

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

In 1991, the main objective was to bring a contribution to a better knowledge of the radioactivities in components of a graphite-gas reactor, both by sampling and by computing.

Progress and results

B.1. Dose rate measurements and analysis of core samples

Since their shut down dates (G2 : Feb 1980 - G3 Jun 1984), investigations were carried out to establish data (event partially) about the nature and the quantity of generated radio-elements.

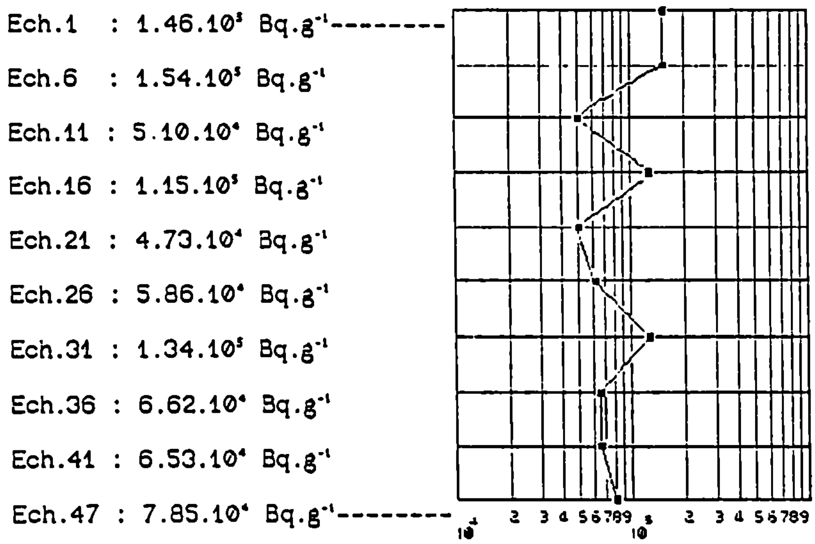
G2 has been, since recently, the object of several operations because the benefit of its earlier shut down resulted in a longer radioactive decay for isotopes present within the installation, thus presenting a lower activity (consequently a lower risk). With a view of normalization,. we decided to publish the different results occuring at different dates only on december 31st 1993.

The graphite stock composing the central part of the core is certainly, among the various elements, the one that is directly best known. During reactor operation, graphite was indeed, because of oxydation risk, under close monitoring by means of numerous samplings. Finally, the recent radial drillings right across the core improved the radiological knowledge of graphite, in the field of the nature of radioelements as well as their absolute quantity and their distribution.

Tritium

A few samples used to quantify graphite were taken from a vertical diameter of the vessel equidistant to the graphite rear and front faces. The results of the Tritium analyses show that this hydrogen isotope is largely preponderant (from 51,4 % to 96,3 %). The resulting values are shown below.

The hereafter results can never show any gradient of the specific activity of Tritium in Graphite



^{14}C

The available information concerning the specific activity of ^{14}C in graphite leads to a volumic distribution of this radio-element. Samples were taken according to the radial and symetrical axis.

Axial distribution

Samples were taken from 2 canals, one is representative of the central graphite, the other of the peripheral graphite. Both are of a pure quality. Results are the average of 2 different samplings carried out at the same level. They are depending on the distance between the bleeding points and the reactor front face.

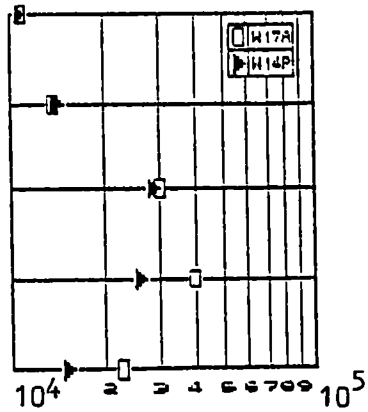
Elevation 2.5 m : W17A : 1.04.10^4 Bq.g^{-1}
W14P : 1.04.10^4 Bq.g^{-1}

Elevation 3.7 m : W17A : 1.37.10^4 Bq.g^{-1}
W14P : 1.48.10^4 Bq.g^{-1}

Elevation 5.5 m : W17A : 3.02.10^4 Bq.g^{-1}
W14P : 2.85.10^4 Bq.g^{-1}

Elevation 7.3 m : W17A : 3.89.10^4 Bq.g^{-1}
W14P : 2.52.10^4 Bq.g^{-1}

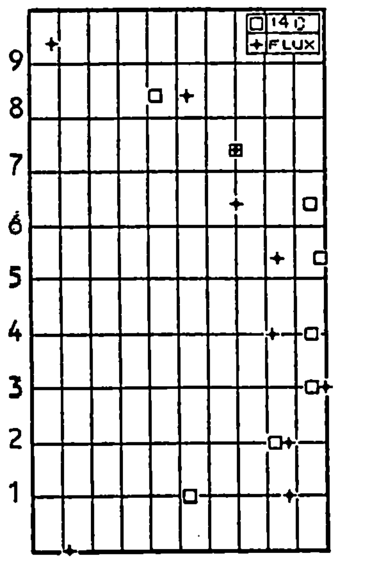
Elevation 8.5 m : W17A : 2.28.10^4 Bq.g^{-1}
W14P : 1.63.10^4 Bq.g^{-1}



Radial distribution

We take for the radial distribution of ^{14}C the same step as for its axial distribution. Samplings obtained by drilling through the full height of the stock gave results close to the flux evolution in a similar position (distribution of the neutronic flux in the precise axis of the drilling could not be obtained). Results are expressed in terms of relative values of the maximum obtained for the particular measure of flux or specific activity. Distance was measured from the down side of graphite up.

The axial distribution of the ^{14}C specific activity in graphite is directly proportional to the evolution of the neutronic flux.

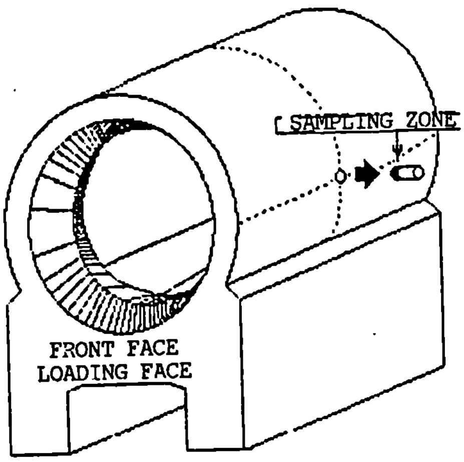


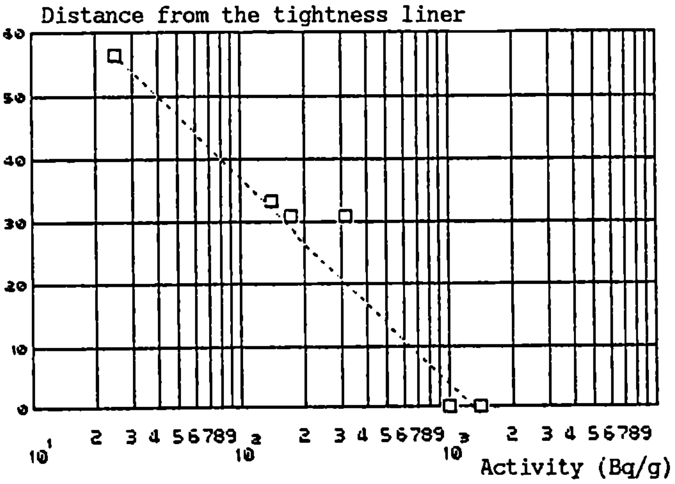
Concrete

The nature of the different isotopes and the value of the specific activity present in the upper part of the reactor vessel can be defined thanks to the horizontal drilling carried out in a region close to the maximum flux region.

^{60}Co	: 361.00 Bq/g^{-1}	^{137}Cs	: <3.80 Bq/g^{-1}
^{133}Ba	: 4.24 Bq/g^{-1}	^{152}Eu	: 916.00 Bq/g^{-1}
^{134}Cs	: 17.10 Bq/g^{-1}	^{154}Eu	: 76.00 Bq/g^{-1}

The examination of the results obtained during measurement of the other drilling fragments, representative of the evolution of the total specific activity of concrete depending on the position, shows an important gradient leading to a non-significant activity, of the major part of the reactor vessel.





B.2. Analysis of geometric, physical and radiological data

The physical or geometrical inventory of the 2 GCR core constituants was carried out, these 2 reactors G2 and G3, located at Marcoule, are being expected to be dismantled.
The features of the main constitutive elements were presented with superposed figures to facilitate the understanding of the ECC audience.
Description and diagrams related to :

- the concrete vessel
- the tightness liner
- the graphite stack
- the thermal shield
- supporting equipment and hooping
- the primary circuit
- various specific materials

B.3. Development of a computer programme to calculate gamma activity levels in the entire core.

Among the 60 elements present in the reactor as constituants or impurities, some can transform into significative isotopes from a dismantling point of view (half-life > 1 year and specific activity > 1 $Bq.g^{-1}$ at reactor shut down).

The "EVODIF" software was specially developed for dismantling calculation. Its first unit "EVOMAJ" will select elements and significant reactions. It calculated the maximal activity of each of the produced isotopes from its actual concentration in the most exposed mesh of the physical environment (exposure to thermal, epithermal and fast neutrons). It is then easy to keep only the evoluton chains (or parts of them) necessary to the production of significant isotopes.

EVOMAJ will automatically search its own library for :

1- the neutrons cross-sections (n, γ), (n, n') (n, 2n), (n, α), (n, fission), (thermal, epithermal and fast neutrons),

2- the radioactive decays (α, ß, ß+, CE) necessary for the evolution of the different isotopes of the selected element and its daughter products, in the actual neutronic field and according to the reactor operation chart.

This search will go as far as it is necessary in order to see every significant isotope. The most unexpected radio-isotopes are Tm-171 coming from Dy-164 after 10 generations of captures and decay, while only Disprosium traces are found in graphite (8 10^{-9} g/g) and Pt-193 coming from Tungsten (0,12 ppm) after at least 11 generations.

This library is the master piece of the EVOMAJ unit. 7000 values characterizing isotopes, cross-sections and radioactive decays are recorded.

Most of the constants are coming directly from recent publications ; 25 % were shortened especially when the isotope had a half-life < 1 hour. "Short circuits" were then created, trimming the programmes while calculating more quickly and more precisely. Quality assurance is given by an auxiliary software.

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

Summary of main issues

In Germany, the utilities are engaged to build a fund for the later decommissioning of nuclear power plants. Therefore, yearly calculations for decommissioning costs are performed with the computer code STILLKO. The contractor now uses this computer code to update the 1977 EC-study of decommissioning costs with regard to the actual state-of-the-art. All results of the realized decommissioning projects Niederaichbach and the pilot EC-decommissioning project Gundremmingen are considered.

There are two important tasks for the investigation: first, calculation is made of the cost of decommissioning of the German LWR types, based on the German decommissioning boundary conditions. Second, we prepare the results for a later comparison with the decommissioning costs of other EC-countries. This includes a specification and an interpretation of the cost factors.

Progress and results

1. Description of the boundary conditions for the decommissioning (B.1.)

The boundary conditions for the decommissioning in Germany are identified with a view to the cost influence. The most important influences are:

- the boundary conditions for the licensing procedure, a decisive criterion for the time of planning and licensing procedure and the extent of the licensing papers;
- the boundary conditions of radiation protection with limit values of dose rates and doses to the workers, which influence the working time;
- the boundary conditions for recycling and reusing materials from the controlled area. It is an important criterion for the amount of radioactive waste and for the cost of decommissioning. The working packages decontamination and waste management including treatment of secondary influence are directly related to the limit values of recycling and reusing;
- the boundary conditions of packaging and final storage of radioactive waste. For the further final storage facility Konrad, the licensing procedure is going on. The regulations for the packaging of radioactive waste and the storage containers are already defined.

2. Detailed technical description of decommissioning concepts (B.2.)

With reference to power plants Biblis A (PWR) and Brunsbüttel (BWR), a description of all decommissioning works is given, based on ten working packages. These working packages are:

- planning and project management
- licensing procedure
- decommissioning operation
- new installations and decommissioning equipments
- decontamination
- dismantling operations
- waste management
- radiation protection
- site recovery and restoration
- research and development.

The highlights of each working package are described, e.g. the technical measures to remove the reactor pressure vewwel and the biological shielding. The descriptions include the identified advanced techniques in some cases with alternatives. The treatment of primary radioactive waste will be shown, e.g. dismantled components and secondary waste, e.g. effluents.

3. Calculation of the decommissioning costs (B.3.)

Based on the boundary conditions and the identified technical concept for decommissioning of LWR's in Germany, the decommissioning costs are calculated with the computer code STILLKO. The results are given in Tables I and II.

The calculation method is based on the detailed structuring of the decommissioning measures in working packages, working areas, working groups and working steps. For each step, the manpower, equipment cost and the consumable cost, taking into account the mass (volume, surface or others) of components, systems and buildings. Cost calculations with STILLKO include the producing of timetables, mass tables and the calculation of the collective dose amount.

The actual results will be compared with the results of the 1977 study. Differences between the cost calculations are commented.

4. Comparability with costs of other EC decommissioning projects (B.4.)

The methodology of the cost calculation with STILLKO 2 includes a detailed description of the cost of each working step. This information is usable to specify the decommissioning costs to permit a comparison with other decommissioning projects or other decommissioning cost calculations. The comparison will be a further task and is not included in this study.

Table I : COST OF THE DECOMMISSIONING OF A BOILING WATER REACTOR

working package	dismantling after a safe enclosure period			dismantling
	Stage 1 [Mio DM]	Stage 2 [Mio DM]	Stage 3 [Mio DM]	Stage 3 [Mio DM]
1. planning, project management	10.749	8.901	67.546	76.906
2. licensing procedure	2.615	-	20.190	21.986
3. operation	9.550	19.099	59.016	70.076
4. new installations	5.394	-	31.002	20.922
5. decontamination	1.981	-	42.930	57.370
6. dismantling	-	-	52.734	60.832
7. waste management	4.019	-	114.570	166.699
8. radiation protection	2.507	1.755	60.135	77.010
9. site recovery	-	-	13.478	35.406
10. research and development	-	-	-	-
complete cost	528.171 Mio DM			587.207 Mio DM

Table II : COST OF THE DECOMMISSIONING OF A PRESSURIZED WATER REAKTOR

working package	dismantling after a safe enclosure period			dismantling
	Stage 1 [Mio DM]	Stage 2 [Mio DM]	Stage 3 [Mio DM]	Stage 3 [Mio DM]
1. planning, project management	9.100	7.663	71.984	78.117
2. licensing procedure	2.466	-	16.327	14.218
3. operation	5.922	19.551	53.008	51.088
4. new installations	4.946	-	31.163	24.996
5. decontamination	1.464	-	33.121	39.845
6. dismantling	-	-	39.155	47.131
7. waste management	3.184	-	90.147	115.137
8. radiation protection	1.544	1.566	64.340	69.155
9. site recovery	-	-	33.187	33.187
10. research and development	-	-	-	-
complete costs	483.544 Mio DM			472.874 Mio DM

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 32

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 RESULTS OBTAINED

Summary of main issues

1. Constitution of the apparatus

It is composed of four specific parts:

- Objective part:
  - . a double cone-shaped collimator: for localization of the different objects and radioactive areas
  - . a shutter: only open for the visible exposure
- Detection part:
  - . luminescent screen: transparent for visible rays; it converts gamma photons into light (scintillator)
- Signal intensification part:
  - . front intensificaton tube + optical fibers taper + very high sensitivity intensified CCD video camera: light amplification and transformation of the signal coming from the scintillator in a standard video signal (625 lines 50 Hz)
- Video signal treatment part:
  - . treatment system: acquisition of visible and gamma images and superimposition of both pictures with gamma densities coloration.

2. Work programme between October and December 1991

2.1. **Testing under real conditions**

Six pictures were taken of fuel element parts and radioactive dustbins in the cells of OSIRIS reactor in Saclay, see Figure.

The performances obtained were:

. visualization of small sources within 10 seconds with an equivalent dose rate of 4 mSv/h

. angular resolution was about 4°.

2.2. **Performance evaluations**

After real conditions tests, the whole detection chain was sent to an expert laboratory to check the performances of each component.

Improvements of different features are planned:

. angular resolution

. sensitivity

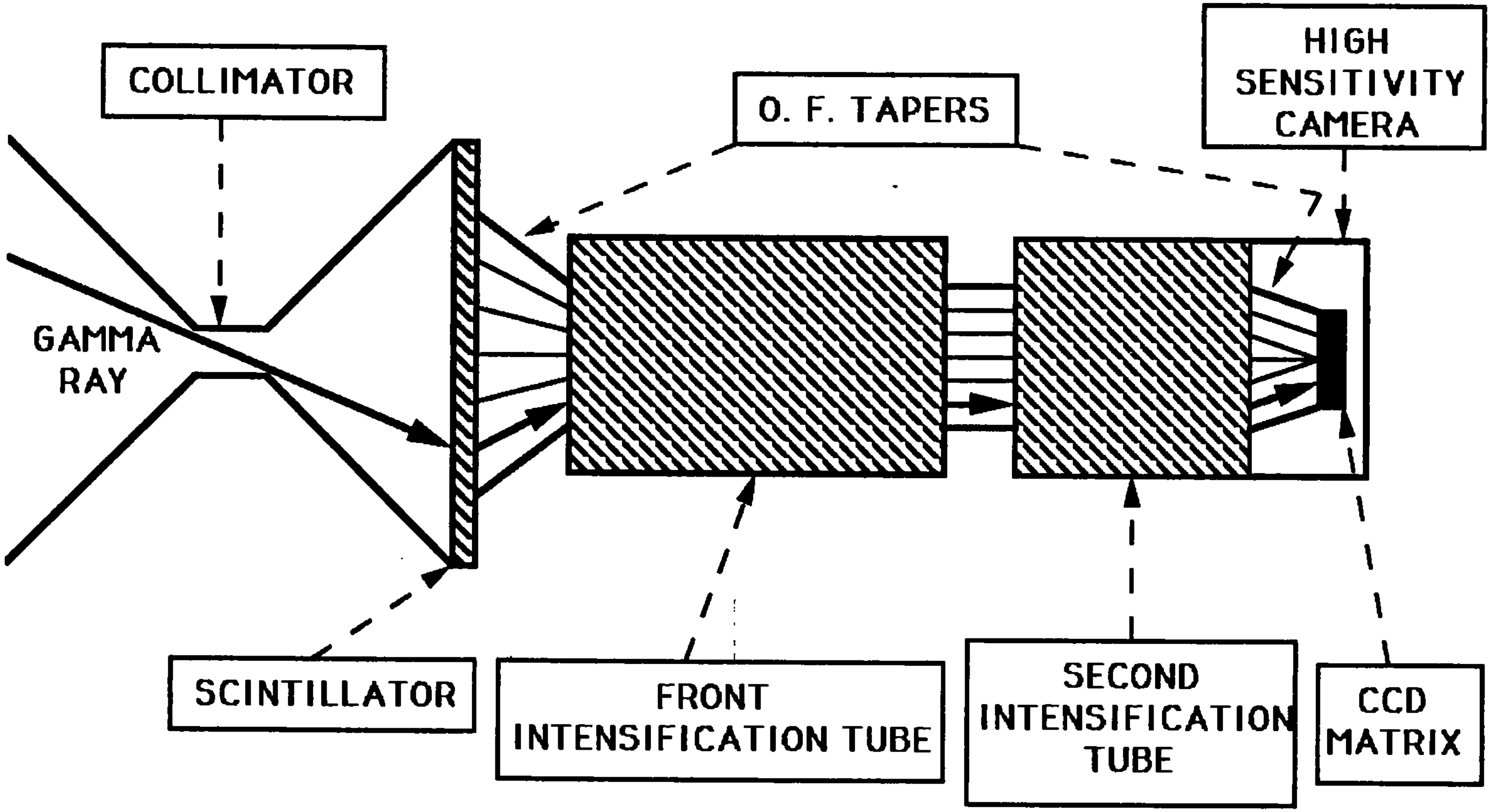
. electronic noise reduction.

These improvements will be obtained by modification of components assembly:

. command of a new front intensification tube

. replacement of high voltage tube alimentation

. removal of central optical fibers taper.



CHAIN CONSTITUTION DURING THE OSIRIS REAL TIME TESTS OF 1991

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. PRINCIPLES, REGULATIONS AND POLICIES FOR DECOMMISSIONING

A study is being performed by the Commission together with a group of experts, with the objective of assembling and discussing principles, regulations and policies for decommissioning, and of making recommendations for Community actions in this field.

The study is structured as follows:

1. Introduction
2. General principles and international recommendations and their relevance for decommissioning
3. European Community requirements and recommendations
4. Regulations, standards and policies in Member States
5. Present decommissioning practice
6. Conclusions and recommendations

Annex 1 - National regulations, recommendations and policies
Annex 2 - Selected decommissioning cases

The first meeting of the group of experts was held in September 1991. By the end of 1991, a preliminary draft of Chapters 1, 2, 3 and of the two annexes had been prepared.

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

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

At the end of 1991, the nine contracts were in an advanced stage of negotiation. The work is to be carried out in a period of 14 months.

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 condition of radioactivity configuration, size, accessibility and the state of the plant. The four large Pilot Dismantling Projects (WAGR/Windscale, BR-3/Mol, KRB-A/Gundemmingen 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 will be 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:
  Major dismantling actions on other nuclear reactors and other installations of the nuclear fuel cycle representative of important decommissioning tasks in the EC.
- secondment of scientific staff from Member States to the pilot dismantling projects:
  The operators of pilot projects receive staff from organization 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

At the end of 1991 the four pilot dismantling projects entered into the second contractual working phase. Eighteen contracts to perform alternative tests will become effective before 31 December 1991 and two further contracts are to start in January 1992. Ten of these projects selected after a second call for proposals limited to this research area, had been evaluated in May 1991.
Six contracts to establish the two data studies were also concluded.

In the following, reports concerning studies and projects beginning in the second half of 1991 are limited to the objectives, scope and programme items as no significant results can be reported for these new contracts.

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. 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.6., B.7. and B.8..

B.6. Dismantling of a WAGR heat exchanger, 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. Remote dismantling of the hot box, 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. Remote packaging of intermediate level waste, 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 OBTAINED

Summary of main issues

Refuelling standpipes have been cut in five locations. Cut four permitted the task of top dome removal to commence. Following removal of the top dome, the fifth and final cut (prior to the start of remote dismantling) reduced the standpipes to within 300 mm of the hot box.

The top dome of the reactor pressure vessel has been dismantled and sent as low-level waste. This task began in the previous period when all preparations were completed. During the present period, the top dome has been separated from the reactor pressure vessel, removed from the reactor vault, size-reduced and sent for disposal as low-level waste. This task was completed on March 28, 1991.

Associated with inactive trials of remote dismantling machine, construction of the test facility has been completed and installation of the remote dismantling machine is in progress.

An option study for the dismantling of a WAGR heat exchanger has been carried out from which three possible methods are being considered.

An initial preparatory phase in dismantling the hot box has been completed in cutting 130 stay tubes which served to provide additional stiffness to the top and bottom plates of the hot box.

Progress and results

1. Cutting and handling of the refuelling standpipes (B.2.)

The reactor was served by 247 refuelling standpipes which are being progressively cut and removed. Four vertical locations for cutting were identified; just below the top biological shield, directly above the top dome, just inside the top dome and directly above the hot box of the reactor. An industrial water-cooled plasma arc cutting torch was adapted to use gas cooling and optimised for minimum fume generation using a plasma gas of Argon with 5% hydrogen. Torches to this specification are inccorporated into a family of tools for internally cutting the standpipes. In all cases the tool is lowered from above the reactor into the standpipe.

The standpipes have an internal diameter of 143 mm and a wall thickness of 7 mm. Contamination on their inner surfaces results in a radiation dose of 20 μSv.h^{-1}. Background dose from the reactor pressure vessel varied as the operations progressed. General radiation levels in the work area clear of the reactor were very low at < 2 μSv.hr^{-1}, prior to top dome removal. Following removal, the general level rose to 5 μSv.hr^{-1}, whilst above the reactor on the temporary floor it was 78 μSv.hr^{-1}. However, planning estimates for the standpipe cutting had assumed a value of 150 μSv.hr^{-1} at the start of the operation, rising to 450 μSv.hr^{-1} as the standpipes were removed. In the event the value at the end of the operation was 225 μSv.hr^{-1}. The cutting coevered the period 9 May - 7 June 1991.

Dose uptake and team size for this operation is given below, showing that in addition to lower radiation fields there was a reduction caused by lower working times in the highest fields.

	ESTIMATED	ACTUAL
Team size	7	7
Total man-hours	1890	1407
Dose rate at start ($\mu Sv.hr^{-1}$)	150	78
Dose rate at finish ($\mu Sv.hr^{-1}$)	450	225
Total dose uptake (man.μSv)	29596	7297
Highest individual uptake (μSv)	17611	5895

TABLE 1: Summary of information for cut number 5

2. Dismantling the pressure vessel top dome (B.3.)

The top dome is the upper hemispherical end of the steel reactor pressure vessel, approximately 70 mm thick. Six independent pressure loop tubes and the 247 standpipes pass through the crown section which is thickened to 98 mm. The weight of this structure is approximately 45 tonnes. It is not activated but is contaminated to 370 Bq/cm^2 and radiation levels at its inside surface from other reactor components are estimated to be in the order of 4 $mSv.hr^{-1}$. The planned removal which incorporated a bagging out technique to contain the spread of contamination was carried out in seven distinct phases which required two cuts around the circumference of the dome to release the crown and plain dome sections.

Phase 1 Cut around the loops and general preparation
Phase 2 Cut around the crown section
Phase 3 Remove the crown section from the reactor vault (figure 1)
Phase 4 Size reduce the crown section in a temporary containment
Phase 5 Cut around the plain dome section
Phase 6 Remove the plain dome section from the reactor vault
Phase 7 Size reduce the plain section in a temporary containment.

The task, which commenced in the previous reporting period, was successfully carried out and completed on 28 March 1991. The actual man days expended on the work and the associated radiation uptake with each phase estimate is shown in the table below:

PHASE	ACTUAL MAN.DAYS	ESTIMATED DOSE (man.mSv)	ACTUAL DOSE (man.mSv)
1	87	3.74	2.22
2	56	1.00	1.12
3	116	3.12	1.74
4	118	3.83	3.22
5	64	2.75	1.53
6	15	3.61	2.84
7	95	3.05	2.83
TOTAL	551	21.10	18.99

TABLE 2: man.day and radiation uptake for top dome

3. Inactive trials of the remote dismantling machine (B.4.)

Inactive trials of the WAGR remote dismantling system are to be carried out in a test facility to be constructed adjacent to the reactor bioshield. The position of the test facility has been chosen such that the Remote Dismantling Machine can be lifted out whole and positioned over the reactor on completion of the trials. Construction of the test facility began in May 1991 with installation of the RDM commencing in December 1991. In addition to the test facility, the erection of the local control station located on the pile cap has been completed and the installation of the control equipment is currently in progress. During the initial testing phase, control of the main remote station for the completion of the trials at a later stage.

The control panels for the main remote dismantling machine are currently being modified to incorporate the latest design requirements and will be returned to site in mid-February 1992.

4. Dismantling of a WAGR Heat Exchanger (B.6.)

Heat exchanger A was one of four heat exchangers (identified as A, B, C, D) provided within WAGR for the production of steam from hot carbon dioxide gas. The heat exchanger has a vertical cylindrical shell of carbon/manganese steel 20.6 m high x 3.35 m diameter with a wall thickness varying between 36 mm to 64 mm. The total weight is approximately 180 tonnes. A concrete biological shield surrounds the heat exchanger. The internal surfaces of the heat exchanger are contaminated with mainly Cs-137 and Co-60. Total contamination is estimated to be between 0.5 and 1.5 x 10^{17} Bq. Recent surveys indicate that radiation decreases radially from the centre outwards and axially from the bottom upwards.

An option study for the dismantling tasks has been carried out examining three possible methods. The methods range through local decontamination and hands-on dismantling, limited decontamination and modular removal and limited decontamination together with localised shielding to allow semi-remote size reduction. These options are currently the subject of detailed assessment.

5. Remote dismantling of the hot box (B.7.)

Remote dismantling of the hot box cannot be undertaken until the Remote Dismantling Machine is installed above the reactor in 1993. The development programme essential to ensure an effective system of work is currently in progress. Following the completion of standpipe cutting the opportunity was taken to do practical tests on the hot box with a small bore pipe plasma cutting tool. This work has effectively severed the 130 stay tubes which were used to provide additional stiffness between the top and bottom plates of the hot box. Of the 130 stay tubes, 100 were cut successfully both top and bottom. The remaining 30 were cut at the top only. The 30 single cuts were made to provide the top plate with some residual support during the remote removal stage.

The test began on 31 July 1991 and was completed on 21 August 1991.

FIGURE 1 - Final removal of the crown section

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:	W STANG, 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 under 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 - June 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.Chemical decontamination of the Primary Loop (B.1.)

Using the CORD process developed by Siemens, the BR3 primary loop and associated auxiliary circuits have been decontaminated between April 9 and April 18, 1991; during this period of nine days, the CORD process has been applied in three successive cycles, using the BR3 primary pumps to circulate the chemicals at low pressure (20kg/cm²) and to reach the required primary water temperatures (80-100°C).

A mean Decontamination Factor of the order of 10 has been obtained in the primary loop, an activity of 2 E12 Bq (55Ci), mainly Co-60, being removed from the circuits and trapped on ion exchange resins.

The operation of the primary loop and auxiliary circuits after a shut-down period of almost 4 years (plant shut-down on 30.06.87) has implied a large effort of maintenance and preparation which resulted in an uninterrupted decontamination process without any technical incident.

2. Segmenting of the Reactor Internals (B.2.)

The main objectives of the 1991 activities were first to test in cold conditions the segmenting equipment designed and constructed in 1990 and afterwards to demonstrate this equipment in a radioactive environment, dismantling a first activated reactor internal, the reactor vessel thermal shield.

The thermal shield is the thickest (3 inches or 76.2 mm) of the stainless steel reactor internals but also the most simplistic one as far as its geometry is concerned : a cylindrical annulus (∅ : 1.4 m, H : 2.4 m) with a weight of 5.5 tons.

As illustrated on Figure 1, the thermal shield has been segmented first in-situ, i.e. inside the reactor vessel, and afterwards rings separated from the thermal shield have been transferred and cut into angular segments inside a flooded cutting chamber installed in the reactor refueling pool.

Three cutting techniques have been used : mechanical sawing, electro-discharge machining and plasma arc torch cutting; their performances and harmfulness, more particularly the quantities and properties of the secondary waste resulting from the cutting operation, have been compared.

The cutting campaign of the thermal shield extended from August, 30 (first vertical cut by Electro Discharge Machining) to December, 18, 1991 (arc torch last vertical cut into the last ring); before this campaign, an intensive working period, mainly in July and August, was devoted to the unloading of all the reactor internals at the exception of the thermal shield; these internals were stored under water into the refuelling pool and the cutting equipment was transferred to and installed inside the plant containment.

Progress and results

1. Chemical decontamination of the primary loop (B.1.)

The decontamination operation can be divided into 3 main phases :

Phase 1 : a preparatory phase which comprises mainly the closure of the reactor vessel, the repair and maintenance of components (pumps and valves) in the primary loop and in the purification system and some modifications to the purification system, enlarging its ion exchange capacity and providing connection to the chemicals preparation kit.

Phase 2 : the decontamination operation itself which was performed between April, 9 and April, 18, 1991.

Phase 3 : the post-decontamination operations, essentially the evacuation of liquid and solid wastes generated by the decontamination operation and the opening of the reactor vessel. Except the evacuation of the active resins which is not yet completed, all the other tasks were performed.

The collective doses related to the execution of these phases are summarized hereafter :

Phase 1 - Preparatory phase	:	135.5 man-mSv
Phase 2 - Decontamination operation	:	6.4 man-mSv
Phase 3 - Post-decontamination operations	:	14.9 man-mSv
		156.8 man-mSv

The operations required for the closure of the reactor vessel by fastening the reactor cover on the reactor vessel flange are responsible for 69% of the radiation dose associated to the Preparatory phase; in a small old-design plant like BR3, these operations are not done remotely.

Nevertheless, this chemical decontamination remains very cost-effective in man.Sv exposure reduction when the dismantling of the primary loop is considered.

The CORD process is a low chemical concentration and a low temperature process resulting in absence of corrosion of the base material; the contamination present in the erosion-corrosion products deposited on the inner surfaces of the circuit components is displaced and concentrated on the ion exchange columns of the purification system. Three decontamination cycles were performed in 9 operating days. Each cycle comprises an oxidation step with permanganic acid, a reduction-decontamination step with oxalic acid during which the released activity is trapped on the ion-exchange beds and finally a cleaning step. For the two first cycles, permanganic acid was used for the cleaning step whereas for the last cycle, hydrogen peroxide was used to remove the excess organic acid.

Gamma radiation dose rate measurements have been done, using thermoluminescent detectors (TLD), before and after the decontamination, in one hundred different locations into the plant. As far as the primary loop is concerned, not considering the measurements done in the vicinity of the reactor vessel upper flange where the gamma dose rate is mostly due to the radiations emitted by the activated reactor internals, a mean DF (Decontamination Factor, ratio of the gamma dose rate before to the dose rate after the decon) close to 10 has been obtained with a broad spreading of individual values ranging from 1.5 (bottom of pressurizer) to 53 (steam generator body), as far as measurements done in contact with the component are concerned.

Table I gives values for Average Decontamination Factor in contact with the equipment and at some distance of them, designated as ambient values.

Values reported are from TLD (thermoluminescent detectors) measurements and also from measurements done with a Teledetector, a portable device for dose rate direct measurements; generally, there is a quite good agreement between the various sets of values : a factor 2 difference is not significative for such measurements. Along the purification system, where the operating temperature was kept lower (40-80°C) than in the primary loop (80-100°C), a mean DF of 6.2 was obtained with a spreading of individual values ranging from 0.7 (activity deposition in a low contaminated C steel heat exchanger) to 9.2 for the Regenerative Heat Exchanger.

At the exception of locations close to the reactor vessel cover at the bottom of the refueling pool and of some localized hot spots, radiation dose rates higher than 0.3 mSv/h are no longer found on working sites inside the plant container; the mean ambient dose rate in the vicinity of the primary circuit is about 0.08 mSv/h and about 0.06 mSv/h for the purification loop.

During the 3 CORD cycles (Table II) a total gamma activity of 2.05 TBq (55.35 Ci) was released, about 99% of which being Co-60. The other isotopes measured in significant quantities were Cs-137, Sb-125 and Mn-54. This large Co-60 percentage is due to the cooling time, not far from 4 years between the plant shut-down and the decontamination operation, short life emitters being practically disappeared. An estimation was also done of the alpha activity release; a total alpha release of about 2.33 GBq was measured corresponding to 185 mg plutonium.

A total amount of 33 kg of corrosion products as metal (iron, chromium, nickel) oxides was removed from the system inner surfaces.

The gamma- and alpha-activity and the chemical species in cationic (Ni, Mn) or in anionic form (Fe, Cr) were removed during the decontamination and the cleaning steps by using ion exchange beds filled with anionic or cationic resins. The total volume (1370 l) of resins saturated during the decon process is distributed as follows :

- Anionic resins	632 l
- Cationic resins	524 l
- Mixed bed (anionic catalytic + cationic)	214 l.

Most of the activity (about 97.5%) is concentrated in the 524 l cationic resins.

The full reactor coolant system decontamination with the CORD process demonstrated that :

- It is possible to perform this task with only minor modifications brought to the circuits.
- The 3 cycle application of the CORD process led to important dose rate reductions.
- The process is safe, reliable and easy to keep under continuous control.
- The released activity is concentrated in a low volume of waste.

2. Segmentation of the reactor internals (B.2.)

For the three cutting techniques, cold tests on a 1:1 scaled mock-up of the thermal shield (Figure 2) were carried out with as main objectives to confirm the cutting parameters defined previously on SS thick plates and to train the personnel in charge of the cutting operations.

The main cutting parameters used during the cold tests and the thermal shield cutting campaigns are summarized in Tables III to V.

The total period for segmenting the thermal shield amounted to 3,5 months. The performance time for each cutting method is reported in Table VI.

The remaining time was devoted to the preparation of the cutting tools and equipment and to the removal of some equipment to be evacuated between the different cutting campaigns. All the cutting operations were achieved successfully within the time period mentioned, the thermal shield being now transformed into 40 segments of about 500 x 500 x 76.2 mm (Figure 1).

It is interesting to compare the amount of waste generated by each process to the active metal released from the thermal shield by the cutting process. For the EDM process and the mechanical cutting, the produced particles or chips were sucked off close to the production site (i.e. the EDM electrode or the milling cutter) and were kept on filters before return of the sucked water into the refueling pool. For the plasma arc torch, carried out inside a flooded chamber (Figure 3), the water of the chamber was circulated on filters of different mesh sizes and on ion-exchange resin columns. For the mechanical cut, the volume ratio between waste produced and released metal amounts to 21 while for the plasma arc torch this ratio amounts to 36 and for the EDM to 58.

Some unexpected observations done during this cutting campaign are worthwile to be noted :

- the very low contamination remaining on the cutting tools (electrodes, saw blades) after flushing them under water by a high pressure (150 bar) water jet, allowing to carry out the tool exchange for a very limited radiation dose, contrary to our expectations
- the quite low contamination of the air and water circulated in the flooded chamber for arc torch cutting, allowing to avoid any exchange of filters during the whole cutting campaign; the water and air contamination levels were noticeably lower than foreseen on the basis of the literature survey and even of the measurements done during the cold tests.

The lessons learned from the thermal shield cutting do not differ significantly from those which resulted from the cold tests :

- the mechanical sawing is a very efficient and clean method producing secondary wastes in the form of several mm metal chips which are easily collected, not contaminating the equipment and surroundings
- the secondary wastes produced by EDM are already a little more difficult to control, but the contamination of the water and of the equipment can be kept to a low level, provided an efficient suction and cleaning system is available. EDM with 6 mm thick electrodes is a very slow method in comparison with mechanical sawing and should be limited to fine surgery and not to bulky cutting.
- plasma torch is a very fast and efficient cutting method; plasma arc torch cutting in a closed chamber allows to avoid the spreading of the contamination; an efficient cleaning system allows to recover rapidly a good water quality; aerosols and gas produced by this cutting process are, in these conditions, easily kept under control.

In all cases, the tool consumption was very moderate and replacement of worn tools or cutting head maintenance became only a radiation dose concern in the case of arc torch cutting when ejected molten metal particles were sticking on the equipment.

The 40 segments of the thermal shield will be transferred from the refuelling pool to the deactivation pool situated outside the reactor building, where they will be stored, waiting for evacuation; a stainless steel lead cast container (≈ 12 tons) has been constructed as the main equipment for transfer of the segmented reactor internals.

Loaded with six angular segments (Figure 4) originating from the most activated region of the thermal shield, the container is characterized by contact radiation dose rates of the order of a few mr/h (max : 5 mr/h or 50 μSv/h); the transport of this container between the two pools will therefore not contribute significantly to the operator radiation dose : this has been indeed its basis for design.

Table I : Average decontamination factors

	Measurement by Teledetector Contact	Measurement by Teledetector Ambient	Measurement by TLD Contact	Measurement by TLD Ambient
Primary piping + Reactor Coolant Pump	9.4	5.6	9.1	3.6
Steam Generator	62	33.3	31	33
Pressurizer	5.2	2.9	3.7	2
Purification System	6.8	3.8	5.5	2.2

Table II : Activity released during each decon cycle

Cycle nr.	Gamma-emitters TBq	Gamma-emitters Ci	Alpha-emitters MBq	Alpha-emitters mCi	Alpha-emitters mgPu
Cycle 1	1.11	30	865	23	68.5
Cycle 2	0.67	18.2	531	14.3	42.1
Cycle 3	0.27	7.15	935	25	74.0
Total	2.05	55.35	2331	62.3	184.6

Table III : Mechanical cutting of thermal shield

Cutting into			Cold Mock-up		Activated thermal shield	
Milling cutter diameter	d	mm	250	320	250	320
Milling cutter thickness	s	mm	4	4	4	4
Max. cutting depth per step	C	mm	20	20	20	20
Feed of cutting tool	V/-V	mm/min	≤ 30	≤ 30	20 ± 3	18 ± 2
Revolving speed	Ω	m/min	8 - 10	8 - 10	≅ 9	≅ 12
Max. total cutting depth	T	mm	56	80	56	80

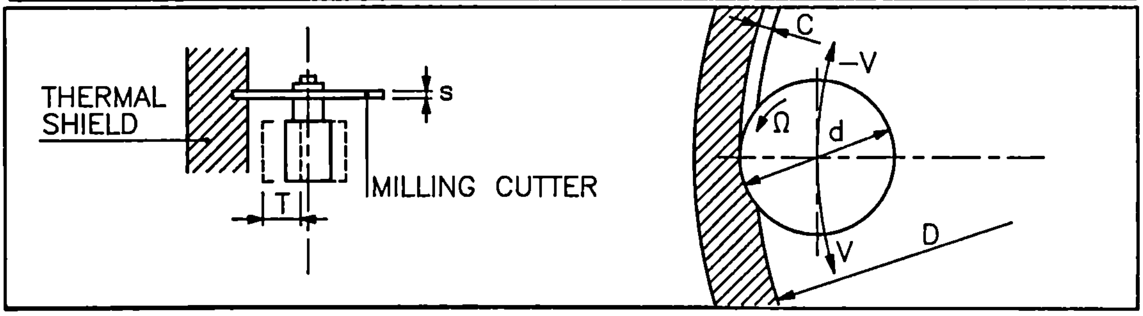


Table IV : EDM for segmentation of thermal shield

	ELECTRO DISCHARGE MACHINING (EDM) Vertical	Horizontal
L = Electrode length	380 mm	380 mm
T = Electrode thickness	6 mm	6 mm
Kerf width	≅ 7 mm	≅ 7 mm
Feed	2 - 8 mm/h	2 - 8 mm/h
EDM current	24 - 44 A	24 - 44 A

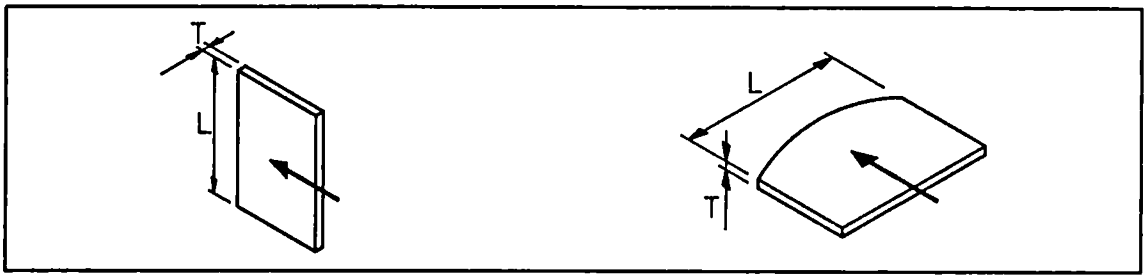


Table V : Plasma arc torch parameters for the vertical cutting of the thermal shield rings

Plasma current	: 800 A
Cutting speed	: 0.3 m/min
Starting speed	: 0.15 m/min
Cutting direction	: downwards
Torch stand off	: 7 mm
Torch tilt	: ≅ 5° (from horizontal, downwards)
Kerf width	: ≅ 11 mm
Primary plasma gas	: N_2
Secondary fluid	: H_2O

Table VI : Performance time for each cutting method

Cutting process	Time for performance (working hours)	Cut length (m)
EDM vertical	184	2.92
Milling cutter	192	12.5
EDM horizontal	240	4.15
Plasma arc torch vertical	136	15.4

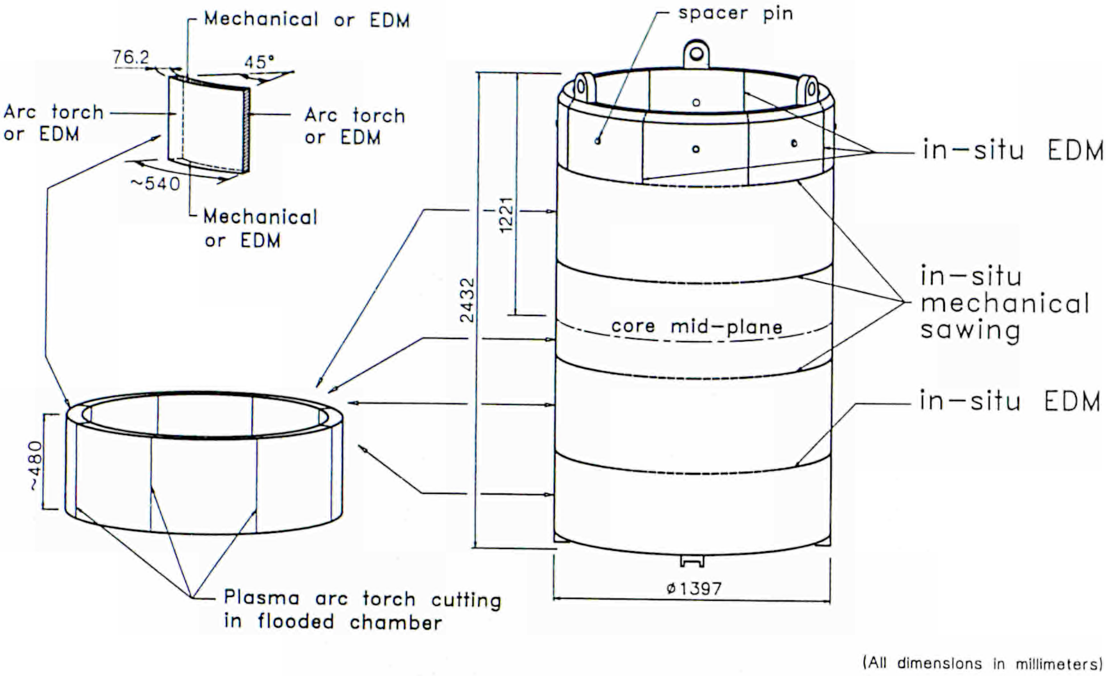


Figure 1
BR3 Thermal Shield Segmentation

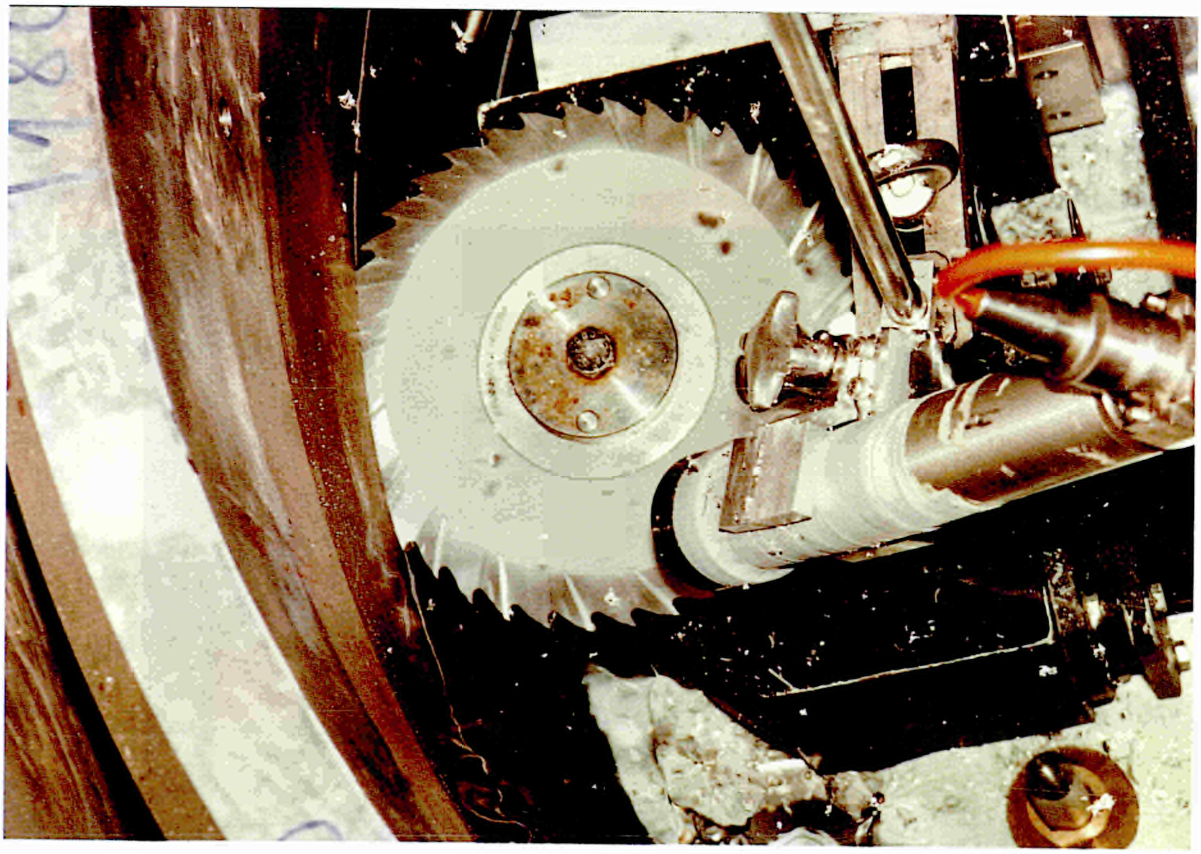


Figure 2
Milling cutter at work through the wall of the thermal shield mock-up

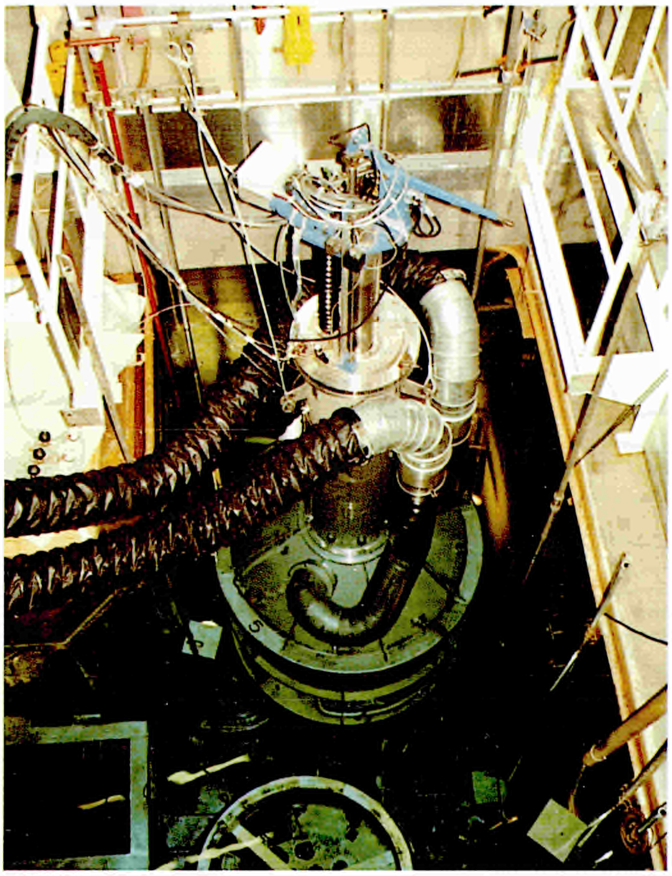


Figure 3
The plasma arc torch flooded chamber inside the refuelling pool

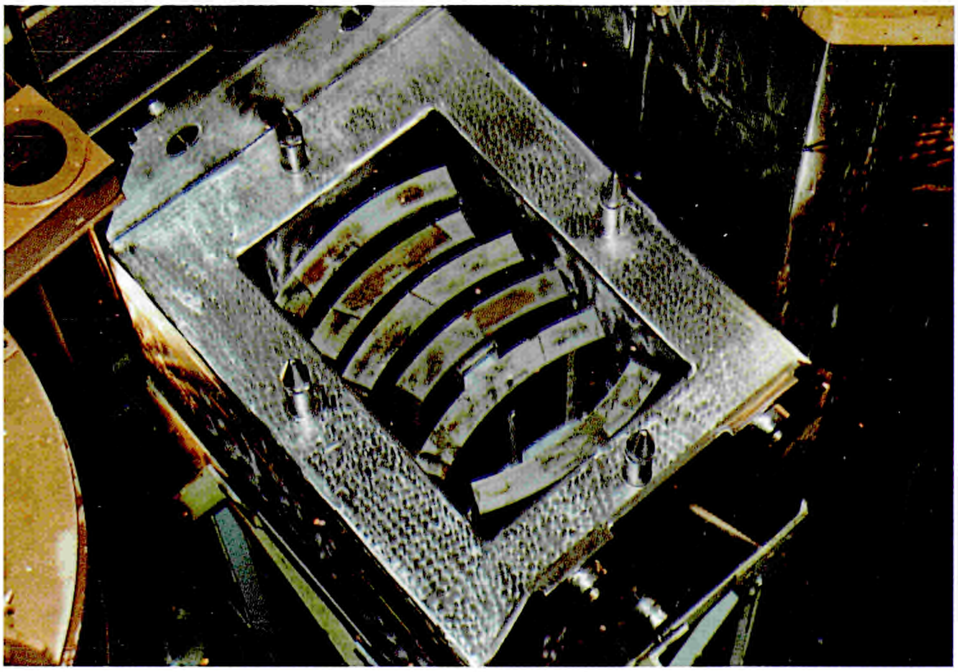


Figure 4
Six segments of the Thermal Shield loaded inside the transfer container

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 1992
Coordinator: F CORNU, 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. Dismantling of the plant has started and is planned to be completed by 1992.

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 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 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 almost been totally completed (90%) using the ATENA Machine (B.1.), see Figure 1 presenting concerned cells in the AT-1 Building.

To enter Cell 903 (Dissolution Cell), it has been necessary to make an access through the concrete wall separating Cell 903 from Cell 904. This operation was remotely carried out with the ATENA Machine equipped with a diamond-tipped saw (B.5.).

The cell used for the conditioning of waste was put into service and permitted to remove almost half of the waste bulk from Cell 904 (1st extraction cycle) (B.2.).

After 2 years of operation, the experience gained with the remote equipment, used to dismantle the high activity cells of AT-1 and to remove waste, was investigated (B.4.).

1. Remote operating dismantling of equipment out of the strongly contaminated cells (B.1.)
   Cell 905 (2nd and 3rd extraction cycles), completely dismantled at the end of 1990, except for the recovery pan, was cleaned and a layer of vinyl put on the floor. No strong decontamination was carried out because this cell is part of the waste removal process.
   No further dismantling operation was performed in Cell 904 (1st extraction cycle), dismantling work being almost over. Waste removal has started.
   After cutting the separation wall between Cells 903 and 904 (see hereafter B.5.), dismantling of the equipment of Cell 903 (dissolver) has been performed up to 95%.

2. Measurement of radioactivity and conditioning of waste (B.2.).
   The "workshop cell" (cell used for conditioning high-activity waste) described in the previous Progress Report, was put into operation in April 1991.
   The teleoperation tools include 2 mechanical M8 remote manipulators and the RD500 robot. This prototype needed to be modified because of previous cabling failures.
   It was then replaced in September with one of the remote electrical manipulators MA23 mounted on the ATENA Machine. From then on, the waste removal from Cell 904 was expected to progress normally.
   Waste conditioning progresses according to the following steps:
   - filling of a bin with waste with the ATENA Machine mounted with the MA23 arm;
   - removal of the bin from Cell 904 to workshop cell with the 5kN hoist;
   - cutting of pipes in the workshop cell with shears fixed on a work bench. Pipes are handled with the MA23 or the M8 remote manipulators;
   - radiological mapping in the workshop to sort out waste before drumming;
   - drums (either 120 l or half-drums with or without shielding)
   - drumming with the MA23 and the M8 manipulators;
   - removal and checking of drums:
     . beta-gamma activity
     . alpha counting
     . weight
     . assessment of external non-contamination
   - filling the 3 m^3 standard containers and check.

   At the end of 1991, 33 drums have been removed. This represents:
   - 1,800 kg of conditioned waste (apart from drums and shieldings)
   - 81 GBq of total activity divided into 48 GBq of beta-gamma radioelements and 33 GBq of alpha radioelements.

   To-date, it can be considered that 40% of the waste have been removed from Cell 904.

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 semester of 1991.
   This programme is completely finished.

4. Generation of specific data (B.4.)
   The efficiency balance of the teleoperation material, ATENA Machine + MA23 remote manipulator, was drawn up for the years 1990 and 1991.
   In the course of 1990, ATENA was mainly used (in Cells 905 and 904) for cutting with the MA23 manipulator associated to an abrasive disc-saw. The shears were hardly used.
   In the course of 1991, the ATENA Machine was mainly used to:
   - cut the separation wall between Cells 903 and 904 (B.5.) with a tipped diamond-saw;
   - dismantle Cells 903 and 904 with an abrasive disc-saw,
   - remove the waste from Cell 904.

   One can record the pretty high reliability of the ATENA carrier, an average value of 0.8, the MA23 telemanipulators reliability is about 0.4 and 0.5 for each arm.
   This rate is average value but nevertheless satisfactory considering about the fragile "tapes" dragging along system; conditions are presently particularly harsh and tapes broke regularly and are to be replaced.
   Using a 2nd MA23 arm on the ATENA carrier allows to siginificantly increase the reliability of the system ATENA + MA23 since maintenance is then carried out in concealed time.

5. Remote dismantling of a reinforced concrete wall (B.5.)
   A concrete wall separates Cell 904 from Cell 903. To enable the dismantling of Cell 903 equipment with the polyarticulated arm of the ATENA Machine, it has been necessary to cut this wall.
   This separation wall was: 5.5 m high, 2 m long and 0.2 m thick. It was reinforced in two places.
   To introduce the ATENA Machine for the dismantling of Cell 903, it has been necessary to make an opening of 4.5 m high and 1.2 long in the wall.
   Besides, irradiation was such in Cell 904 and especially in Cell 903, that no direct work was possible because the demolition of the wall lowered the biological protection level of the operator.
   The cutting device consisted in a diamond-tipped disc-saw directly mounted on the polyarticulated arm of the ATENA Machine. The saw was light: about 10 kg. The rotatory motion provided by lateral rollers allows the use of small diameter discs (35 cm) and to cut 20 cm in thickness.
   Liquid nitrogen cooled the disc. Another cooling system would have produced liquid effluents in a cell with concrete not sealed and therefore easily liable to be contaminated.
   Qualification tests were performed on a representative mock-up on the wall built in the maintenance cell of the ATENA Machine. They resulted in the confirmation that the process feasibility was perfect. The fixing device of the tool on the ATENA Machine was adjusted and the control system permitting a delicate guiding of the tool during cutting was modified.
   The tests on the cooling system proved its efficiency. At first, pipes and elements protruding from the wall and disturbing the cutting were removed. Inserts were placed in the wall, with the MA23. The cutting itself started with 2 vertical breaches and the cutting of 4 concrete blocks. Subsequently, 2 other blocks were cut (after placing inserts) in order to have better access into Cell 903.

6. Semi-automatic decontamination and contamination measurements of concrete walls and floors (B.6.)
A first project study for a scraper and an associated measurement system was carried out during the 2nd semester of 1991.

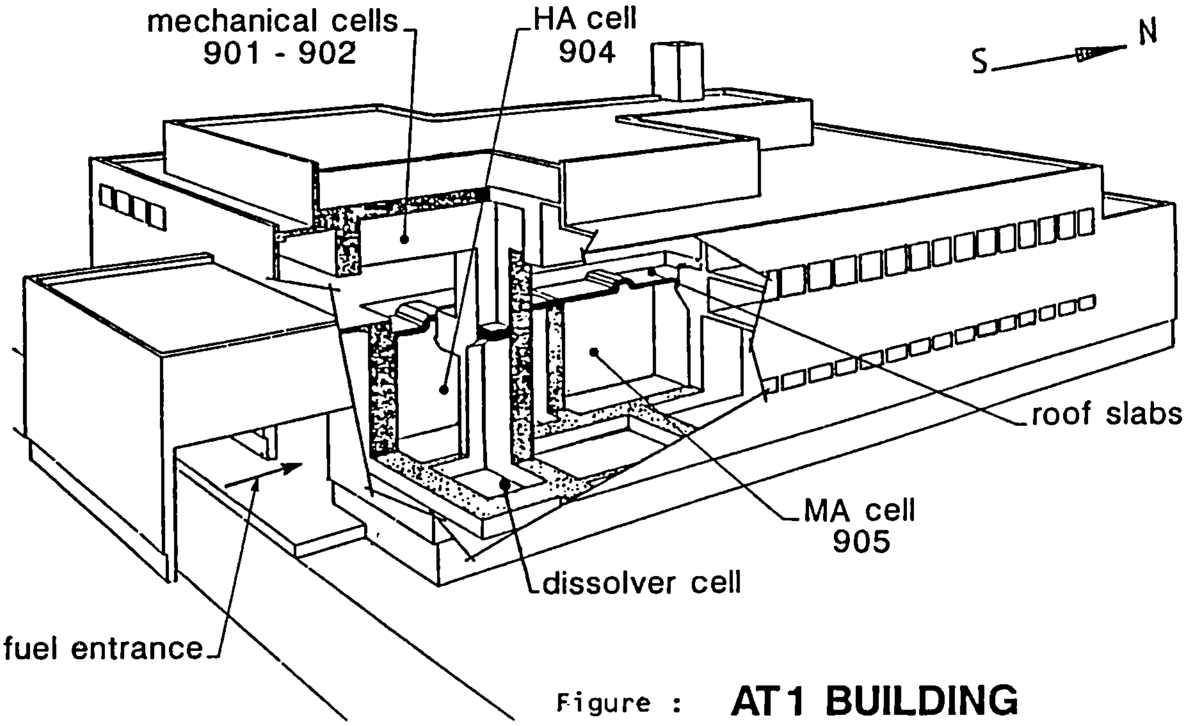


Figure : **AT1 BUILDING**

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: W STANG, 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/cm², 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 1991 the dismantling of KRB A was continued. The following activites have been carried out in the framework of the CEC pilot project:

- preparation of the dismantling of a secondary steam generator,
- dismantling of a shutdown cooler,
- dismantling of a primary cleanup cooler,
- pre-segmenting of the head of the reactor pressure vessel.

The main targets in this year were the application of the ice-sawing technique and and the preparation of under-water cutting of the steam dryer.

Progress and results

1. Dismantling of a secondary steam generator (B.1.1.)

Besides the reactor pressure vessel (RPV), the secondary steam generators (SSG) are the largest components in the reactor building of KRB A. Each of the three existing SSG is installed together with a primary recircultation pump in a separate loop room. Because of the large dimensions of the components (2.2 m diameter, 9.2 m height) it is not possible to remove the whole component without segmenting in-situ. The ice-sawing technique was developed with special regard for this cutting task. It is intended to freeze the whole component and to cut off several slices with a maximum height of 0.8 m for a subsequent transport and treatment in the turbine house. A detailed study has already been performed in order to assess the feasibility of a dismantling inside the loop rooms. The following topics have been studied:

- modification and transport of the existing mobile bandsaw into the loop room,
- technique and equipment for settling down the total SSG to the soil,
- fixation of the bandsaw at a vertical guide rail welded at a SSG,
- determination of position of the guide rail with regard to the restricted space for the movement of the saw during cutting,
- installation of a crane and other equipment necessary for lifting and transporting of cut-off slices,
- work cycle for the total dismantling in-situ.

Due to the lowest dose rate in the installation place the SSG-2 was selected to be dismantled first. Presently, all disturbing components for ice-sawing of the SSG in loop room-2 have been removed. The primary recirculation pump with the electric motor and about 20 tons of pipes and valves have already been taken off.
Pipes and valves have been dismantled by applying mainly mechanical cutting techniques such as mobile saws, hydraulic shears and tube cutting-off machines.

About 7 tons of this material was decontaminated by electropolishing in phosphoric acid for unrestricted release. The remainder was packed into drums for transportation to external controlled melting. Manpower and job doses for this work is given in Table I. Figure 1 shows the SSG-2 in the empty loop room waiting for ice-sawing. Several service companies have been asked to offer a bid for manufacturing and installation of the necessary equipment and for the execution of the dismantling of the SSG-2. The work will start mid of 1992.

2. Dismantling of a shut-down cooler (B.1.4., B.1.6)

The first large scale application of the ice-sawing technique was executed with a high contaminated shut-down cooler. The average contact dose rate was 2 mSv/h. Technical data for this component are given in Table 2.

The cooler was first transported out of the reactor building into the turbine house. After filling up with water and freezing the component was cut with a big bandsaw into several slices with 800 mm height by applying the ice-sawing technique. The cut-off pieces were then transported back into a cooling container. A special device for defrosting of an ice-block including filtering and drainage of the molten water was used. About 12 kg of corrosion products with an activity of 9E+9 Bq have been collected in the drainage filters of the defrosting device. The heat exchanger tubes and the spacer plates have been mechanically picked up and put into 200 ltr. drums for final storage. The shells was successfully decontaminated by pickling in phosphoric acid.

For the total job an effort of 1100 manhours and a job dose of 21 mSv was recorded. A considerable time had to be spend for preparational works such as scaffolding, installation and building of equipments for lifting and transport purposes. About 40% of the total dose was used for segmenting and especially for handling of the heat exchanger tubes. The distribution of manpower and job dose is shown on Figure 2.

3. Dismantling of a primary clean-up cooler (B.1.3.)

A second large-scale application of the ice-sawing technique was executed with a primary clean-up cooler. The design of the U-tube heat exchanger is similiar to the shut-down cooler. The total length is about 6 m, the diameter 600 mm. The maximal contamination was detected to be 1600 Bq/cm^2. The procedure of removal and dismantling was the same as for the shut-down cooler. However, handling and packing of the highly contaminated tubes into drums was improved. In order to save personnel dose for this work step, remote handling was used. A shielding wall made out of concrete stones was built up. By using two mechanical master-slave-manipulators from behind this shielding it was possible to avoid direct contact to the tubes and to reduce the personnel exposure by a factor of 2 compared with manual operation. The procedure of packing is illustrated on Figure 4.

Ice-sawing and packing of the tubes have already been performed. Nowadays, the decontamination of the remaining shell will start. The total effort was calculated to be 1100 manhours, the personnel exposure to be 32 mSv.

4. Dismantling of the reactor pressure vessel head (B.1.5.)

It is intended to test several cutting techniques, such as plasma arc cutting, flame cutting and sawing at the head of the reactor pressure vessel (RPV) in order to compare cutting times, costs, aerosol release and waste quantities for each method. The head is made out of ferritic material with an inside austenitic cladding of 7 mm thickness (see Figure 3).

The cutting tests will mainly be carried out in a new cutting cell which has been installed in the turbine house.
Because of the large diameter of the head, a transport of the whole component through the lock gate was not possible. A pre-segmentation at the 31.8 m reactor floor was neccessary.
In order to find a suited thermal cutting technique for thick-walled components with an austenitic cladding preliminary inactive tests have been initiated. It could be demonstrated that flame cutting is applicable under the following conditions:

- using a special propane torch,
- starting the cut from the ferritic outside.

By applying a special propane-powder torch it was possible to cut the material. In a further test it could be demonstrated that it is also practicable to use the same torch, however without adding powder. This resulted in a reduction of smoke generation and filter consumption.
For the active pre-segmentation of the head at the reactor floor, a mobile cutting tent with two suck-off filtering devices was neccessary to avoid spreading of aerosols during flame cutting. First, the dome of the head was segmented into three single pieces. In a further step the flange was segmented also into three parts. Table III gives the main data obtained with the pre-segmenting of the RPV-head.

Mid of 1992, the tests of several cutting techniques will be continued in the new cutting cell under constant conditions.

5. Dismantling of the steam dryer (B.2.1)

It is intended to dismantle the reactor pressure vessel and its internals by applying mainly thermal underwater cutting techniques. In the framework of this R&D work optimized cutting techniques will be applied to segment the steam dryer and the water separator of the RPV.
The inactive test phase for underwater segmenting of the steam dryer at University of Hannover has been finished. Plasma arc cutting and wire-arc-cutting underwater proved to be to be qualified for an active application.
The tool carrier and a platform for the underwater segmenting of the steam dryer has been already delivered. All the equipment will be assembled in the next future. The first active cuts are scheduled for mid of 1992.

Table I: Manpower and job doses for dismantling of pipes and valves out of loop room-2

	Manpower [manhours]	Job doses [mSv]
Planning	20	0
Preparation	1250	6,7
Dismantling	1000	13,6
Transport	710	2,3
Segmentation	1440	6,0
Decontamination	350	1,1
Release measurements	40	0
Total	4810	29,7

Table II: Technical data of the shut-down cooler

		Shell	Internals
Material		Ferritic steel	CuNiFe-alloy
Mass	[tons]	0,8	2,3
Surface	[m^2]	14,0	156
Contamination	[Bq/cm^2]	1000	1000
Activity inventory	[Bq]	1,4 E+8	1,6 E+9
Diameter	[mm]	1104	
Length	[mm]	5323	

Table III: Results obtained with flame-cutting of the RPV-head

Range of wall-thickness	[mm]	80 - 200
total cutting length	[m]	21.4
Cutting surface	[m^2]	3
Width of the cutting gap	[mm]	10
Cutting velocity	[mm/min]	
- for 80 mm thick material		50
Max. aerosol concentration	[Bq/cm^2]	100
Consumption of		
- prefilters		16
- Hepa-filters		19

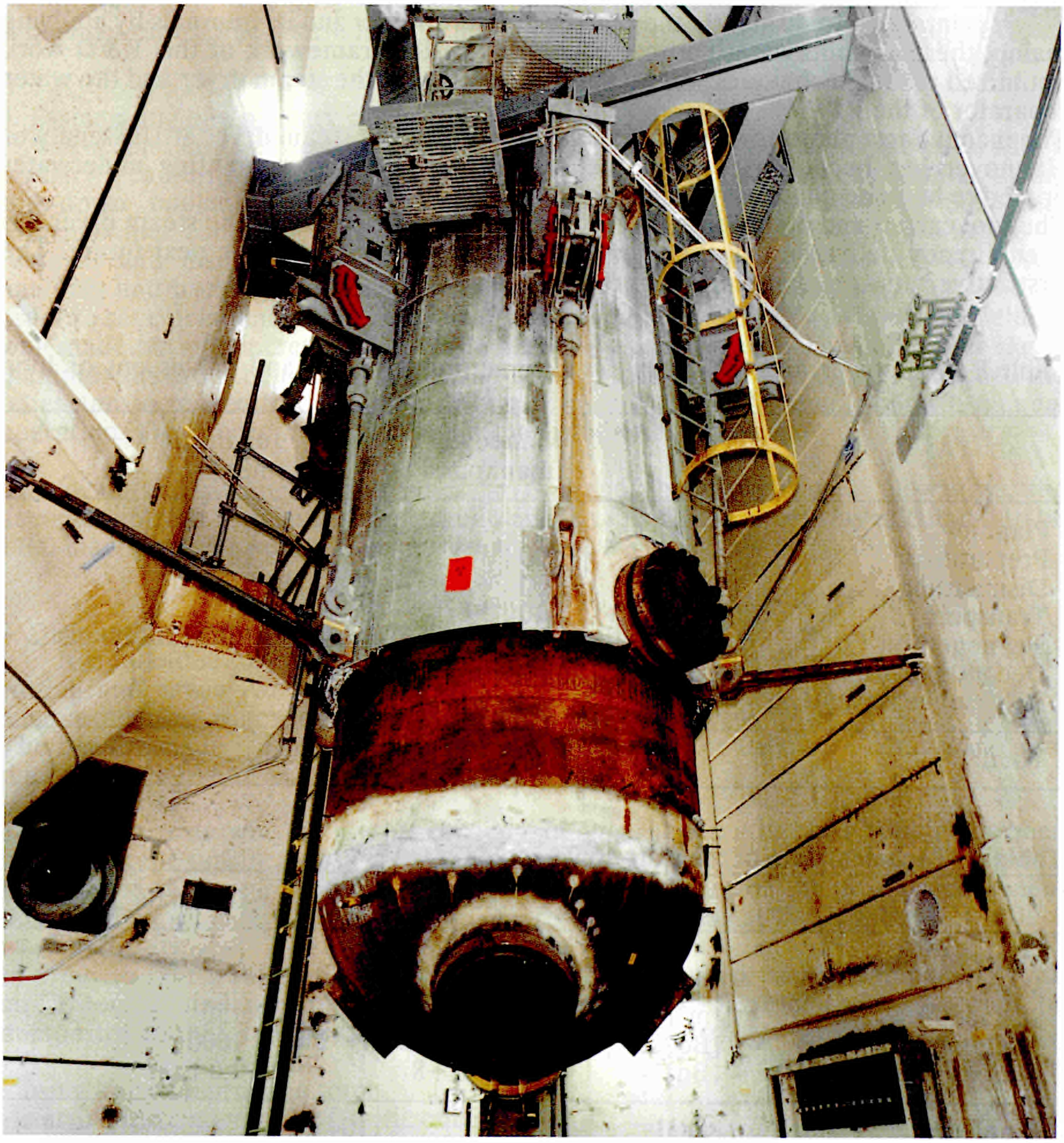


Figure 1: Looproom-2 with SSG-2 after removal of all pipings

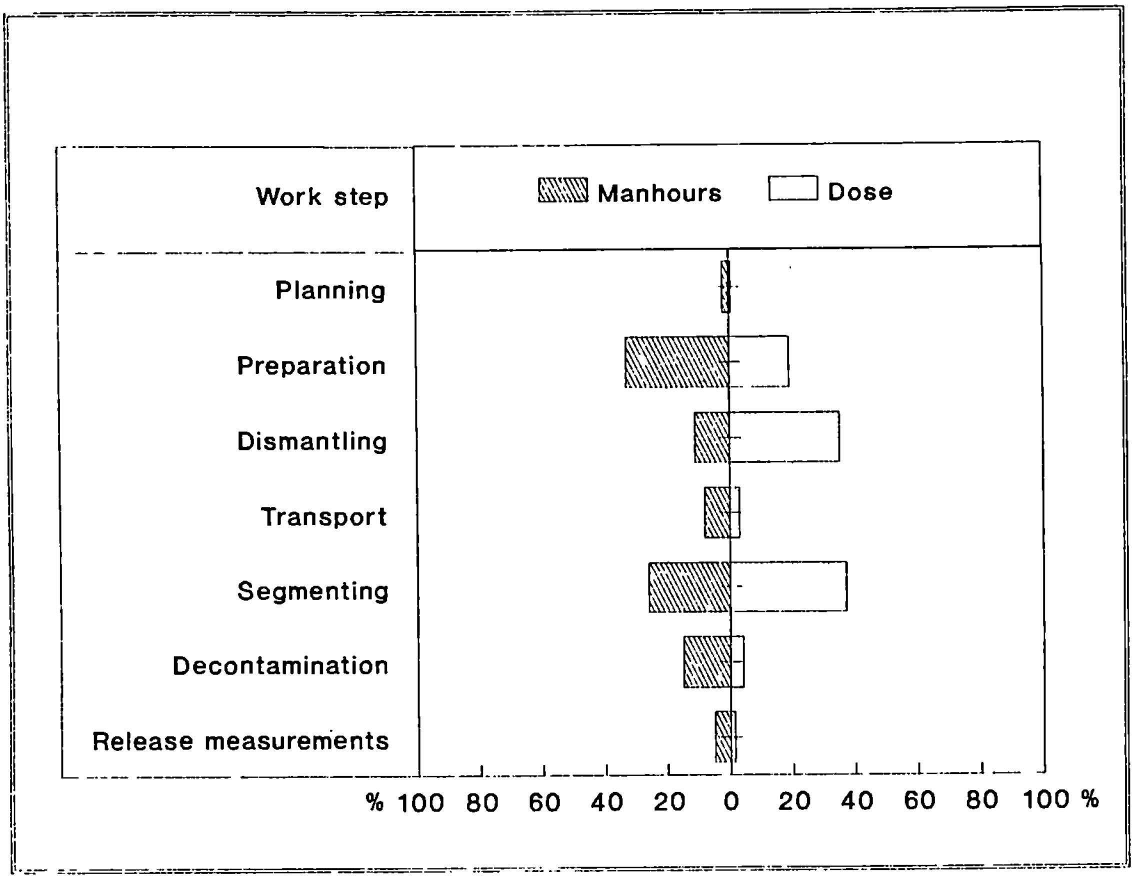


Figure 2: Distribution of manpower and job doses during dismantling of the shut-down cooler

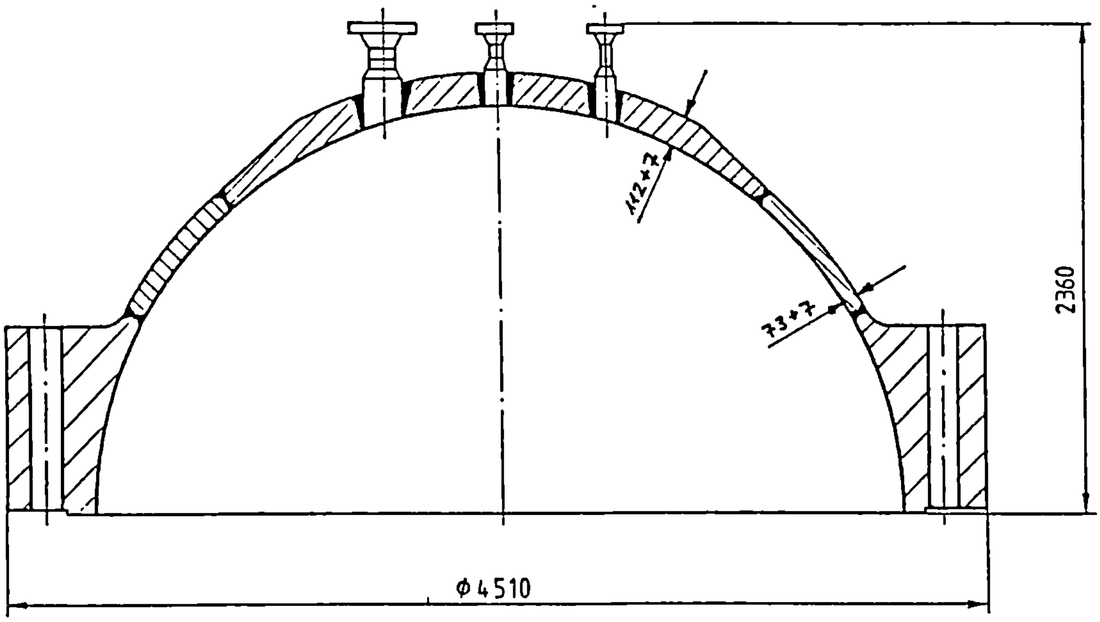


Figure 3: Dimension of the RPV-head

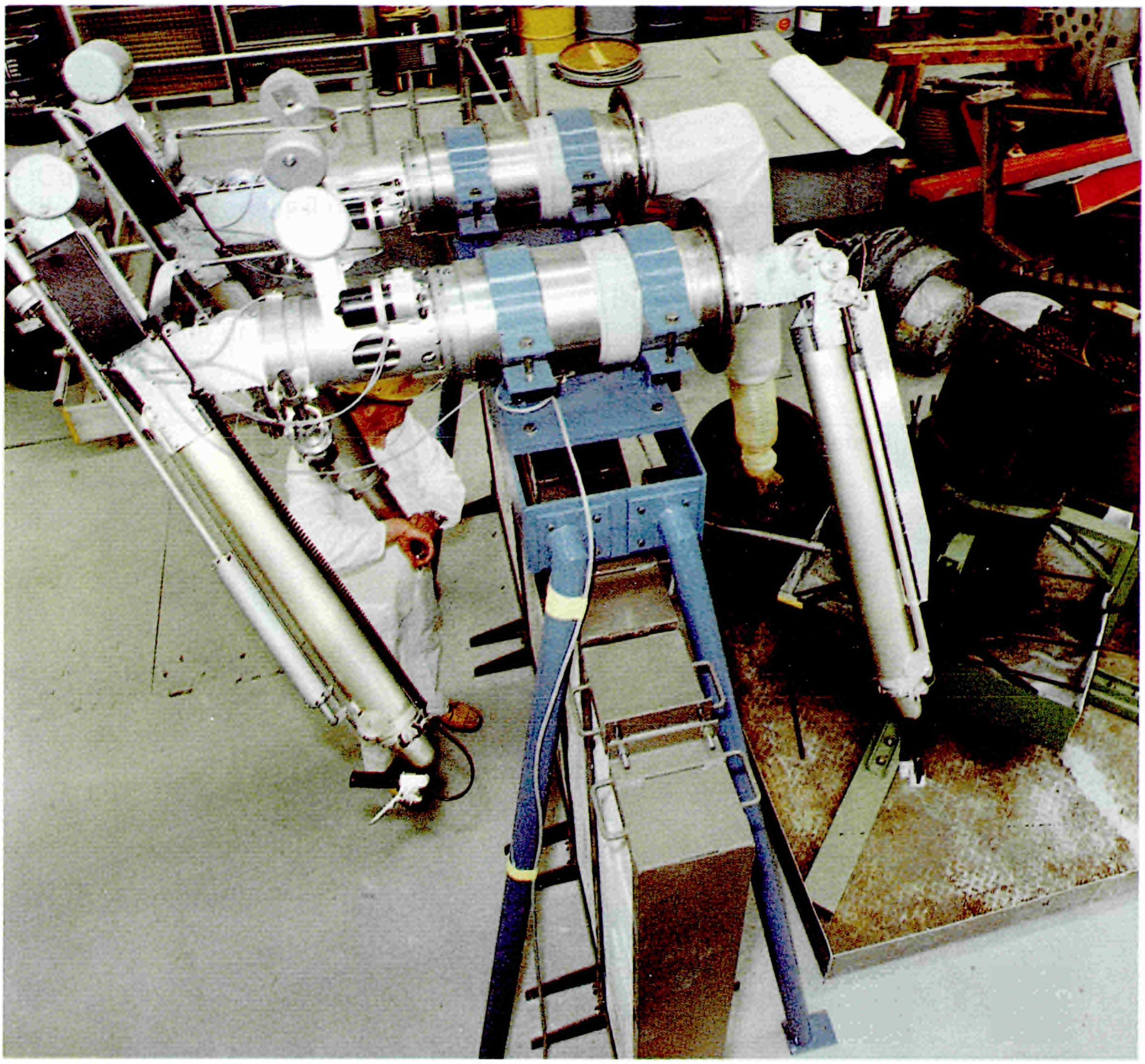


Figure 4: Handling of tubes of the primary clean-up cooler with mechanical master-slave-manipulators

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 obtained results

Summary of main issues

The main efforts during 1991 have been to remove fissile material, glove boxes, lead shielded steel boxes, the microscope and the tube transfer system, together with a large amount of scrap material from the concrete cell line. A few initiating contamination and radiation measurements from a shutter between the cells, from the shutter housing and from one relatively clean cell have been analyzed.

Progress and results

1. Removal of fissile material (B.1.)

The facility was nearly emptied for all fissile material. All solid material was packed in stainless steel containers. Some acid solutions from isotope analyses were neutralized with sodium hydroxide and mixed with cement contained in stainless steel containers. The material was transferred to the Risø Waste Treatment facility for storage. A minor batch of material remains in the concrete cells, packed and ready for export in 1992.

2. Removal of large contaminated equipment (B.2.)

An excess stock of ^{60}Co in the concrete cells was packed in shielded containers and transferred to the Risø Waste Treatment facility for storage. A large amount of scrap material including experimental equipment from the cells was packed and transferred to the Waste Treatment facility for storage. This extensive task is continued in 1992.

3. Removal of large contaminated facilities (B.3.)

Seven glove boxes used for experiments were either decontaminated, re-used at the DR3 reactor site, or filled up with further active scrap material and transferred to the Waste Treatment facility for storage. For the decontamination several cotton cloths impregnated with general household detergents were used as well as sponges with mild abrasives. A few "hot" spots were removed by sand paper.

Six α-tight lead shielded steel boxes for experimental use were decommissioned. Two boxes were re-used at the DR3 reactor site after minor cleaning. Four boxes were so heavily contaminated that decontamination would require too much work and give too high doses; they were stored at the Waste Treatment facility. The main contaminants in the glove boxes and the steel boxes were ^{152}Eu, ^{154}Eu, ^{134}Cs, ^{137}Cs and α-emitters.

At the end of the concrete cell line a remotely operated microscope was housed in a lead shielded glove box, situated on a reinforced concrete base. Only the microscope and the glove box were contaminated; these are now decommissioned. The lead shielding and the concrete base had to be removed as big units by a truck.

The out cell parts of a tube transport system between a concrete cell and a lead shielded steel box have been removed. They were lightly contaminated and stored as active waste.

4. Decontamination of concrete cells (B.4.)

4.1 Initial mapping of radiation levels in remote operation (B.4.1)

Above each door between the concrete cells a shutter is placed in a housing, which is directly connected to the cell volume. Therefore, the inner surface of the shutter housings as well as the shutters themselves are contaminated. In order to get a first indication of the level of contamination, a few smear tests were taken on surfaces of the shutter arrangement between cell 3 and 4, and in cell 4, together with some radiation measurements. Cell 3 is known to be a highly contaminated cell, while cell 4 is a relatively clean cell, primarily used for repair work of equipment in the cell line. The results are shown in Table I. The quoted contamination levels are on the smear tests themselves, i.e. a similar value for the tested surfaces assumes a 100% uptake of contaminants in the smear test. The 1000 mSv/h on the table in cell 4 is on a single "hot" spot; it is not representative for the whole table. The contamination level of the shutter housing is - based on these few measurements- an order of magnitude lower than the level on the shutter; this conclusion appears reasonable. In summary, the results are as expected for an area classified as highly contaminated. The main contaminants are as for the glove and steel boxes, i.e. ^{152}Eu, ^{154}Eu, ^{134}Cs, ^{137}Cs, ^{60}Co, and α-emitters.

Possible items for collaboration with Sellafield, UK, have been discussed. Sellafield will try to develop a method to remove the painting from the inner cell surfaces by some chemical agent. This approach is expected to be much more efficient than conventional cleaning of the surface of the painting. The open questions are whether it is possible at all to remove such old and extremely strongly bonding painting and how far it is possible to do it remotely.

Table I: Contamination and radiation levels on the shutter in cell 3/4 and on cell 4. The quoted contamination levels assume a 100% uptake of contaminants in the smear test.

Location	β-contamination MBq/m^2	α-contamination MBq/m^2	Radiation mSv/h
Shutter Cell 3/4	6	0.6	
Shutter housing Cell 3/4	0.2	0.02	
Walls/floor Cell 4	4	0.8	2
Table Cell 4	30	1	< 1000
Filter box Cell 4			15

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 for continuous vitrification of waste generated by reprocessing FBR fuel (PIVER II).

PIVER is the first vitrification cell for fission product solutions 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 1.1×10^{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.

The work to be carried out under this contract 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. 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 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 OBTAINED RESULTS

Summary of Main Issues

Dismantling and decontamination of the PIVER cell reduced the in-cell irradiation level to 0.1 $mGy \cdot h^{-1}$ by the end of 1990. Research and development work on enhanced decontamination processes was carried out from January to July 1991. Four mature processes (electrolytic, shot-peening, cryogenic and gel) were tested under actual, representative hostile conditions to assess their advantages, drawbacks and effectiveness.

The final decontamination operations were carried out on only a portion of the cell wall, and did not significantly affect the overall ambient irradiation level in the cell.

The dismantling and decontamination operations in the PIVER cell have now been successfully terminated and quantitatively assessed.

Progress and Results

1. Research & Development of Enhanced Decontamination Processes (B.2)

1.1 Decontamination Test Objectives

The PIVER cell was ideal for *in situ* evaluation of the four final decontamination processes (electrolytic, shot-peening, cryogenic and gel) under actual radioactive conditions. The tests were designed to compare their effectiveness, to obtain a maximum of information concerning large-scale utilization, to identify points requiring special attention for nuclearization, to specify particular applications for which each process is best suited, and to determine the decontamination limits feasible in highly radioactive environments.

1.2 Enhanced Decontamination Processes

The **electrolytic** process removes metallic ions from the contaminated surface by fixing them on the anode moistened with an electrolytic agent. This technique was developed by the CEA at Cadarache, and has been used at Pierrelatte and La Hague.

The **microblast** technique is a shot-peening process using glass microbeads as grit under a pressure of several bars. The mechanical action of the impacting grit allows removal of superficial contamination and induces work-hardening of the treated surface.

The **cryogenic** process is a similar shot-peening method using ice particles. Previous tests had shown that the method is effective in stripping off organic coatings, with deoxidation and degreasing effects as well as biological and chemical cleaning action.

The **gel** process involves applying a soluble gel on the work surface, allowing it to react and rinsing with a high-pressure water spray.

1.3 Results

The gel process diminished the activity by about 90% on average, but without ensuring complete decontamination. This process involves no major difficulties except the need to set up a barrier to protect the remainder of the cell from water spray during rinsing.

The other three processes are capable of achieving complete decontamination, after several applications. Shot-peening was found to be the most effective. All three processes require specific and often bulky equipment, although it may be set up outside the contaminated zone.

The electrolytic and cryogenic processes generate secondary liquid waste. Shot-peening generates solid, noncompactible waste, although the waste volume may be limited by recycling the grit.

Table I summarizes the comparative assessment, with criteria evaluated from 1 to 4 (where 1 is the best rating). These results could also be weighted by a coefficient to allow for specific working conditions at the decontamination site.

2. PIVER Cell Dismantling Operation: Quantitative Results (B.3)

2.1 Solid and Liquid Waste

Solid waste produced during the dismantling operations included:

- 37 cylindrical packages with a final volume of 71 m^3, containing approximately 1.7×10^{13} Bq (450 Ci).
- 25 cubic (5 m^3) containers with a final volume of 125 m^3, containing approximately 2.2×10^{13} Bq (590 Ci).
- 800 drums (100 l) of operational wastes with a final volume of 32 m^3 after compaction and conditioning, containing approximately 2.4×10^{12} Bq (65 Ci).
- 4 canisters of high level waste transferred to the vitrified waste facility for interim storage, containing 4.1×10^{13} Bq (2190 Ci).

Liquid waste production included the following:

- 4.5 m^3 of high level waste containing about 1.1×10^{14} Bq (2900 Ci) from internal decontamination of process equipment.
- 4.7 m^3 of high level waste containing about 8.1×10^{13} Bq (2200 Ci) from external decontamination of process equipment, and from washing the cell walls.
- 17.3 m^3 of intermediate level waste with about 8.9×10^{12} Bq (240 Ci) from decontamination of the cell walls.

The total activity removed from the cell was about 3.2×10^{14} Bq (8655 Ci), broken down as follows:

- 61% liquid waste (58% HLW and 3% ILW) representing about 26.5 m^3,
- 39% solid waste (25% HLW and 14% conditioned in ANDRA containers for surface disposal) representing about 230 m^3.

2.2 Radiological Assessment

The initial in-cell irradiation level of several $Gy \cdot h^{-1}$ (1 $Gy \cdot h^{-1}$ at the cell entrance with peaks of up to 10 or 20 $Gy \cdot h^{-1}$) was reduced to an ambient dose rate of about 0.1 $mGy \cdot h^{-1}$. Additional biological shielding has been set up around some highly localized hot spots.

The ambient dose rate cannot be further reduced, as it is primarily attributable to the fission product liquid storage facility located beneath the cell.

The overall dosimetry results totaled 0.47 man-Sievert, itemized as follows:

- 50% for tool maintenance and repairs;
- 30% for decontamination of the cell and adjacent rooms;
- 20% for removal of the waste packages.

2.3 Human Resources

A total of 100 000 man-hours were required to complete the PIVER decommissioning operations. This figure includes:

- 50% by CEA technicians (cutting, maintenance, waste removal),
- 25% by subcontractors with piecework contracts,
- 25% for supervision and engineering studies.

2.4 Cost Assessment

The total budget for the PIVER decommissioning operation was about 50 million French francs ($8 million), itemized as follows:

- 26% for engineering studies and preliminary operations (including equipment investment costs),
- 24% for general services (waste treatment, health physics, etc.),
- 24% for CEA labor costs,
- 15% for subcontracting firms,
- 11% for supplies (biological shielding, cutting tools, etc.).

2.5 Overall Assessment

Figure 1 illustrates the percentage variations of three major parameters for the PIVER decommissioning operation: work progress (evaluated by assigning points to each scheduled task), dose rate at the cell entrance, and financial cost.

3. Conclusion

The program provided valuable information relevant to decommissioning of PIVER, a nuclear installation with high irradiation levels and severe fission product contamination, which may be applicable to other facilities of a similar nature.

Activity removal in liquid form is highly effective: it can avoid the use of equipment in very hostile environments, and thus minimize maintenance requirements and occupational doses. It is the least expensive decommissioning method, and generates the smallest amount of secondary waste requiring disposal.

PIVER was not only the world's first waste solidification facility capable of vitrifying high level fission product solutions, but also the first of its kind to be decommissioned.

Table I. Comparative Assessment of Final Decontamination Processes

Comparison Criteria	Electro-lytic	Shot-peening	Gel	Cryogenic
Decontamination efficiency per operation (stainless steel)	3	1	-	2
Final decontamination quality	1	1	-	1
Cost	2	3	1	4
Process (constraints)	1	2	1	3
Waste production	1	3	1	1
Risks, hazards, exposure & ease of implementation	2	3	2	1

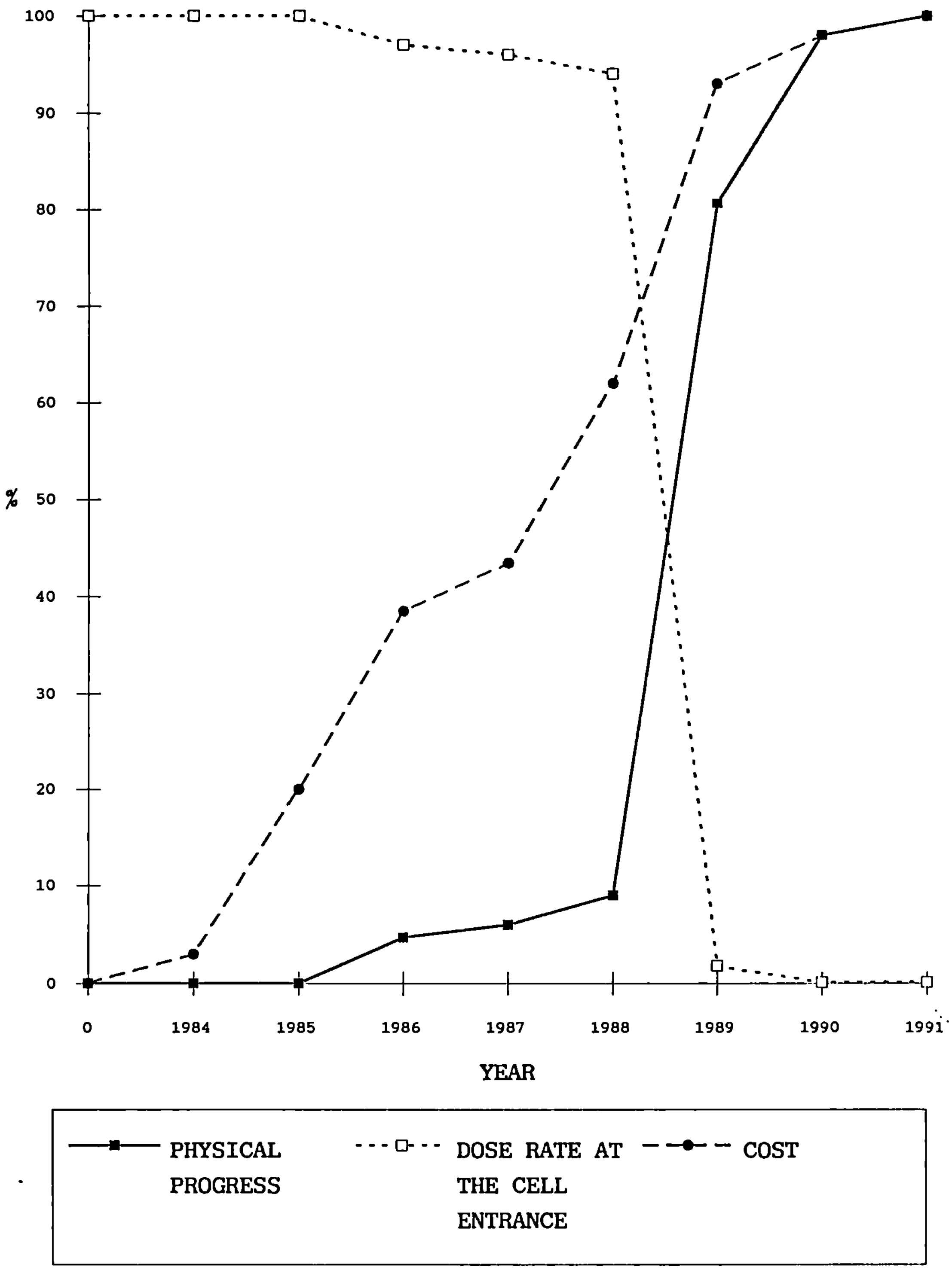


FIGURE 1 : DISMANTLING PIVER BALANCE

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: P ANTOINE, 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

Abstract and main results

All basic studies started in 1990 and the draft safety report are completed.
The equipment, divided into four groups, is being manufactured. The construction of the "soda" group is finished and the plant is installed in the containment of Rapsodie. The start-up of the Noah prototype was delayed after various operations of nuclearisation and modifications. Active commissioning tests are planned for the beginning of 1992.

Progress and results

B.1.1. Studies for the industrial application of NOAH

This study has three main objectives:

1.1. Improving the quality and the reliability of the main components on the NOAH prototype which has been used with non-contaminated sodium.
1.2. Testing a hydrogen drying process based on a combination of filters and condenser;
1.3. Establishing a method to determine the balance of contamination (activity):

- Concerning the first point, tests carried out on NOAH have allowed us to get an important experience feedback:
  - . due to a small leak of sodium, without fire, the test has been stopped and a new concept of totally welded valves box has been foreseen for DESORA;
  - . due to the presence of gas, the oil of the pump had to be removed in order to recover the right delivery pressure.

  On account of these two occurrences and the delay for reception of the gas drying system, the start-up is now scheduled for the beginning of 1992.
- In this system the submerged filter 01F1 breaks up the hydrogen bubbles and screen mesh 02F1 serves as demister of the rising gas stream; both systems have been supplied.

  A first couple of mesh size have been chosen, in order to check the choice made for DESORA; to improve the filtering effect, two other sizes will also be tested for each filter 01F1 and 02F1 (see figure N° 1).

  The gas circuit tightness has been checked.

  Then, the operating conditions of the cooling circuit (cold water) have been changed in order to improve the condenser operation.
- The first tests with moist helium on the gas drying process will be carried out within the next two months.
- Concerning the third point a strategy to perform measurements on the sodium, soda and hydrogen streams has been proposed for DESORA and is being optimized.

  To this purpose, an overall mass balance using the flow-sheeting process simulator "PROSIM" has been developed to estimate the efficiency of each filtration level, which has to be checked on the NOAH facility.

B.1.2. Studies for the installation of NOAH into the RAPSODIE containment building including the required auxiliary equipment

The plans concerning the manufacturing and installation of all equipment are finished. The supply was divided into four distinct groups, as follows:

- The "soda" group is made up of all equipment ensuring the transformation of sodium into caustic soda, with all its auxiliary circuits and equipment.
- The "sodium" group includes the different circuits and tanks necessary for the transfer and supply of sodium to the process. However, the third group is made up of certain equipment specific to sodium technology; it includes the pumps, the flowmeters and the level indicators.
- The last group includes the control and instrumentation equipment, including monitoring and

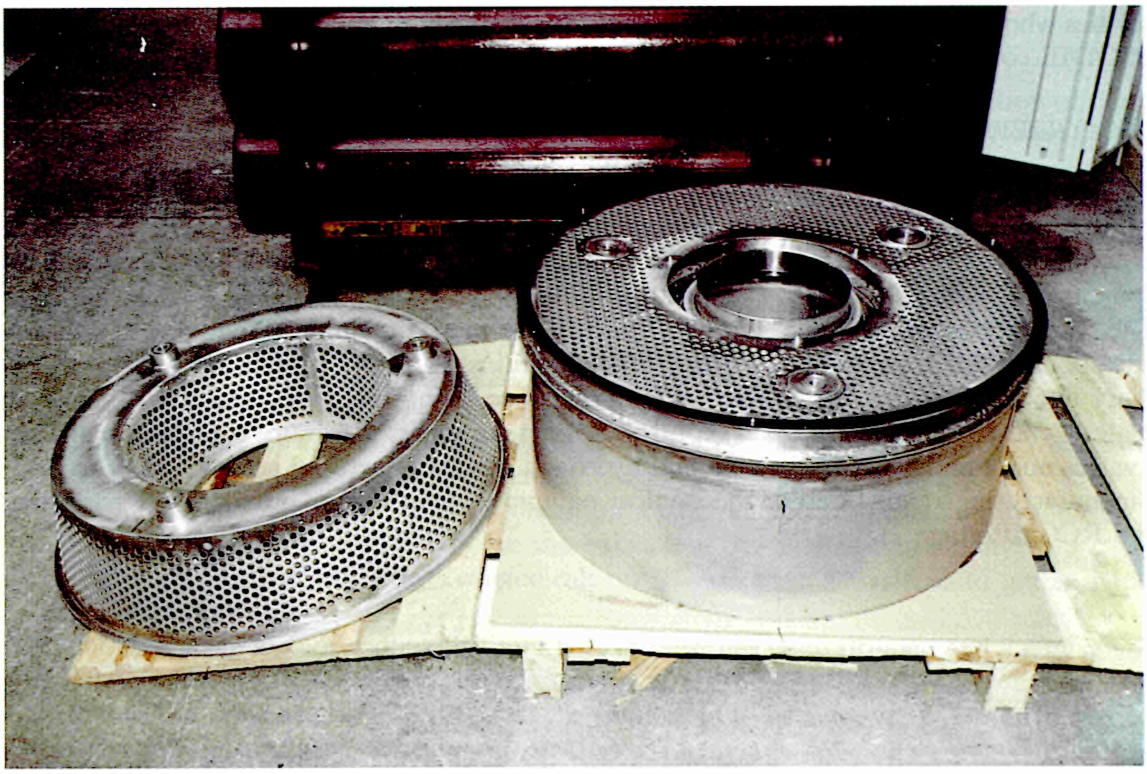
power supplies.
A safety report, with a view to operating the facility, was written and examined by the safety authorities who gave their agreement for manufacturing the equipment and its installation in the RAPSODIE containment building.

B.2. Manufacturing, installation and testing of equipment

B.2.1. Manufacturing and installation of equipment

All the orders dealing with the equipment manufacturing and installation have been placed. After tender procedures, the same firm was entrusted with the supply of both "soda" and "sodium" equipment.
The construction of the "soda" group is finished, the rack containing all equipment necessary for the sodium-soda transformation was delivered and installed. Its connection to the different auxiliary circuits is under way. The other groups are being manufactured according to the estimated general schedule. A quality control programme covers the entire construction process. The valves and equipments for sodium and hydrogen are manufactured according to the design and manufacturing regulations of mechanical equipment of the fast breeders nuclear islands (called RCCMR stage 3).

On the left stand 01F2 On the right stand 01F1



On the left disc for 01F1 On the right conical section for 01F2

Figure 1: Filter housings and supports

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 - December 1993
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. Development, manufacturing, testing and subsequent installation in the JEN-1 reactor of an underwater cutting facility 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. Underwater dismantling of reactor internals 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. Decontamination of reactor internals (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. Melting of aluminium waste (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. Assessment of radiation protection 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. Generation of specific data 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 obtained results

Summary of main issues

The main aim of this Project is the study and development of decontamination, cutting and melting techniques on contaminated or neutron activated materials using some components of JEN-1 reactor.

By November of present year, license has been awarded to take out the fuel elements from the pool to shielded wells, sited in the reactor building itself. Forty spent fuel elements have been taken out from the pool. This operation carried out, has allowed to start the first activities of dismounting and characterization of components from JEN-1 reactor core.

On the other hand, and in a parallel way with the above operations, work went on during this year as design, manufacturing, plasma arc cutting tests, decontaminations tests and melting facility construction are concerned, just like activities focussed to the purchase of different systems, needed to undertake the operations related to the dismantling of JEN-1 reactor core components.

Progress and results

1. Radiological characterization of components to be dismantled, decontaminated and melted (B.1) (CIEMAT)

The first component which has been dismounted for characterization was one control blade housing. Measurements contact raditation rate were done upon the two faces, resulting in a maximum radiation rate of 20 mGy/h. By means of mechanical drilling, 24 rings 15 mm ø were taken out. The radioactive content analyses are given in Table I. The shown results are those obtained by means of γ-spectrometry. Analyses for α and β emitters on the same samples are being assessed.

2. Development, manufacturing, testing and subsequent installation of an underwater cutting facility (B.2)(CIEMAT, IW)

2.1. Cutting tests with a plasma torch and a consumable electrode tool (B.2.2).

Cutting tests with both tools were performed on aluminium sheets and profiles with representative dimensions relating to the components to be dismantled in JEN-1. These were in detail aluminium plates with a thickness of 3mm, 6mm, 20mm, 40mm, 80 mm and 120mm (only for the consumable electrode tool). Furthermore cutting tests were performed on L-profiles 60x60x6mm, T-profiles 25x25x3mm and double-T-profiles 100x100x10mm.

X-ray diffraction analysis were done on dried suspended particles samples, inferring that they are about Aluminium oxides Al_2O_3. Coagulation tests were performed on these coloidal particles in order to notice their precipitation behaviour, pointing at the visibility problems taking place when cutting Aluminium materials.

Sodium phosfate has proved to be a successful electrolyte, which in a 1:1 molar ratio, with regard to Al_2O_3, gave rise to a whole precipitation of suspended particles, keeping a completely transparent water in less than 24 hours time.

2.2 Desing and manufacturing of a cutting facility (B.2.4)

The JEN-1 reactor core components, are going to be cut inside a basin, placed on its turn, in the reactor pool. Its design has been achieved, and it will consist of a stainless-steel basin 3x2x5m, provided at its bottom with a tray helping for support of components to be cut and collection of sedimented dross.

In the first stage was intended to use the auxiliary bridge of the JEN-1 reactor as one axis of the tool guiding device for motion in three dimensions (x,y,z). Because this would have to be modified to a large extent in order to obtain a motorised bridge with a sufficient rigidity required for the cutting operation, it is advisable to construct a completely new device. The design for this is in progress at the moment, while that concerning the two other axes, a vertical mast with a square cross section with an overall length of 6.5m performed by a framework of square tubes, and the carrier moving the mast horizontally along the bridge, is completed. All components of the handling device consist of stainless steel.

The fine positioning of tool will be done with the aid of a sensor, which was developed at IW. This sensor device consists of two independent sensors, one ultrasonic and one eddy current sensor, of which the windings are positioned around the ultrasonic head.

The system for gazeous effluents coming up from cutting operations has been designed. Such effluents will be extracted by means of a 2000 m^3/h fan through a 300 mm zinc coated duct and HEPA filters battery, ejecting them to the JEN-1 general ventilation.

Activities were undertaken for specifications and procurement of following equipment: welding rectifier for consumable electrode cutting, pressure pump for waterjet (aiming at expelling away the kerf material when consumable electrode cutting is taking place), control video system to pursuit operations development, analyser for ultrasonic sensor signals for fine positioning of cutting tools and exhaust system for gazeous effluents evolving up from cutting operations.

3. Underwater dismantling of reactor internals (B.3) (CIEMAT, IW)

Once the spent fuel elements moved out from JEN-1 reactor pool, the first reactor core dismounting activities were started by CIEMAT. The work has been initiated without lowering the pool water level. In this way the dismounted components were the control drives, the control shafts, the control blades and one control blade housing.

4. Decontamination of reactor internals (B.4) (ENSA, LAINSA)

ENSA has carried out a planning work selection about different electrolytes to achieve electropolishing tests on Aluminium samples.

Several chemical decontaminant reagents have been tested by LAINSA on cold Aluminium samples in order to remove Ferric and Aluminium oxides from their surfaces. Some of these tests have been enhanced by means of ultrasonic generation. The different chemical decontaminants were HNO_3, OHNa, OHNa + $KMnO_4$, followed this last with step of $C_2O_4H_2$ and chemical reagent composed by a surfactant, H_3PO_4, HF and EDTA.

The main conclusion from the above tests is the advantage in the use of ultrasonic generation as an enhancing on decontamination operations in what a lower time consuming and attack to the base metal are concerned.

5. Melting of aluminium waste (CIEMAT,ENRESA) (B.5)

Induction furnace was installed in CIEMAT in order to undertake melting of Aluminium scraps as a recycling material method. The engineering project of such facility has been achieved, just like the desing of the civil work, whose construction will be started on the early 1992.

6. Assessment of radiation protection (B.6) (CIEMAT)

In order to keep the operator doses As Low As Reasonably Practicable, the really operations of dismounting and sampling are being carried out at an enough water depth, so that the radiation level over the water surface be kept at a background one (<0.2 mrad/h).

Table I. Control blade housing nº 2. Activity (Bq/g)
Face A

Sample	Cs-137	Co-60	Eu-152	Eu-154	Eu-155
0-1	$4'92 \times 10^{1}$	$7'73 \times 10^{2}$	$6'29 \times 10^{1}$	----	----
0-2	$6'03 \times 10^{1}$	$7'47 \times 10^{2}$	$6'40 \times 10^{1}$	----	----
0-3	$4'29 \times 10^{1}$	$4'18 \times 10^{2}$	$3'69 \times 10^{1}$	----	----
0-4	$1'43 \times 10^{3}$	$1'31 \times 10^{5}$	----	$1'93 \times 10^{3}$	$6'07 \times 10^{2}$
0-5	$2'38 \times 10^{3}$	$2'01 \times 10^{5}$	----	$2'76 \times 10^{3}$	----
0-6	$6'96 \times 10^{2}$	$7'92 \times 10^{4}$	----	----	----
0-7	$4'74 \times 10^{2}$	$6'81 \times 10^{4}$	----	----	----
0-8	$9'58 \times 10^{2}$	$1'16 \times 10^{5}$	----	----	----
0-9	$3'96 \times 10^{2}$	$4'00 \times 10^{4}$	----	----	----

Face B

Sample	Cs-137	Co-60	Eu-152	Eu-154	Eu-155
E-1	$5'11 \times 10^{1}$	$4'22 \times 10^{2}$	$5'44 \times 10^{1}$	----	----
E-2	----	$7'07 \times 10^{2}$	$8'10 \times 10^{1}$	----	----
E-3	----	$4'51 \times 10^{3}$	$4'44 \times 10^{2}$	----	----
E-4	$9'25 \times 10^{2}$	$9'73 \times 10^{4}$	----	$1'16 \times 10^{3}$	$2'88 \times 10^{2}$
E-5	$3'02 \times 10^{3}$	$2'56 \times 10^{5}$	----	$3'49 \times 10^{3}$	$8'88 \times 10^{2}$
E-6	----	$1'26 \times 10^{5}$	$3'43 \times 10^{2}$	----	----
E-7	$3'36 \times 10^{2}$	$3'74 \times 10^{4}$	$3'92 \times 10^{2}$	$3'60 \times 10^{2}$	$6'44 \times 10^{1}$
E-8	----	$1'34 \times 10^{5}$	$3'57 \times 10^{2}$	----	----
E-9	----	$6'81 \times 10^{4}$	$8'92 \times 10^{1}$	----	----

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: L PACHL, 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 5×10^{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/cm² (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 OBTAINED RESULTS

Summary of main issues

After evaluation of seven decommissioning studies (two for VAK and five for KRB-A) a number of questions were still incomplete. Therefore, we performed the "additional evaluation" of these studies with NIS and NUKEM to make the principal choice of the cutting and handling technology. It resulted in a dismantling concept based on cutting technologies that produced a minimum of aerosols.

As a result, plasma melt cutting (PMC) for dismantling of core internals was exchanged for mechanical cutting techniques like milling, grinding and electro-discharge machining (EDM).

In future however, we will generate specific data on secondary wastes from these cutting techniques. We will evaluate these techniques against PMC under comparable application conditions.

The importance of a bi-arm electromechanical master slave manipulator (EMSM) in our dismantling concept is considerable. At this time it will not be used for handling mechanical cutting tools, but only for installation of a video system, placement of sucking nozzles, changing of tools, and support of the container handling, etc.

The RPV as well as the internals shall be cut to shipping package size on site. The EMSM shall support this undertaking. The presence of an EMSM is necessary to achieve high performance for all the decommissioning equipment components.

The RPV wall and the rotation-symmetrical internals like the core case, thermal shield, and similar components shall be cut by milling or EDM using a circumferential tool support. We have successfully performed sawing and milling tests with a welding cladded RPV sample, both under water and in the atmosphere (Figure 1).

For dismantling of RPV internals other special tools shall be used. We have therefore planned grinding cutter and EDM tests with typical internal material samples under water using these tools (Figures 2, 3).

Progress and Results

1. Conceptual studies of a 1:1 scale facility for UW testing of cutting tools (B.1.)

After the "additional evaluation" of the above mentioned seven decommissioning studies it was established that the components for the cutting and handling technology and for the secondary waste collection and filtration systems should not be developed separately. This is the primary reason for the extension of tasks for the planned VAK test installation. The VAK test installation should not just simply perform underwater cutting tests, but also simulate the RPV space relations during dismantling of internal samples using the workshop approved cutting/handling and waste/aerosol treatment equipment.

The space conditions at the VAK turbine hall will allow simulation of the space relations of the reactor side (+3m level). The planned test installation will also be able to simulate the 1:1 scale space conditions of the future "hot works" under water.

We have therefore planned an "adoption pool", tightly connected to the RPV flanges to increase the water shielding during UW container loading.

The "adoption pool" would be used (supplied with a bottom only) as a part of the planned test installation at the VAK turbine hall.

2. Cutting tool tests for RPV wall sample and rotation symmetrical internals (B.2.)

We carried out cutting experiments at the Siemens/KWU workshop in Erlangen (FRG) to gain experience with mechanical cutting tools for the RPV wall and for rotation-symmetrical internals.

The cladding technology and the cutting sample material are identical to the KRB-A wall. However, the RPV cladding material and the RPV wall material differ from those of the VAK wall.

Due to the existence of the so-called "heat affected zone" (HAZ), caused by welded cladding, the cladding technology leads directly to different material parameters. For that reason we needed a mechanical cutting tool which is able to cut safely along the HAZ.

Regarding the VAK RPV wall material (6% of the entire wall thickness is cladding material), our cutting experiment was carried out with a material sample which was more difficult to cut than the original VAK wall.

The cutting levels S1 and S4 represent the successful performance of the most difficult cutting depth: cutting along the HAZ (Figure 1). Thus, the cutting tests carried out represented a more conservative approach to this problem.

We vacuumed different kinds of milling swarf using a nozzle which is connected to the filter equipment. The filter equipment can be used under water and is composed of a simple strainer as prefilter, two independent HEPA- filters, and a pump.

Due to the large size of the milling swarf, it is possible to collect the bulk of this material (>95 mass-%) in the prefilter. The strainer unit is easy to handle and its geometry is compatible for transport and storage containers. For example, after one circumferential cut through the VAK RPV wall we will receive approximately 30kg of swarf, which corresponds to 50 liters of uncompacted waste.

3. Cutting tool tests for internals (B.2.)

For dismantling of complex geometry internals, we have planned cutting experiments using grinding and a back-up technique using EDM. EDM offers relatively simple secondary waste handling and high reliability.

The application of EDM to thin-walled internals such as the chimney and primary water distribution ring is one of the best methods for their dismantling.

3.1. Underwater grinding tests (B.2.)

Underwater grinding is a state of the art offshore dismantling technique, yet applications to nuclear components are also known (Ref./1/).

Underwater cutting experiments by grinding are planned for dismantling of internals like grid plates. The planned cutting method for our upper grid plate is based on the application of a hydraulically driven grinding disc, moving only in a vertical direction. This method allows the internals to be cut into a transportable size. The cross section to be cut is 8X147mm (Figure 2).

When designing the underwater cutting tool, the priority is given to the collection of the grinding swarf using a special grinding disc cover. The grinding stream is directed to the sucking nozzle which is connected to the filtration and pump unit. The tool guide will allow for the vertical movement of the cutting tool. The reaction force "F" is variable and measured by a spring balance. The cutting disc circumferential speed depends on the air pressure (Figure 3).

The use of compressed air has advantages versus hydraulic motors although the torque depends more on the rotation speed. The tool is watertight up to 60m in depth (because of the toolpressure). The use of compressed air presents no risk from leaks, which is not the case for hydraulic motors.

References

/1/ BRANT, A W; BELL, A; WILLIAMS, R C, ENC 90 Conference, Lyon, September 1990, Text of Presentation, "Isolation Of B5 And B15 Water Ducts At BNFL Sellafield"

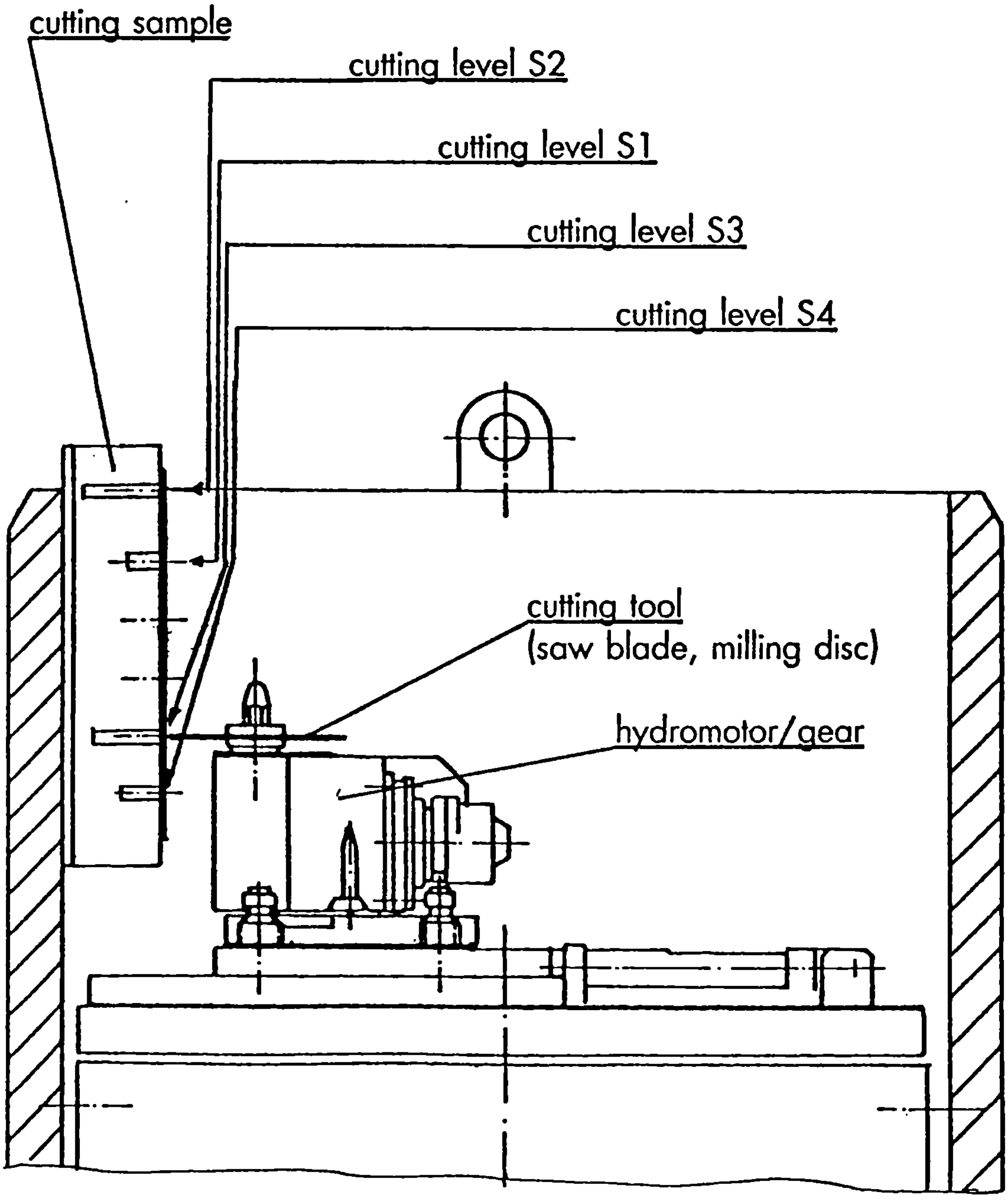


Figure 1: Underwater milling cutter test

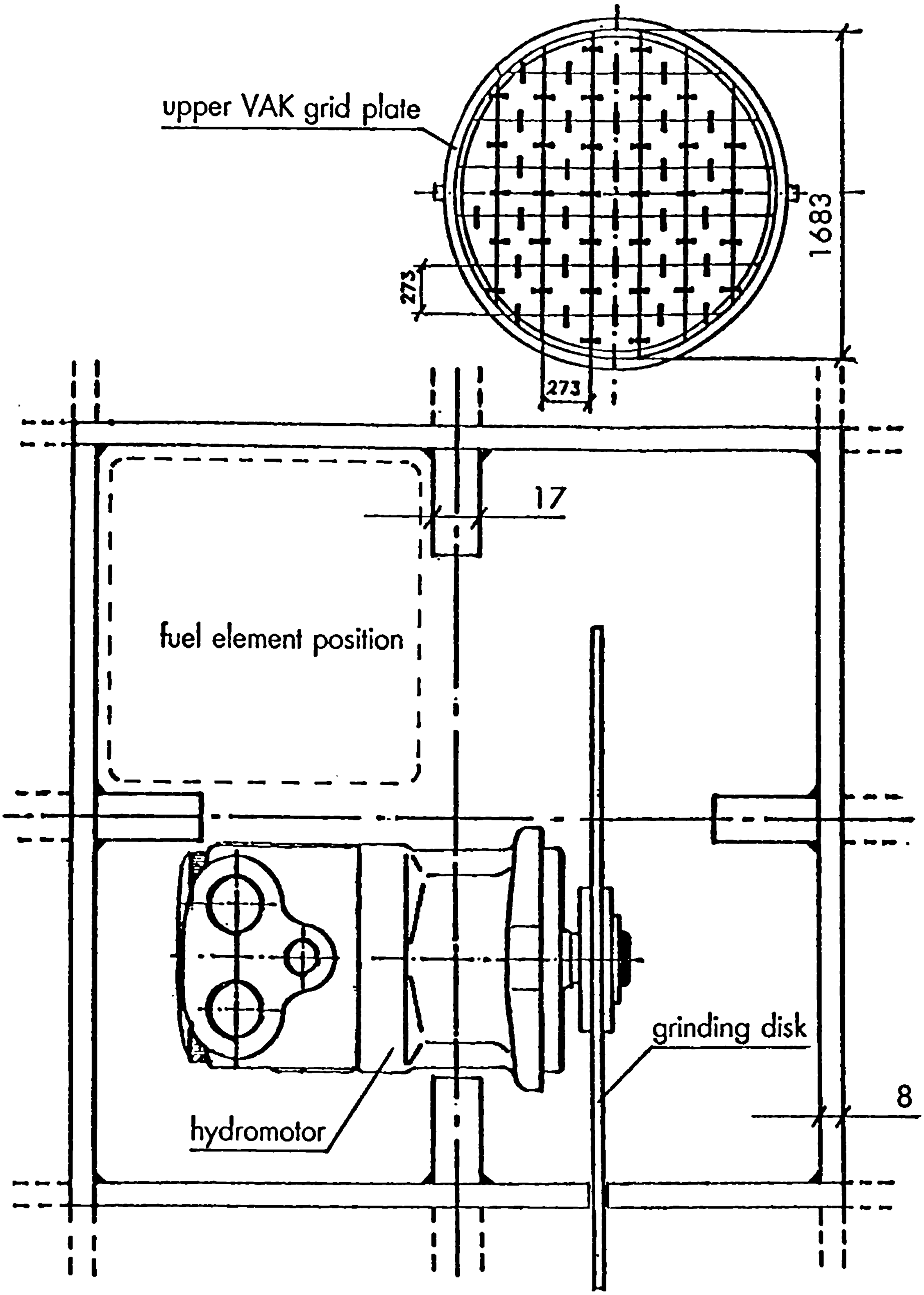


Figure 2: Planned cutting method for upper grid plate

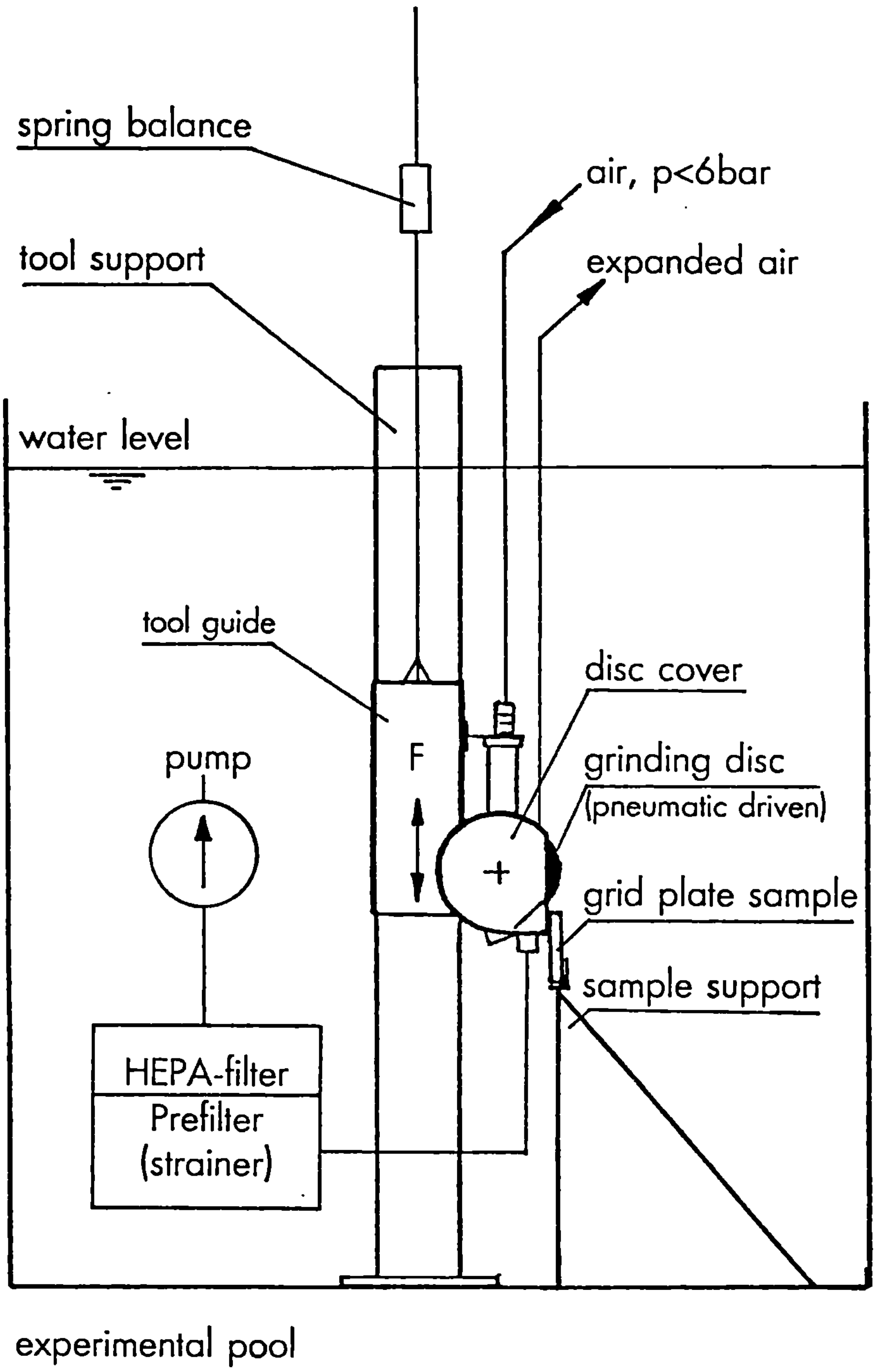


Figure 3: Underwater grinding cutter test

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 - December 1991
Coordinator: J L DECITRE, 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 studies carried out during 1991 1st semester, the melting installation has been installed inside the G3 reactor building. Inactive testing started on October 3rd 1991 and will be followed by real tests with active scrap during 1992 1st semester.

Progress and Results

1. Conceptual studies and categorisation of waste (B.1.)

1.1 Studies for the installation of the melting furnace in the reactor building. (B.1.1.)

The installation studies of a melting facility have now been completed. They consist of :

- nuclearization and proportions studies of an arc furnace
- specific civil works studies.

The steel melting plant and its biological shielding provisions were designed on the basis of working hypotheses for :

* the melting mass balance,
* the activity of the scrap steel to be melted,
* the activity distribution during melting operations.

The following assumptions were postulated for the activity level of the steel scrap :

* 250 $Bq.g^{-1}$ maximum design basic activity,
* 30 $Bq.g^{-1}$ maximum for melting operations carried out during decommissioning to stage 2.

The mass balance per metric ton of cast steel produced was estimated on the following assumptions.
The total quantity of dust particles generated during the melting process (including graphite and refractory particles) is estimated at 10 kg per metric ton of cast steel.
The slag production is estimated at 10 kg per metric ton of cast steel (i.e. 1%).
The ventilation system is assumed to trap 97,5% of the particulate matter, i.e. 2,5% of the particles generated will remain as fallout in the melting zone.
Hot gases are cooled through a water-jacket running from the furnace and through an air-water cooling system before the HE/VHE filters.

1.2. Investigation into the management of waste streams before and after melting. (B.1.2.)

The standard radioactivity spectra for the G3 and G3 circuits are well known, and it is unnecessary to carry out systematic spectrometry measurements on the steel supplied to the furnace. The activity will be quantified by γ scanning of the dominant radionuclide (^{60}Co) at various points outside the 1,7 x 1,7 x 1,7 containers used for transportation and storage of dismantled equipment.
Several cylindrical core samples weighing about 100 grams each will be taken inside the furnace during the melting operations. The samples will be measured by spectrometer.

The 25 kg ingots produced will be sampled. After cooling the samples will be submitted to individual external γ counting.
A preliminary fusion has already been carried out in an industrial furnace controlled by the Radiation Protection Service of Marcoule (S.P.R.).
It consisted of 3,8 metric tons of sections of piping coming from G2 and G3 reactors with a residual activity of 2,4 $Bq.g^{-1}$.
The objectives were threefold :

1) to confirm the estimate of radionuclides partition between the molten scrap, the slag and the dust.
2) to analyze after casting the homogenization of radionuclides in the resulting ingots similar in shape to those of the future melting facility.
3) to measure radiations at the different work stations.

Concerning the above second point :

* ^{60}Co (95 % of the activity in stage 2 decommissioning scrap metal) ; cobalt tends to remain in the steel ; it is assumed that ^{60}Co will be found in the dust particles, slag and castings in the same proportions as in the molten scrap.
* ^{137}Cs (5 % of the activity in stage 2 decommissioning scrap metal) ; based on reports of contaminated steel melting experiments, and allowing for the small quantities of slag expected from melting of previously cleaned G2/G3 scrap metal, it is assumed that 90 % of the ^{137} Cs will be found in the dust particles and 10% in the slag.

The activity balance shows that the specific activity of the dust particles produced will be approximately 5 times higher (150 Bq.g-1 for 30 Bq.g-1 in scrap metal).
The slight drop in the activity of the cast steel (0,91 $Bq.g^{-1}$ compared with 1 $Bq.g^{-1}$ in the scrap metal) is due to the inclusion of uncontaminated additives (which lower the specific activity to 0,95 $bq.g^{-1}$), and to the elimination of ^{137}Cs.
The mass and activity balance studies resulted into the definition of the right equipment for the cleaning system :

- a cleaning filter capturing the majority of dust particle : bag filters (JET-LINE type),
- a finishing filter : cleanable high efficiency filter
- a VHE filter as the last confinement barrier.

The last two are together in an ANKE type caisson.

2. Manufacturing, installation and testing of equipment (B.2.)

2.1. Manufacturing and installation of equipment including auxiliary and control systems (B.2.1.)

After some dismantling and demolition works of equipment and zones located on the East of the G3 reactor building, the building of the melting facility was carried out.
It comprises a melting furnace, a casting ladle and preheating system, a graphite and silicon injection machine, a ladle tipping device, control stations and utility rooms, electrode nippling station.
The 15 ton capacity furnace (overall dimensions 5700 x 4860 x 5140 mm) includes the following major components :

* The refractory brick roof pivots horizontally and includes four ports : three for electrodes and one for the furnace off-gases. The off-gas port or "4th port" collects hot furnace gases that are ducted toward the water jacket, a double-wall water-cooled heat exchanger. The furnace roof is moved upward and pivoted to load the furnace vessel with scrap for melting, then is immediately closed again.
* The melting vessel receiving solid and molten scrap has an outside diameter of 3200 mm.
* A pouring spout is used to transfer the molten metal from the furnace into the casting ladle.
* A system of actuators is provided to tilt and secure the furnace, to open the descaling door, to raise and lower the furnace electrodes and to lift the furnace roof.

The casting line comprises 104 ingot molds, each designed to cast three 25 kg ingots.
The feed rate of the ingot casting line is variable.
The ladle pouring rate is continuously modulated.
After solidification, ingot cooling is accelerated by spraying with water so the ingot temperature does not exceed 800°C at the end of the casting line. The line is designed for casting 6 to 12 metric tons per hour, during a casting period not exceeding two hours.
The nominal graphite load is supplied when the furnace is charged. The injection machine is used to feed graphite or silicon to the melt if the composition does not meet specification requirements. The machine is used only for composition adjustment purposes.
The electrode nippling station is used to remove or reinstall spent electrodes.
Control stations are located in the melting control room and in the ingot casting control room.
The ventilation zone houses the bag filters, the HEPA filters, the exhaust and ventilation blowers.
The Operating Utility Rooms house, the maintenance room, the hydraulic room, and the electrical rooms.

2.2 Commissioning testing of the melting facility (B.2.2.)

Operations test are half-done. Active testing will begin at the beginning of 1992.
Inactive testing were carried out to :
- verify the good running of the melting facility and adjust the equipment,
- verify the ventilated anticontamination confinement of the facility during normal, incidental and safety operating conditions,
- verify the resistance of materials and filters during the different phases of the process in normal, incidental and safety operating conditions,
- verify the expected balance of dust (volume, distribution in the melting area, filters and the various components of the extraction system).

Seven melting tests were carried out between October and December 1991, when 77 tons of inactive scrap were introduced in the furnace excluding the additives (4 tons of graphite and 2,4 tons of silicon).

These tests proved the good resistance of the different materials. They also confirmed the assumption related to heat removal and dust covering of the melting area.
The ventilation system, major component of the melting facility has been commissioned: flux regulation and pressure of areas, good operating and performance of the devices of the cooling system (water-jacket, RE003 exchanger and air-coolers) and the efficiency of the filtration barriers (jet-line, pre-separators, HE filters and VHE ANKE of the last confinement barriers) have been verified.
The control instrumentation has been tested.
During these tests, melting operations in real or simulated conditions (incidental and safety) were performed.

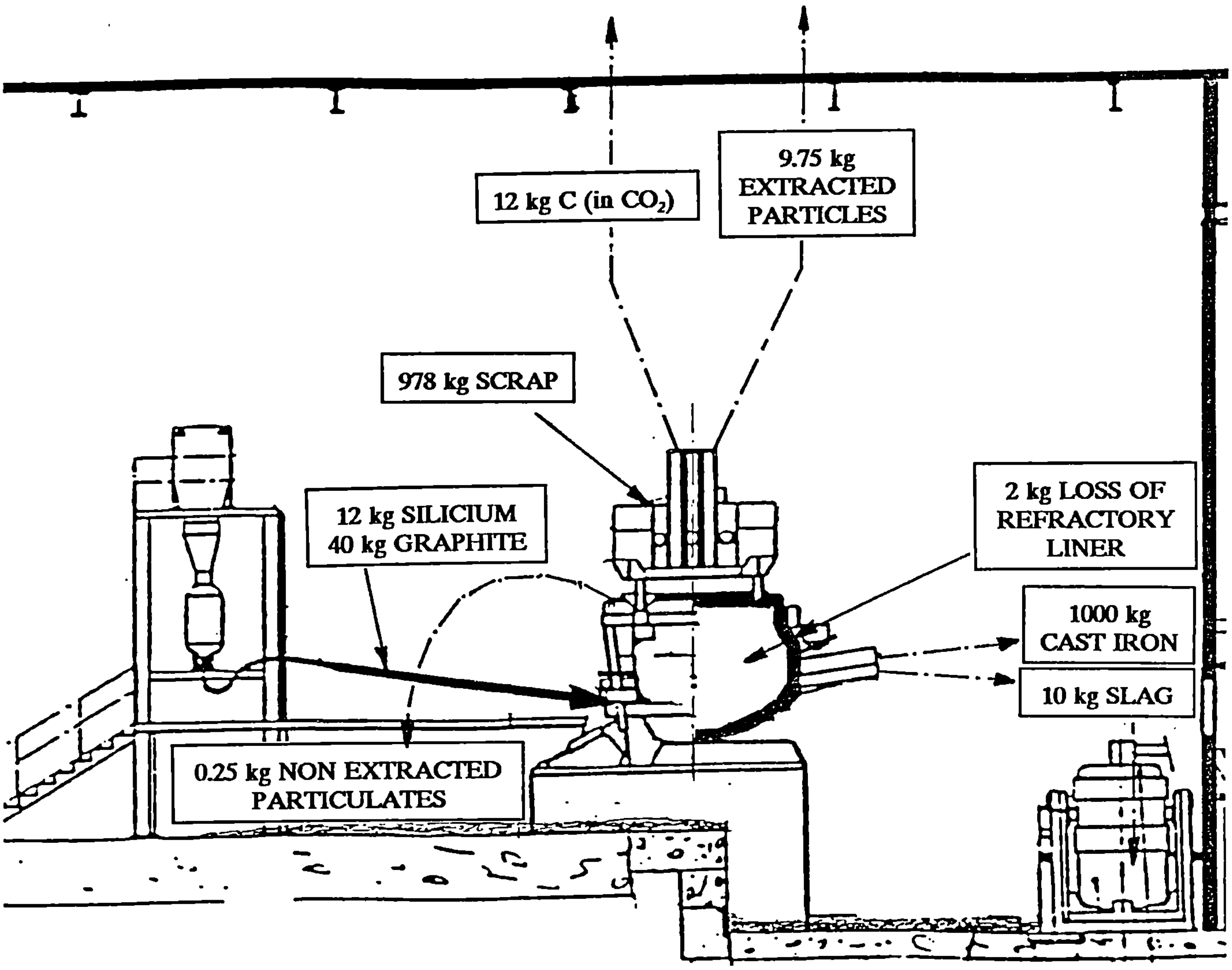


Fig. 1: Mass Balance of the G3 melter

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

At SG, a part of the necessary procurement has been realised for this research programme to-date.

The delivery of 100 t of material for the melting campaign at SG has been terminated. The melting approval has been given, the permit for the handling of the radioactive substances for the residual materials due to nuclear fuel concentration has not yet been obtained from the authorities.

The test programme for the performance of the trials at SG has been elaborated and must still be coordinated with the authorities.

During the first working period in the Siemens laboratories the experimental equipment for the melting of steel scrap was installed and tested.

The main components are the electrical power supply unit and the inductively heated furnace.

Progress and results

1. Identification of appropriate materials for crucible and slag formers (B.1.)
   Special crucible material and slag former material are not necessary. SG uses a conventional Al_2O_3 crucible and common slag binding material.

2. Installation of an induction-heated laboratory furnace and execution of reference tests with non-radioactive material (Siemens) (B.2.).
   An experimental plant was designed for the melting of contaminated steel scrap at laboratory scale. It consists of an electrical power supply with a generator, an inductively heated surface, a carrier gas supply and an aerosol-filtering system (Figure 1). We intend to use the equipment also for the melting experiments with radioactive contaminated aluminium scrap. This work will be part of the contract No. FI2D-0037.
   Electric power supply
   A generator for the inductively heated furnace was procured. The technical data are: 500/250 V, 197/294 A, 7.3 KVA, 10 MHz.
   The outer dimensions are 150 x 80 x 182 cm. During operation, there is a need of nearly 1 m^3 of cooling water for the electric power supply together with the installation of the cooling system and the connection between generator and furnace was finished and already tested.
   Inductively heated furnace
   An old furnace from earlier experiments at Siemens KWU was modified for the contractor's purpose. A new coil and new flanged covers with a new passage for the pipes of the coil were added. The volume of the furnace is 200 l. The walls are made of stainless steel. Within the coil, there is space for a crucible with a volume of 1.5 l.
   During a melt experiment, a carrier gas flow can pass through the furnace. Before reaching the air ventilation system of the laboratory, the gas stream will be filtered by an aerosol filtering and sampling system.
   First test runs with the heating of a carbon crucible was successfully performed.

3. Laboratory-scale melt tests with U-, Th-, and Pu-contaminated scrap (B.3.)
   No significant work was completed in the period.

4. Procurement of alpha-contaminated material (B.4.)
In order to be able to take over the 100 t of alpha-contaminated materials of Siemens Brennelemente Werk Hanau (SBW), a license was necessary.
The measuring device at SBW was therefore inspected together with the authorities. The U-238 activities, measured with the measuring device must be converted (see Annex) by means of a calculation direction for the investigation of the total activity of alpha-contaminated material (SBW), elaborated by SG. The calculation direction by SBW was not accepted by the responsible authorities.
The 100 t of alpha-contaminated material have already arrived at SG. The average activity of the material is fissile activity (0.43 Bq/g), other activity (11.41 Bq/g); total activity: 11.85 Bq/g. An estimation of the amount of nuclear fuel content showed that the nuclear fuel concentrates in the residual material. The melting approval was given to SG.

5. Main melting programme of about 100 t of alpha-contaminated scrap
SG developed a test programme for the performance of the melting campaigns. This test programme must yet be approved by the authorities. It describes the handling of the individual melting ranges as well as the measuring programme for the melting campaign/melting range.
Authorization was given to SG at the beginning of December 1991. The time schedule for the melting campaign was determined from January 1st, 1992 to April 4th, 1992. The furnace is prepared for the new task. The general view of arrangement is shown in Figure 2.
In order to fulfil all safety regulation, SG set up, in close connection with the responsible authorities, labour protection rules, suited to this melting campaign.

Annex: relevant data for melting of alpha-contaminated scrap

1. Requirements for melting alpha-contaminated scrap

Fissile nuclides	U-233, U-235, Pu-239 (Pu-241)
Radiation protection ordinance - § 2.2; concentration limits:	
- specific activity	100 Bq/g
- fissile material	1 g/100 kg
- waste including fissile nuclides	3 g/100 kg

2. Calculation of maximum permissible concentration in scrap containing fissile nuclides

Regulatory limits	a) 1 g fissile mat./100 kg scrap or b) 100 Bq/g scrap
Specific activity for U-235	84,500 Bq/g U-235
Activity concentration U-235	84,500 Bq/100 kg scrap = 0.845 Bq/g scrap
Applicable limit	mass limit (a)
Specific activity for U-233	$3.55.10^8$ Bq/g U-233
Activity concentration U-233	$3.55.10^8$ Bq/100 kg scrap = 3,550 Bq/g scrap
Applicable limit	activity limit (b)

3. <u>Scrap coming from Siemens-Fuel Element Production</u>

Enrichment ± 4%.
Example for activity content of the scrap under the following assumption:

U-238 activity is 1 Bq/g scrap.

	Bq/g scrap		Bq/g scrap
U-238	1.0	Th-281	0.281
U-236	0.083	U-234	8.27
Po-216	0.008	Ru-220	0.008
Po-212	0.005	Bi-212	0.008
Pa-234	1.0	Th-234	1.00
U-232	0.028	Pu-239	< 0.01
Ra-224	0.008	Pu-241	< 0.01
Pb-212	0.008	U-233	0.15
Tl-208	0.003	U-235	0.281
		TOTAL	12.16

Scale up for total activity from max. permissible fissile nuclides,
with respect to U-235 only for:
scrap 0.84 Bq U-235/g scrap -> 36.4 Bq/g total
slag 2.52 Bq U-235/g scrap -> 109 Bq/g total
with respect to U-233 only for :
scrap 100 Bq U-233/g scrap -> 4327 Bq/g total
slag not relevant.

4. <u>Situation after melting</u>

Example:

Quantity of scrap	100 kg
Quantity of slag	2 % = 2 kg
Specific activity in the initial material (scrap)	1 Bq/g
Assumption	all uranium will be transferred to the slag
Specific activity in the slag	10^5 Bq/2 kg = 5 . 10^6 Bq/100 kg
U-235 activity in the slag	1.16 . 10^5 Bq/100 kg
U-235 activity in slag, limit	2.54 . 10^5 Bq/100 kg

Result:
Under the above assumption, a total activity of 2.2 Bq/g scrap is allowed to remain within the limit for slag waste.

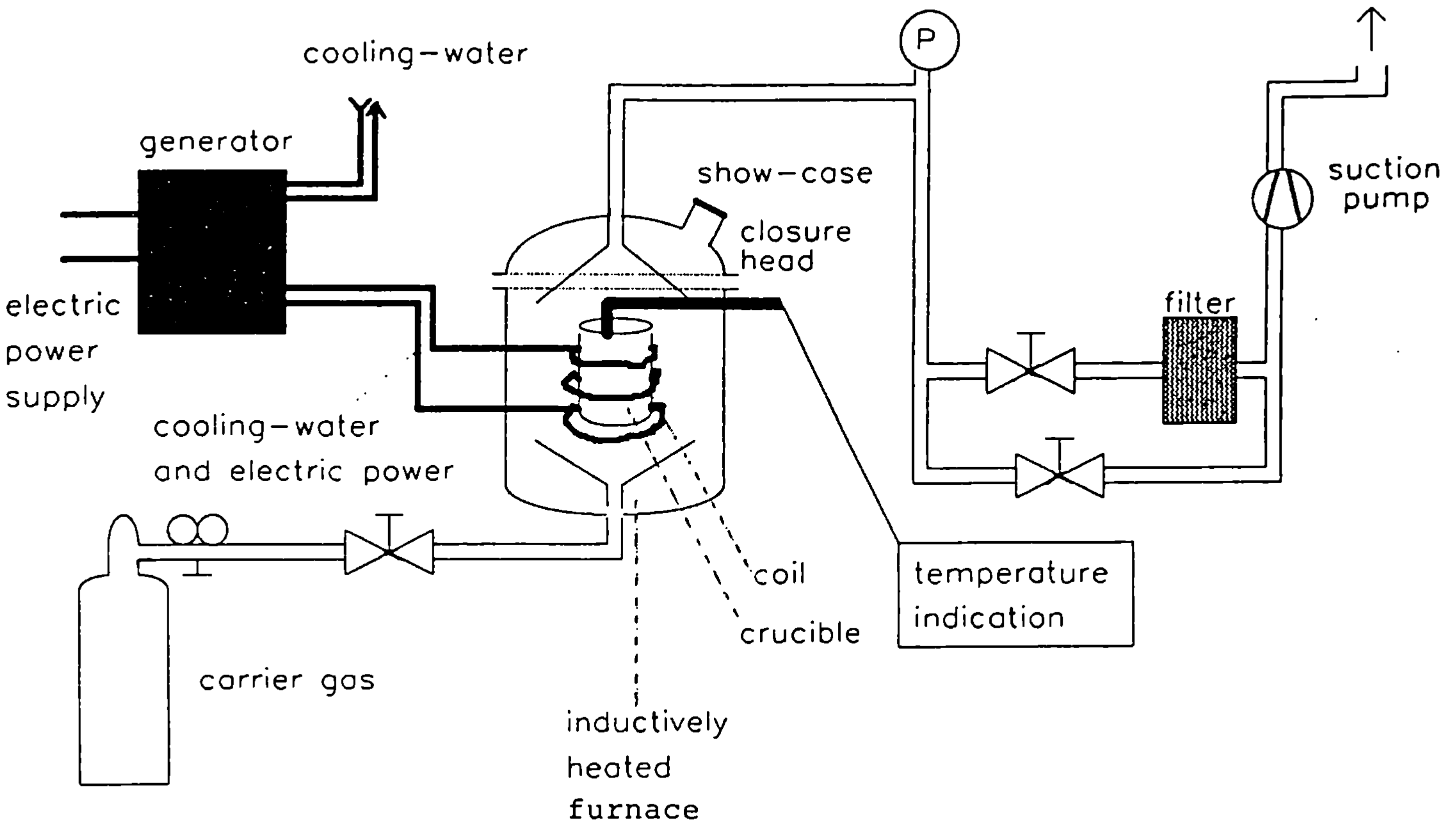


Figure 1: Schematic principle of melting contaminated steel

① melting bay
② inner housing
③ furnace
④ charging device
⑤ tubes to the filter system

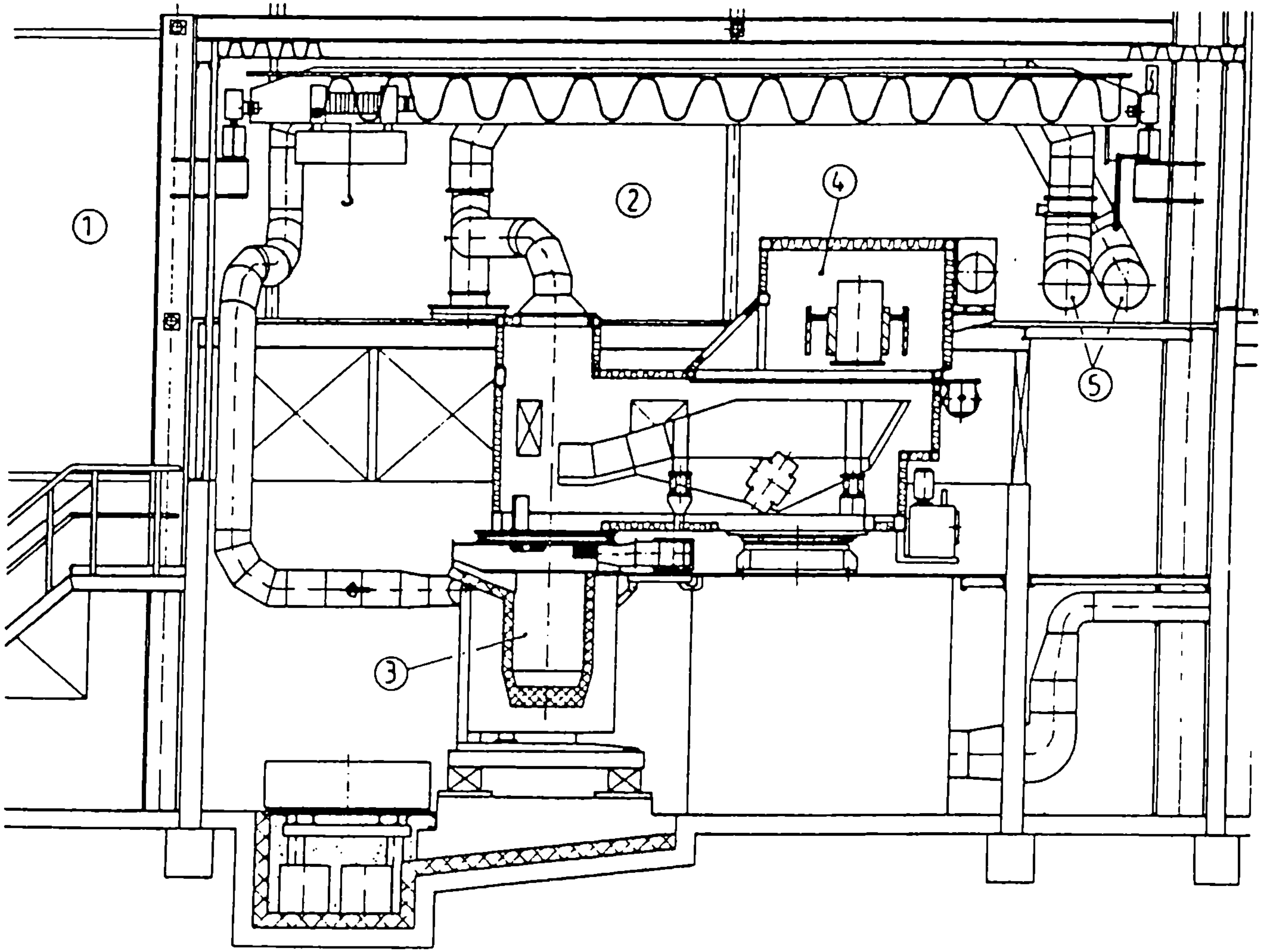


Figure 2: Layout of the new plant - inner housing and furnace.

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 - October 1993
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 3.7×10^9 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. <u>Assessment of structural safety</u> (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. <u>Final assessment of the blasting procedure</u> (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. <u>Related investigations of general applicability</u> 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. <u>Generation of specific data</u> 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 obtained results

Summary of main issues

The bore hole patterns for three selected sections are determined together with the layout of the explosive charges.

Core drilling and full volume drilling will be performed respectively, depending on the presence of reinforcement steel.

Finite element models have been developed to calculate the structural response of the reactor building and auxiliary building using PC programmes.

A multi-compartment blast model was developed to calculate the blast wave loading in various regions of the containment.

The quasistatic pressure rise in the containment is calculated to be 1,5 mbar for a 15 kg blast which is far below critical limits for transient air pressure wave. However, conservative structural response calculations show that a 15 kg blast may exhaust the safety values for vibrational excitation given in the german standard DIN 4150 for industrial buildings in use.

Further calculations show that the maximum amount of explosive charge per blast is restricted by the requirement of structural integrity of blast area installations, e.g. adjustable working platform. The effect of the blast wave pressure on the working platform limits the explosive charge to about 1,8 kg for long vertical holes and 2,1 kg for short horizontal holes. Protective devices may allow to increase these values.

Progress and results

1. Introduction

The biological shield of the Niederaichbach Nuclear Power Plant (KKN) will be used as a test bed to demonstrate the smooth bore hole blasting technique for the removal of radioactive concrete. The tests will be integrated into the total radioactive shield dismantling. The work performed so far is part of the detailed planning for these tests. The compiled results will serve as supporting documents for further use of explosives during the shield dismantling process. The licensing authorities which have granted a general license for explosive concrete removal in the biological shield require such detailed documentation prior to the beginning of the shield dismantling.

The blasting technique will be applied together with conventional hydraulic hammer technique and steel cutting tools.

2. Characteristics of the biological shield (B.1.)

The biological shield is part of a cylindrical concrete structure located off centre inside the reactor containment with access to the shield's inside from the top (former reactor platform) and from the bottom (collector room).

The total wall thickness is 1,20 m with heavy reinforcement. The concrete material contains baryte aggregate for improved γ-ray shielding (density: 3,6 g/cm^3; strength: BN40). The radioactive inventory of the biological shield caused by irradiation of the concrete and the steel reinforcement is very low, estimated in the order of $3{,}7 * 10^9$ Bq or 0,1 Ci.

3. Pattern and charging of bore holes (B.1.1.)

The biological shield column is divided into three sections of different complexity:

section 1: the upper ring (collar) which has a very strong circumferential reinforcement, Section 2: the cylindrical wall with a uniform array of 5 reinforcement layers at the inner surface,

Section 3: the wall openings (feed through for the primary cooling loop) with strong three directional reinforcement

The three sections are shown in Fig. 1 in a flattened view.

The hole pattern will be drilled in manual operation from the adjustable working platform by support-mounted drilling equipment. For vertical drilling heavy equipment will be mounted and operated from the reactor platform. The bore hole diameter will be in the range of 30 or 40 mm with both tools.

The explosive charge will be composed of commercial high explosive of PETN type. The specific charging as deduced from previous experiments is determined to be in the range of 0,25 to 0,4 kg explosive mass per ton of baryte concrete.

In all cases the holes will be plugged after charging by precast cement plugs which will be sealed by rapid bonding cement.

4. Pressure rise due to air blast in the containment (B.6.3.)

The blast wave of the gaseous explosive reaction products will expand from the biological shield into the containment mainly through the top opening but also through the collector room below and vent openings. A four-compartment-model was designed to calculate the pressure build-up in the blast area and in the reactor containment. For 15 kg explosive mass the blast causes an instant pressure rise of 1,5 mbar in the containment which is far below any dangerous level.

5. Limitation of the charge mass per blasting step due to local damage threshold for the blast wave (B.6.3.)

The near field installations (working platform inside the biological shield, air ducts) impose much more severe restrictions on the blast wave and thus the charge mass per blasting step than installations elsewhere in the reactor building. The maximum static load per unit area of the adjustable working platform inside the biological shield is given to be 0,2 bar. Employing dynamic enhancement factors the maximum tolerable quasistatic pressure is calculated to be

P_{QS} = 0,13 bar for long bore holes
P_{QS} = 0,12 bar for short bore holes

Using the theory of Baker et al. /1/ and a factor which describes the damping ratio for the restrained blast of concrete-embedded charges it is possible to calculate the explosive charge mass. For long bore holes the maximum explosive mass is 1,8 kg and for short bore holes it is 2,1 kg. The results are given in Table 1.

These limits may be exceeded when protective measures are taken, which will have to be designed.

6. Structural response to the blast load (B.4.2.)

For general application of the explosive dismantling method the essential characteristics of shock and vibrational behaviour of the structures have to be obtained. The calculation of the vibration of the reactor building and the adjacent buildings are performed using a finite element model. Blast data are obtained from results of the previous HDR tests /2/.

The model for the reactor and auxiliary building is composed of a few structural elements: blocks, hollow cuboids and cylinders as shown in Fig. 2. The finite element model is constructed as a simple 2-dimensional model using bars and springs as shown in Fig.3. A conservative loading force is taken from the HDR experiments: it idealizes the blast wave force for a 15 kg blast as a square pulse of $6 * 10^6$ N lasting for $50 * 10^{-3}$ seconds acting vertically on the shield column.

The modal analysis yields the eigenfrequencies and eigenmodes of the structure. The first eigenmode is included in Fig. 3. As can be taken from the figure this mode (eigenfrequency 0,3 Hz) is essentially an excitation of the pipe whip oscillation of the chimney.

Further results of the modal analysis are used to give a conservative calculation of the oscillation velocities at various locations of the buildings. This calculation uses the square root sum of the eigenmode amplitudes. The results for oscillation velocity amplitudes q_{max} are given in Table 2. The selected locations of interest are given together with respective model nodes and the q_{max} values together with safety values according to the german standard for building vibration limits DIN 4150.

As can be seen the 15 kg blast exhausts the safety margins of the DIN standard. Though a conservative estimate this result suggests that single blasting steps should not exceed about 10 kg of explosives unless proven safe by more detailed calculation.

<u>References</u>

/1/ BAKER, W E, COX, P A, WESTINE, P S, KULESZ, J, STEHLOW, R A, Explosion Hazards and Evaluation, Elsevier (1983)

/2/ FREUND, H U, SCHUMANN, S, Explosive Dismantling of Concrete Structures and Pipes, -Generalisation of the HDR Tests, Results and Consequences for Real Nuclear Installations-; Report Pt SN2 (1990), (Proceedings of the 1990 BMFT-KFK Programme Status Conference)

Table I: Maximum tolerable blast wave amplitudes (quasistatic pressure) p_{max}, related explosive mass m_{ex} for detonation and concrete mass removed m_{conc}

	long (vertical) bore holes	short (horizontal) bore holes
p_{max}	0,13 bar	0,12 bar
m_{ex}	1,8 kg	2,1 kg
m_{conc}	4,4 t	3,4 t

Table II: Oscillation velocities q_{max} calculated for 15 kg blast and safety values according to DIN 4150

Building position	**Node No.**	q_k^{max} [mms^{-1}]	v_{DIN} [mms^{-1}]
Foundation	1	22.9	20
Reactor building	4	21.3	20
Working platform	9	44.6	40
Top reactor building	13	26.4	40
Top auxiliary building	28	37.4	40
Chimney top	31	98.0	

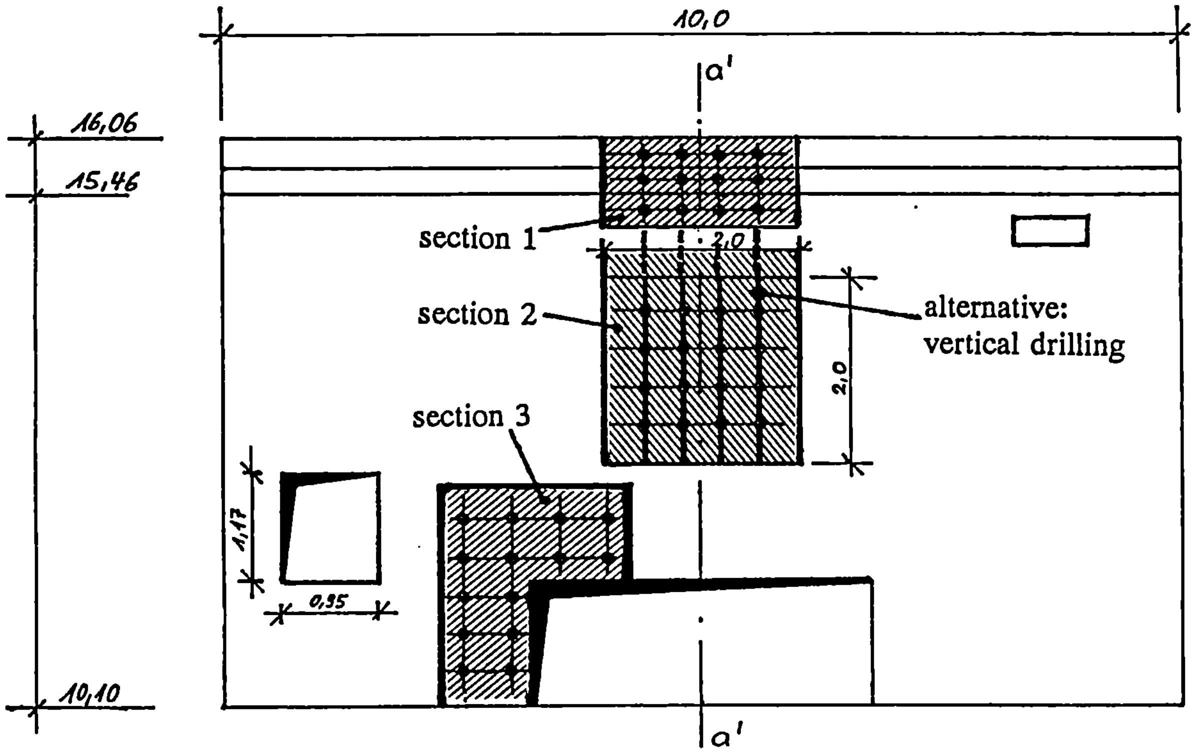


Figure 1: Shield sections for explosive dismantling tests: flattened view

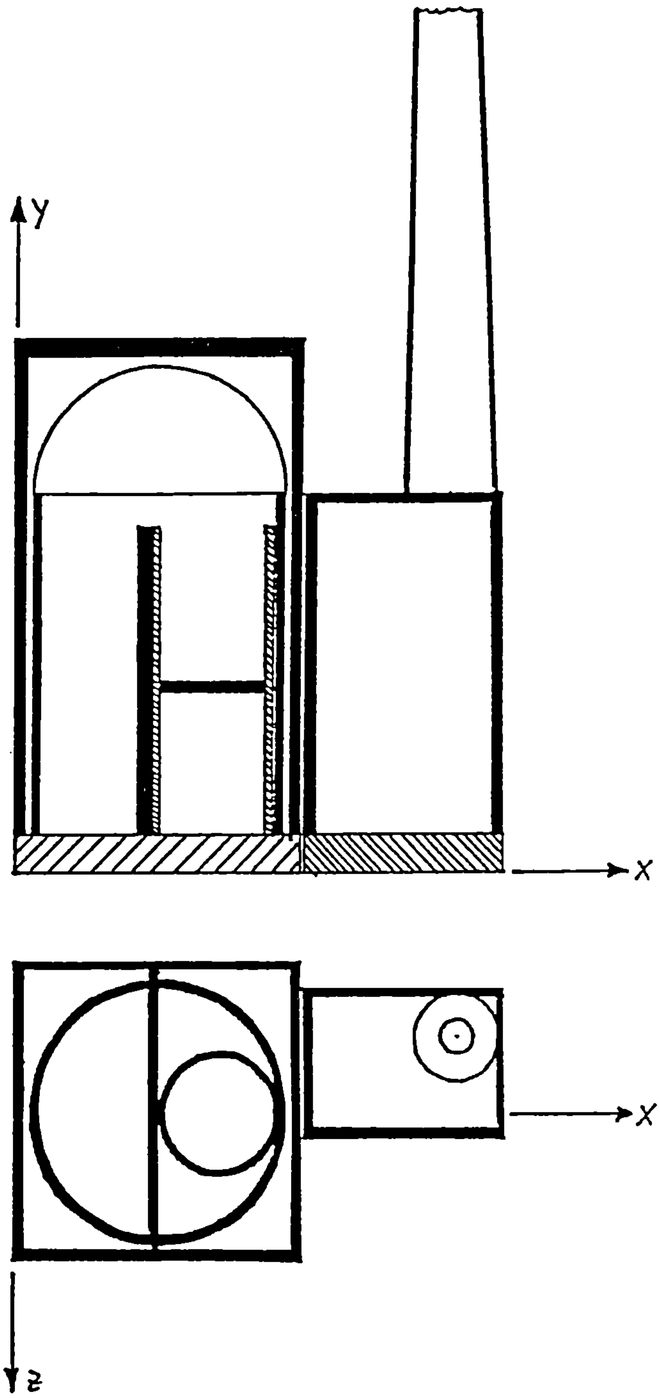


Figure 2: Simple model for computation

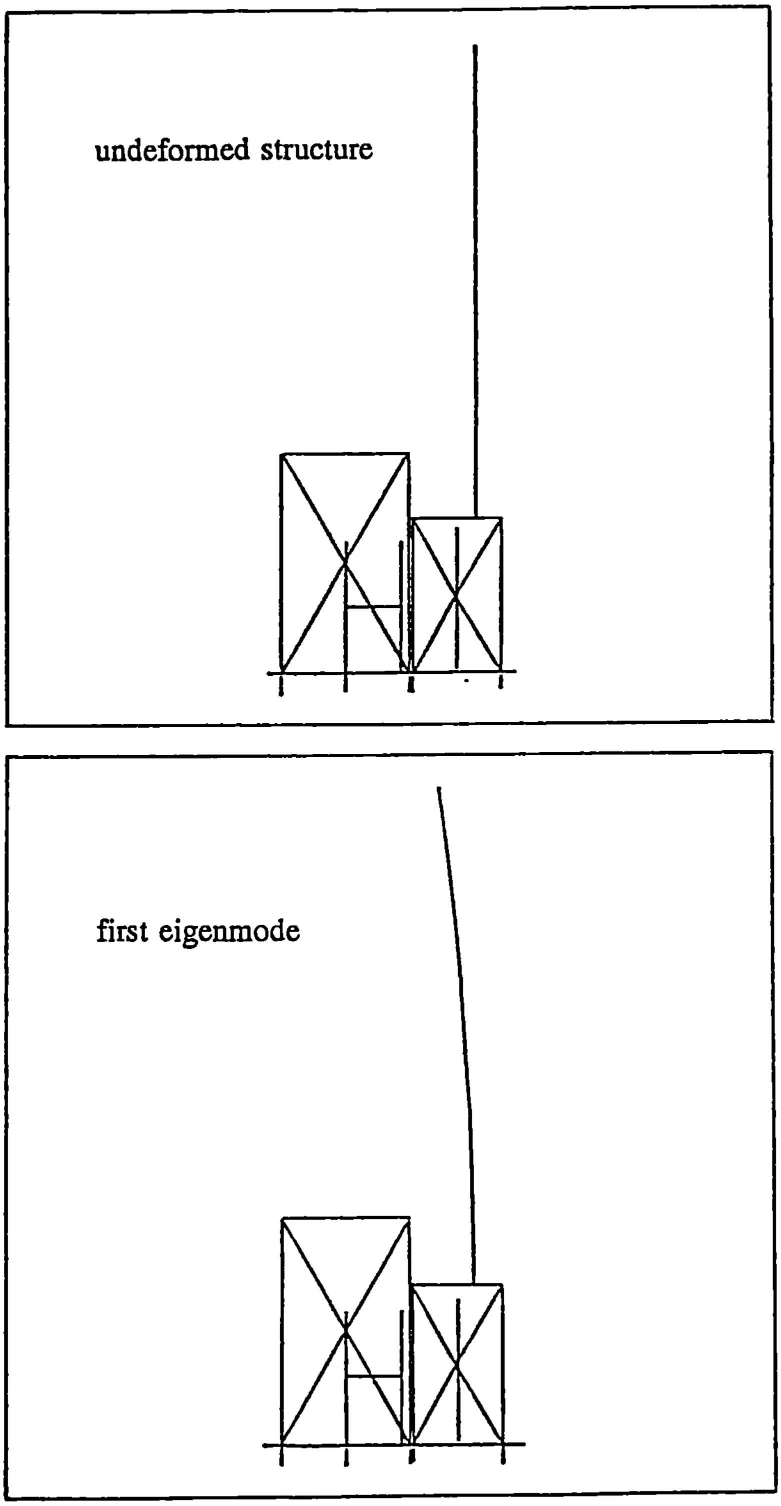


Figure 3: Computer model for reactor and auxiliary building excitation

8.14. DECOMMISSIONING OF THE B205 FUEL REPROCESSING 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. <u>Dismantling of the New Dissolver Cell</u>

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. <u>Dismantling of the Original High Active Cell</u>

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. <u>Dismantling of the Metal Cutting Cell</u>

B.5. <u>Generation of specific data</u>

C PROGRESS OF WORK AND OBTAINED RESULTS

Summary of Main Issues

a This progress report summarises the work carried out from the commencement of the contract, 10 July 1991, to 31 December 1991. The work carried out during this period covers work package B1; Preparatory Work. Generation of contract data will arise from work packages B2, B3 and B4.

b During demolition of the Medium Active Cell additional contaminated vessels, pipework and lead shield material has been uncovered which has increased the volume of alpha contaminated waste and significantly increased the demolition programme timescale.

c A design specification for remote handling equipment, maintenance facilities, containment structures and waste export equipment has been prepared and issued for competitive tender.

d To date waste metals have been decontaminated and disposed as Low Level Waste (LLW) or sent for recovery at Sellafield's own lead recovery facilities. Non-decontaminable materials have been consigned as LLW or Plutonium Contaminated Material (PCM) if above the LLW disposal criteria.

e Dose uptake levels for demolition of the Medium Active Cell have been relatively low, in the region of 1 milli-sievert/man month.

Progress and Results

Work has progressed in Work Programme B1, Preparatory Work including assessment and/or backfitting and installation of auxiliary equipment and access routes. Progress in the Work Package areas is summarised below:

B1.1 Removal and decontamination of redundant service lines, shielded liquor transfer line, fume hoods, internal wall etc.

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 monitored and despatched for re-smelting at Sellafield.

c The Medium Active Cell, Figure 1 Area C, demolition work is in progress. In-cell process equipment and support structures have been decontaminated for disposal as Low Level Waste (LLW). Lead shot shielding and other non-decontaminable equipment has been disposed of as Plutonium Contaminated Material (PCM). To date approximately $5M^3$ has been generated from the demolition operations.

Demolition of the cell walls (Wood and Perspex) and cell floor revealed two previously unidentified raffinate tanks and a number of containers of sludge and liquid located in the hollow cell plinth. The steel raffinate tanks have been sampled (100 mSv/hr beta gamma) and decontaminated to LLW criteria. Significant quantities of alpha contaminated (> 3000 cps) lead shield bricks were uncovered (approx 2.5 te). This lead shield material is being decontaminated using Pentek 603 self strip coating. Contamination levels have been reduced from > 3000 cps to < 500 cps surface contamination. Where surface contamination was not removed by Pentek 603, 4M Nitric acid was applied via an aerosol and removed by dry swabbing. In total 11 te of lead shielding material has been recovered from the cell and sent for recovery.

Dose uptake for demolition and decontamination is approximately 1 milli-sievert per man month.

d All redundant equipment has been removed from the fume hoods and the lead lining and internals have been decontaminated and removed. The fume hood superstructure and windows have been removed to permit plinth demolition.

Redundant services and the active drain have been isolated and removed. A containment tent and change area has been constructed in preparation for the demolition of the fume hoods, this work is due for completion by the end of the financial year.

B1.2 Refurbishment of the Cell Ventilation/Filtration System

a Air flow assessments within the containment tents have demonstrated the adequacy of the installed ventilation system. Additional filters have been installed before the ventilation offtake to reduce airborne contamination entering the existing building extract facility which serves the laboratory 190 complex.

B1.3 Establishment of Waste Decontamination Facilities

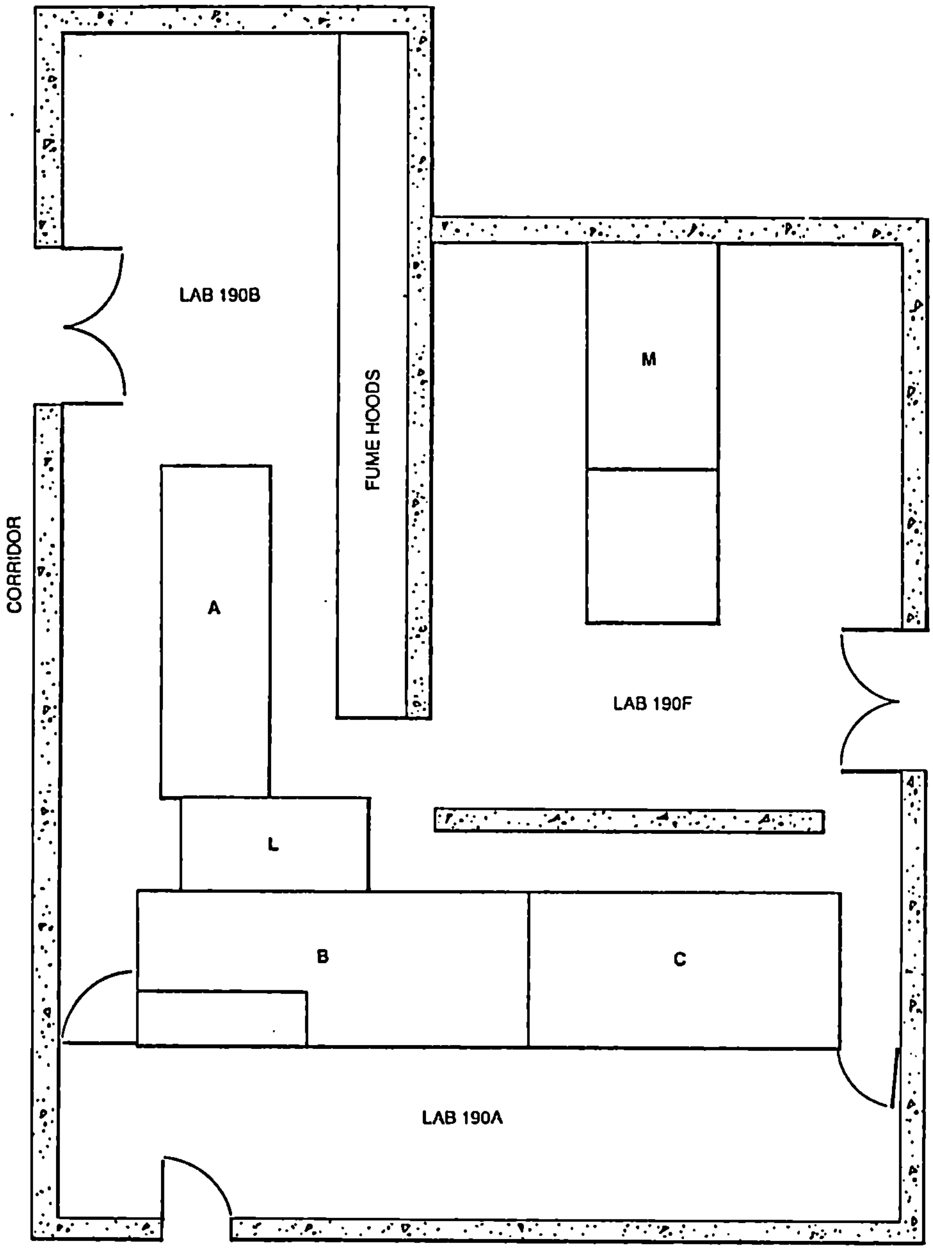
a Decontamination of alpha contaminated waste from the MA cell dismantling and demolition has been carried out either in-situ (lead shielding) or in the Sellafield site decontamination centre (stainless steel tanks and large items of equipment). An adjacent laboratory (190D) is being made available for waste decontamination from Work Packages B2-B4.

B1.4 Establishment of Waste Handling/Export Facilities

a The design and construction of a dedicated waste handling and export facility has been put out to tender and will be included in Work Package B2 as part of section B2.2.

B1.5 Installation of New lifting Beams

a A camera inspection of the installed lifting beams indicates that replacement is probably unnecessary. Load tests will be carried out upon completion of MA Cell demolition to confirm this.



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

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 moveable 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 $\mu Sv \cdot h^{-1}$ 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. Large-scale technique demonstrations on active zones (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. Aerosol characterisation (AEA)

B.5. Structural assessment (Taywood)

B.6. Theoretical assessment (Taywood)

B.7. Techniques assessment and conclusions (AEA and Taywood)

B.8. Collaboration with related programmes

B.9. Generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED

No significant work was completed in this just starting contract.

8.16. 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

In the course of 1991, two meetings (September and December) were organised by the CEC between the partners, in order to define a joint work programme and to identify a common data collection method.

It was agreed to use the commercially available ORACLE V.6 software package for data processing, offering good performance, multi-user access, network support, and good interface possibilities between various hardware systems.

Uni. Hannover, as the service organisation for this database, will coordinate actions between the three partners. The CEC will coordinate actions between both database groups.

Progress and results

1. Development of the data base structure (B.1.)

In cooperation with all partners, it was decided to implement the data base on tooling in the industrial standard software RDBMS ORACLE V.6. on a SUN SPARCSTATION. This software offers a good performance, multi-user access, network support and a maximum portability of the final application. The language that will be used is SQLFORMS V.3. (ORACLE Corp.), offers a good portability between various hardware platforms.

The concept of this database is divided into two levels, i.e. the first level contains data common to a series of cutting operations or cutting results respectively, e.g. the reference and the description of the equipment used (Fig. 1). The second level incorporates data relating directly to a single cutting operation. The main table in this level contains data which are common to all types of cutting operations, e.g. cutting speed or type of material. The other tables contain data such as e.g. gas supply, or single data which are not described very frequently, e.g. geometrical quality of kerf edges. In this second level, there are also tables concerning emissions, secondary waste and occupational doses.

The data will be structured in the database in a classifying Thesaurus system, meaning that the data will be stored in a numeral code and allowing steady completion. An example of classification table is given in Table I below.

2. Collection of data (B.2., B.3.)

In previous contracts (FI1D-0070/71), more than 2500 data sets were collected and analysed, and roughly 400 of these were incorporated into a preliminary database package at Uni. Hannover. Most of these data sets were on thermal cutting tools and water jet limited to metallic components. Besides collecting newly available data by all partners in the present contract, the database was extended to cutting tools not yet considered in the foregoing work, such as mechanical tools, and other materials than steel, mainly concrete. Further on, data on remote handling, size and weight of tools, wear parts and their lifetime, secondary waste (including consumables, wear parts of the tools and assisting material, e.g. gloves etc.), emissions, working time (cutting time plus time for mounting and dismounting of tools), occupational doses etc.

Table I - Principle of classification of data incorpored into the database (example)

1. Cutting techniques 1.1. mechanical techniques 1.2. thermal techniques 1.3. explosive techniques 1.4.	 1.2.1. plasma arc cutting 1.2.2. flame cutting 1.2.3. laser cutting 1.2.4. arc saw 1.2.5.	 1.2.3.1. CO-laser 1.2.3.2. CO_2-laser 1.2.3.3. NdYag-laser 1.2.3.4. Eximer-laser ...

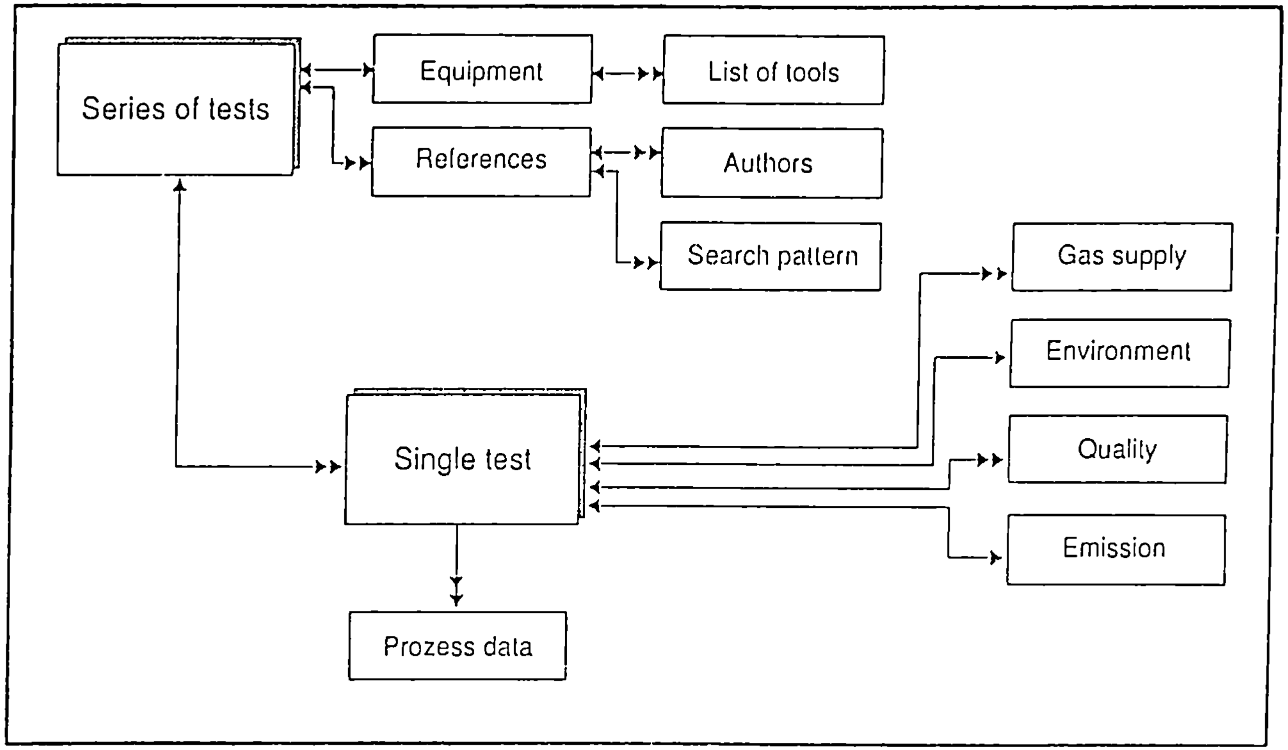


Fig. 1 Basic structure of DB

8.17. 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

In the course of 1991, two meetings (July and November) were organised by the CEC between the partners, in order to define a joint work programme and to identify a common data collection method based on the methodology developed within the FI1D-0074/0075 contracts.

It was agreed to use the commercially available ORACLE V.6 software package for data processing, offering good performance, multi-user access, network support, and good interface possibilities between various hardware systems. This software has already been adopted for the database on tooling (contracts FI2D-0056 to -0058).

A list of keywords was proposed by BNF and is being discussed. NIS started drafting a database general structure. CEA started data collection from AT-1.

NIS, as the service organisation for this database, will coordinate actions between the three partners. The CEC will coordinate actions between both database groups.

Progress and results

1. Development of the database software (B.1.)

The used database system ORACLE is common to the database TOOL to ensure compatible systems.
The database COST will work on a PC (type 486, 8 MB extended memory, 120 MB hard disk). So far, the section on general information of real decommissioning projects is programmed and filled with decommissioning background information (Figure 1).

The main part of the database, (containing decommissioning data on cost and occupational radiation exposure) will be developed in 1992, dependent on the information collected.

At present, a general concept is under development, i.e. the data will be stored in WORKING STEPS (WS), which are the basis for storing and evaluating the incoming data and in future for extrapolating decommissioning costs and occupational radiation exposure.

Each WS gives a complete view on one decommissioning task, which could be a very small task, e.g. dismantling of a valve, or a bigger task, e.g. dismantling of a bioloogical shield. There exists no fixed definition of a WS. The aim of a WS is only related to the clearness of the cost and occupational radiation exposure data. The stored data may be completed by some additional data on e.g. country-specific boundary conditions, currency changes to the ECU a.o.

All WS data will be structured in the specially developed GENERAL DECOMMISSIONING STRUCTURE (GDS), giving a clear view of the decommissioning work with a code system using figures. The principle of the GDS is shown in Figure 2. The GDS is similar to a network of the decommissioning tasks. Since contract implementation, the network is mainly unfilled.

2. Data collection (B.2.)

The data to be stored in the DATABASE COST is collected from various sources mainly the EC-contractors of the present R&D programme, including as much data from worldwide operations as available.

To implement the collection of data, three types of data sheets (Figure 3) have been identified and will be further developed. These will contain an extended data set about single decommissioning tasks, i.e. WORKING STEPS. The required information about a working step makes it possible to analyse any decommissioning task and to evaluate the data with respect to an EC-wide basis. Therefore, such data sheets are to be filled in as completely as possible.

Decommissioning background information of the different countries	Limits for radioactivity and occupational radiation exposure, safety requirements, final storage, policy
General information of real decommissioning projects	Nuclear installation, operating time, status of decommissioning, task of decommissioning
Decommissioning data on cost and occupational radiation exposure	Work load and labour, equipment, dose rate, personnel, employed waste management

Figure 1: Main parts of data in the database

COMPLETE CODE

3 1 12 2 6 41100

working step (WS)
standard working package
working area
specific project working package
decommissioning phase (if necessary)
decommissioning alternative (if necessary)

STANDARD WORKING PACKAGE

1. planning, project leading and staff
2. licensing procedure
3. operation
4. decontamination
5. dismantling
6. waste management
7. radiation protection
8. site recovery
9. research and development

WORKING STEP

4 1 100

numbering figure, if a working step is divided into several working step tasks
code for the working task
code number for the reference value

Figure 2: Principle of the general decommissioning structure

Type I:	Usable for working steps without radiation exposure, e.g. planning and licensing procedures, documentation, supervising out of the controlled area.
Type II:	All decommissioning tasks within the controlled area, e.g. dismantling, decontamination, waste management and radiation protection.
Type III:	All activities relating to storage, final disposal, reusing or recycling a material. For each working step, described in a data sheet II and being connected to a material or a waste, a data sheet III is necessary.

Figure 3: Types of data sheets

8.18. 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 RESULTS OBTAINED

No significant work was completed in this just starting contract.

8.19. 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

No significant work was completed in this just starting contract.

8.20. 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 x 10^3 to 3.7 x 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

No significant work was completed in this just starting contract.

8.21. DEVELOPMENT OF A ROBOTIC SYSTEM (TRR) 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 (TRR) 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 TRR 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 (TRR) layout and of the environmental and radiological conditions (in cooperation with ENEL)
B.1.2. Functional requirements definition of the TRR
B.1.3. TRR design at system level (Two TRR 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. TRR design and manufacturing (ANSALDO)
B.3.1. TRR 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 TRR on the mock-up (ANSALDO)
B.5. Site preparation (ANSALDO with ENEL support)
B.6. TRR operation and material transportation (ANSALDO + ENEL & SIEMP. support)
B.6.1. TRR 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 WORK AND RESULTS OBTAINED

No significant work was completed in this just starting contract.

8.22. ASSESSMENT OF DECONTAMINATION PROCEDURES FOR WWER-PWRs WITH A VIEW TO MINIMIZE THE GENERATION OF WASTE

Contractors: Siemens-KWU, KKWR, SCK/CEN Mol
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 complementary decontamination of component surfaces after an appropriate pretreatment to remove the oxide layer. The specific aim is to determine which method will permit recycling, without generating excessive amounts of secondary waste.

B.2. Decontamination of removed components: the decontamination process for separate components 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, a concept for the decommissioning of reactor components for WWER NPPs shall be developed. This concept would determine an optimised scheme for the application of the various in-situ and component-specific decontamination processes. The general objective of this study is to establish a treatment scheme which will permit recycling of non-activated components while generating a minimum of waste. A further important requirement is the minimization of operator exposure.

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

No significant work was completed in this just starting contract.

8.23. DESIGN, MANUFACTURING AND APPLICATION OF A TELEOPERATED DEPLOYMENT SYSTEM BASED ON THE "NEATER" ROBOT, TO REMOVE THE U2 DRAIN LINE, B14 WINDSCALE LABORATORY

Contractors:	AEA Technology
Contract No.:	FI2D-0068
Work Period:	November 1991 - December 1992
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 teleoperated pipe removal monitoring, and decontamination system to remove highly contaminated pipework from positions where radiation doses are high and restrict man access. The project includes the proving of the system in an inactive mock-up followed by pipe cutting (U2 drain line of B14 building of Windscale Laboratory) removal, monitoring, decontamination and disposal.

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 work has a wide application area and over 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.

AEA Technology brings a range of knowledge and experience to project and is willing to share details of arising technical innovations with other community members.

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. Design, specification and commissioning of the pipe clamping and deployment system
B.6. Service requirements and interface connection will be specified and survey services provided.
B.7. Design, specification and commissioning of the waste handling system
B.8. Design, specification and commissioning of the viewing including pan/tilt unit and controls, cables, lighting.
B.9. Design, specification and commissioning of the mock-up area, installation of complete system, drawing up, defining and agreement of test schedule.
B.10 Preparation of safety case and obtention of approval for robotic safety, active area safety, pressurised suit safety.

B.11 Site-specific activities including delivery of equipment, installation of control room, robot frame, interconnections, Commission of Full Test of All Systems, documentation for maintenance and repair purposes.

B.12 Removal, monitoring and decontamination of pipework, including waste arisings disposal and dispatch to disposal route, as well as removal of robot system and cleaning up work area.

B.13 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

No significant work was completed in this just starting contract.

8.24. 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.

C. PROGRESS OF WORK AND RESULTS OBTAINED

No significant work was completed 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 8655.

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.

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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.

B. 1989 European Conference

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C. Final Contract Reports

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Allibert, M. et al. (1987). Séparation par transport en phase vapeur des constituants d'aciers inoxydables. EUR 11296.

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EUR 14498 — The Community's research and development programme on decommissioning of nuclear installations (1989-93) — Annual progress report 1991

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