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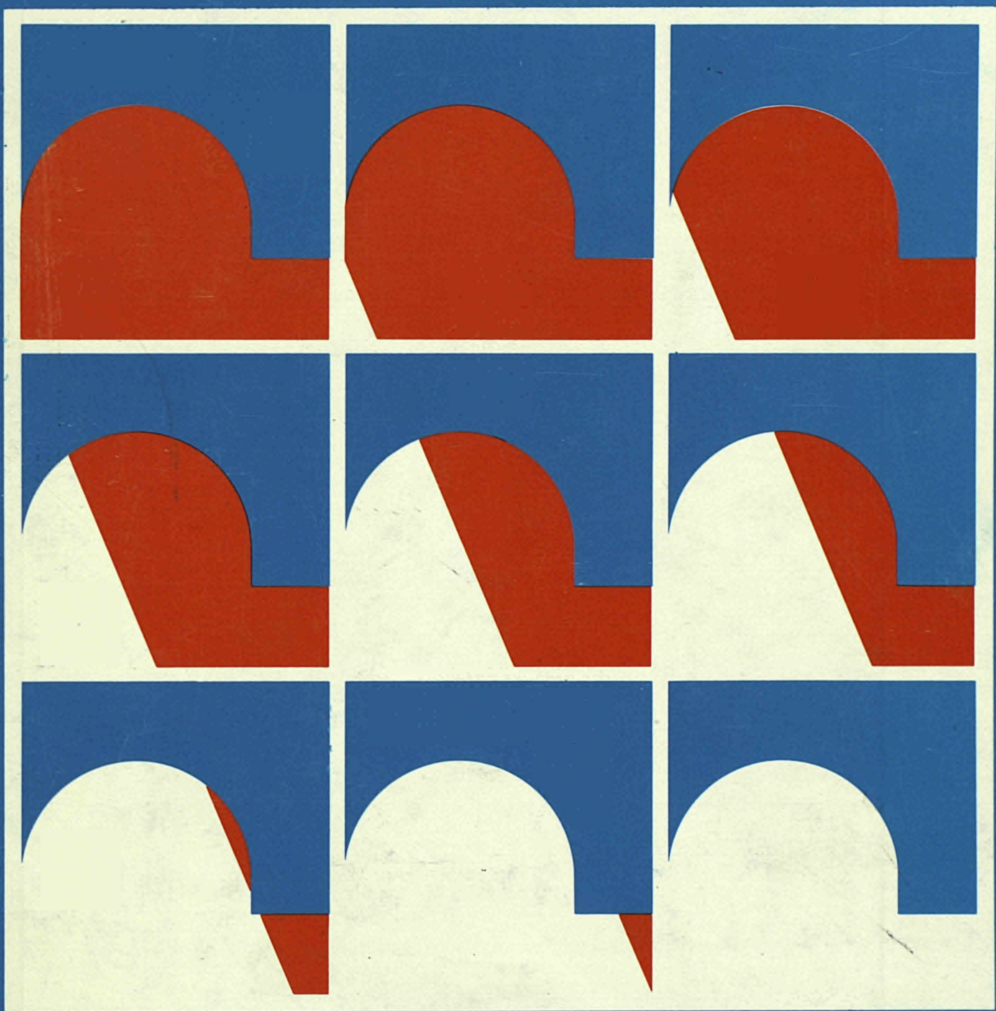
European Commission

nuclear science and technology

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

Annual progress report 1993



Report

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

Annual progress report 1993

Directorate-General
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ABSTRACT

This is the fourth annual progress report of the European Community's programme (1989-93) of research on decommissioning of nuclear installations. It shows the status of the programme on 31 December 1993.

This fourth progress report summarizes the objectives, scope and work programme of the 79 research contracts concluded, as well as the progress of work achieved and the results obtained in 1993.

FOREWORD

The following fourth Annual Progress Report summarises the activities of the European Communities R&D Programme on Decommissioning of Nuclear Installations for the year 1993. (Annual progress reports 1990, 1991 and 1992, see ref. /1/, /2/ and /3/).

This programme was adopted by the EC Council in March 1989 /4/ 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 about 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 Programme's budget for the five-year period 1989-1993 amounted to 33.8 million ECU. In 1993, further funds (5.8 million ECU) were made available to permit continuation of the most important projects during 1994 and 1995.

Work on the four pilot dismantling projects will therefore continue. Also during 1993, Part B "Identification of guiding principles" was completed and the report is due to be published in 1994.

The common action to collect data relevant to cost, occupational doses, working time and waste arisings is now a fully operational part of the programme.

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

For its compilation and editing I wish to thank my colleagues, Messrs R Bisci, K Pflugrad and R Wampach.

This is probably the last annual report of the third EU Decommissioning Programme (1989-1993). In 1994, the annual report will be replaced by the Proceedings of the Third International Conference on the Decommissioning of Nuclear Installations (26-30 September 1994, Luxembourg), which will present the final state of the programme's activities.

R SIMON
Head of the Programme

References

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

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

/3/ "The Community's research and development programme on decommissioning of nuclear installations (1989-1993)". Annual progress report 1992. EUR 15262.

/4/ 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.

/5/ Commission Communication concerning the research programme on the decommissioning of nuclear installations (1989 to 1993). Call for research proposals. OJ No. C 24, 31.01.1991, p. 8.

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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
BI	Battelle Ingenieurtechnik GmbH, Düsseldorferstraße 9, D-65760 Eschborn
BNF	British Nuclear Fuels plc, Sellafield Works B403, UK-Seascale, Cumbria CA20 1PG
BS	Brenk Systemplanung, Heinrichsallee 38, D-52062 Aachen
Bureau A+	Bureau A+, Godsweerdersingel 87, NL-6041 GK Roermond
CEA-Cad.	Commissariat à l'Energie atomique, Centre de Cadarache, B.P. N° 1, F-13108 St. Paul-lez-Durance
CEA-FAR	Commissariat à l'Energie atomique, Centre de Fontenay-aux-Roses, 60 Avenue du Général Leclerc, B.P. N° 6, F-92265 Fontenay-aux-Roses
CEA-Sac.	Commissariat à l'Energie atomique, Centre de Saclay, F-91191 Gif/Yvette
CEA-Valrhô	Commissariat à l'Energie atomique, Centre de la Vallée du Rhône, B.P. N° 171, F-30205 Bagnols/Cèze Cedex
CIEMAT	Centro de Investigaciones Energéticas Medioambientales y Tecnológicas, Avenida Complutense 22, E-28040 Madrid
COGEMA	Cie Générale des Matières nucléaires, B.P. 270, F-50107 Cherbourg
COMEX	Comex Nucléaire, 36 boulevard des Océans, F-13275 Marseille
DLR	Deutsche Forschungsanstalt für Luft- und Raumfahrt e.V., Pfaffenwaldring 38-40, D-70569 Stuttgart
ENEA	Ente per le Nuove Technologie, l'Energia e l'Ambiente, Viale Regina Margherita 125, I-00198 Roma
ENEL	Ente Nazionale per l'Energia Elettrica, Via R. Rubattino 54, I-20134 Milano
ENRESA	Empresa Nacional de Residuos Radioactivos S.A., Calle Emilio Vargas 7, E-28043 Madrid
ENSA	Equipos Nucleares S.A., Plaza del Marqués de Salamanca, E-28043 Madrid
EPC	S.A. d'Explosifs & Produits chimiques, rue de la Dynamite, F-13310 St-Martin de Crau
EWN	Energiewerke Nord GmbH, Greifswald, D-17509 Lubmin
FHGF	Fachhochschule Giessen-Friedberg, Wiesenstraße 14, D-35390 Giessen
Framatome	Framatome, Tour Fiat Cedex 16, F-92084 Paris-la-Défense
Goodwin	Goodwin Air Plasma Ltd, Kernan Drive, UK-Loughborough, Leiston LE11 0JF

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

SECTION A: RESEARCH AND DEVELOPMENT PROJECTS

1. AREA No. 1: LONG-TERM INTEGRITY OF BUILDINGS AND SYSTEMS

A. Objective

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

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

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

C. 1989-1993 Programme

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

D. Programme implementation

At the end of 1993, 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 - March 1994
Project Manager:	C C FLEISCHER, TEL Phone: 44/81/575 46 82 Fax: 44/81/575 40 44

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

B.1. TEL Work programme
B.1.1. Re-inspection of Nuclear Power Plants
B.1.2. Extension to further Nuclear Power Plants
B.1.3. Compilation of Systematic Data Base
B.1.4. Planned Maintenance System Development
B.1.5. Prestressing Options Study in collaboration with SSP
B.1.6. Monitoring Requirements for Prestressed Concrete.
B.2. SSP Work programme
B.2.1. Development of a planned maintenance system, including long-term behaviour of materials and structured components
B.2.2. Study of parameters of a prestressed concrete vessel with regard to partial destressing
B.2.3. Monitoring requirements for a prestressed concrete pressure vessel and recommendations
B.3. 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 results obtained

Summary of main issues

Testing and analyses have been completed on the results of surveys carried out at Berkeley, Bradwell, Wylfa, Trawsfynnyd, Hunterston and Dungeness nuclear power stations. Data from these surveys and those carried out in 1988 on Bradwell, Wylfa, WAGR and Hinkley Point A has been entered into a proprietary software package for use as a database. The practical issues in the development of a planned maintenance system for nuclear power stations have been considered in the light of the survey findings. A desk study of the monitoring techniques available for prestressed concrete has been completed and a review report produced.

The concrete stresses under two adverse cases of geometrical partial destressing have been calculated and assessed together with the strain behaviour for the prescribed cases of prestressing loss. The analyses resulted in stress states, for which only in the case of complete geometrical partial destressing, contain some tensile stresses. The serviceability of the PCPV is, however, not considered to be adversely affected by these tensile stresses since the vessel contains a lot of surface reinforcement. A residual prestressing of approximately 25%, in the case of the THRT-PCPV, is sufficient to keep the vessel walls free of tensile stress zones. For the development of a maintenance system, theoretical investigations have been carried out to define a strategy which is based on inspection classes and intervals. Three categories being dealt with are, stability of the buildings or the structural elements, the leak-tightness of the vessel and the durability of the materials.

Defuelling of the reactors at Berkeley and Latina Power Stations have been performed in an atmosphere of dry air following the initial cooling down under CO_2. Defuelling at Hunterston A is being carried out under a CO_2 environment. Contrary to predictions, no significant quantities of nitric acid were detected in samples of air taken from within reactor 1 at Berkeley and there was no major nitrate contamination of either graphite or steel samples removed from inside the pressure vessel. The reactor environment currently favoured for decommissioning nuclear installations is air. Calculations have also been performed which show that similar rates of formation of nitric acid are likely to occur by radiolysis of moist air during the prolonged care and maintenance phase of decommissioning for Magnox, BWR and PWR plants.

Progress and results

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

The inspection of the selected power stations was completed in 1992. The objective of the inspections has been to determine whether corrosion has been initiated and if not how long before initiation of corrosion will occur. The time to initiation (T_0) has been calculated for both carbonation of the concrete and for chloride ingress using data obtained from the 1992 surveys and from those four stations investigated in 1988. The minimum predicted times to activation have been plotted for all station. An example of typical results is presented in Figure 1.

The summary of results include:

- ❑ predictions indicate that after 150 years from construction, approximately 65% of all areas could still be passive (70% external and 60% internal)

- ❑ Predicted times to activation for internal areas ranged from 44 years to well in excess of 1000 years. Times for external areas ranged from 23 years to well in excess of 1000 years.

- ❑ Measured levels of chlorides within the concrete were low in many areas, making accurate prediction of T_0 difficult. Predictions indicate that a substantial percentage of steel is unlikely to become depassivated within 150 years.

2. Compilation of systematic database (B.1.3)

Results from the inspections undertaken in the preceding research and development CEC programme and this current programme have been entered into a commercially available software program. This software program can manipulate, sort, search and extract data using simple instruction. Data from additional surveys can be added directly into the database and manipulated as required. It should be viewed as a first stage database which can be upgraded to include maintenance requirements, monitoring intervals, costs etc.

3. Planned maintenance system development (B1.4)

The practical issues in the development of planned maintenance system for nuclear power stations have been investigated and stages in a typical system have been considered in the light of the survey findings. It is essential to first define the objectives of such a system and then to set up a database of survey results and findings in order to define the extent and interval of inspection and maintenance. Considerable advantage is to be gained by first subdividing the structures into elements at greatest risk or with the greatest impact of failure. A detailed knowledge is required of the ageing mechanisms of structures, in order to predict their longevity. Substantial research has already been completed on the performance of reinforced concrete that forms the principal structural elements. Estimates of life expectancy and cost of maintenance need to be made for the structural and building elements. The objective of the planned maintenance system must be to optimise the selection and timing of the various activities.

4. Study of parameters of a prestressed concrete vessel (B.2.2)

Using a modified creep function which takes into consideration the age of loading, mechanical analyses of an axisymetric model of the THTR-PCPV have been made, spanning a post-operational period of nearly 100 years. The concrete stresses under full prestressing, 50% prestressing of all tendons, full destressing and two adverse cases of geometrically partial destressing have been calculated and assessed. Likewise the strain behaviour over such a period has been analysed for the described cases of prestressing loss.

The analyses resulted in stress states which, only in the worst cases of geometrically partial destressing, contain tensile stresses. Since these stresses are predominantly caused by stress redistributions they are not considered to represent a problem of load carrying capacity. It can be concluded that, although the serviceability of the PCPV is not affected adversely by these tensile stresses, since the vessel contains a lot of surface reinforcement, the prestressing should not be removed completely in the case of planned reduction of prestressing. A residual prestressing of about 25% in the case of the THTR-PCPV is sufficient to keep the vessel walls free of tensile stress zones.

The findings of the theoretical investigations have consequences on the development of a maintenance system. With regard to this, investigations have been directed towards finding a strategy which is based on inspection classes and intervals.

5. Monitoring requirements for a prestressed concrete pressure vessel and recommendations (B.2.3)

The recommendations deal with three aspects of monitoring, covering the stability of the buildings or the structural elements, the leak-tightness of the vessel and the durability of the materials. The first two usually are mainly dependent on prestressing, the latter one depends on material oriented facts.

The aim of the long-term controlled durability of the concrete at the THRT-PCPV should be controlled safety during the whole lifetime after decommissioning. Some representative areas could be chosen and the selected areas should be easily accessible. The areas should be chosen in such a way that they represent the influences of the environmental conditions. The actual condition of these areas should be recorded by data on concrete cover, carbonation depth and chloride penetration.

Because of the low stressing of the sections, the long-term stability of the PCPV is not put at risk. This will remain valid even when the prestressing failed regardless of the failure mode. From this very important finding it could be deduced that monitoring the prestressing state of the PCPV is not necessary. On the basis of earlier investigations of the THTR-PCPV it can be assumed that the leak-tightness of the liner will remain if the structural integrity of the vessel is guaranteed. This assumes that the liner material itself will work without damage over such a long lifetime. This fact should not remain unnoticed considering the long-term leak-tightness of the PCPV or any other vessel.

6. Prediction of nitric acid concentration (B.3.2)

Rates of formation of nitric acid when decommissioning Magnox, PWR and BWR plant, four years post shutdown have been estimated and comparative values of 113, 37 and 100ml nitric acid per hour respectively have been obtained.

It has been calculated that when air was admitted into the Berkeley reactors during the first year following the "end of generation" and defuelling commenced, nitric acid will have been formed at the rate of a few kilograms per hour. By the end of the first year when the gamma activity had decreased, the rate would have fallen to a few grams per hour and after ten years the rate of reaction will be down to 1 gram per hour.

Defuelling of the reactors at Berkeley Nuclear Power Station and Latina PS has been completed some time ago. Essentially this was performed in an atmosphere of dry air following the initial cooling down under CO_2. The air environment at Latina had a controlled moisture level of 100 to 400 ppm which was then relaxed to 1000 to 1200 ppm on completion of defuelling; the latter was consistent with the moisture content of the air in the Berkeley reactors. Defuelling at Hunterston A continues, but in this plant a CO_2 environment has been maintained virtually throughout the period from final shutdown.

Although predictions relating to the formation of nitric acid because of the radiolysis of moist air during defuelling indicated that significant quantities of nitric acid should have built up in the reactors, evidence from measurements at Berkeley NPS suggests levels much lower than expected. The total concentration of nitrate ions measured in samples of air drawn from the reactor atmosphere at Berkeley NPS was equivalent to 0.26 wpm nitric acid. Also, the weights of nitrate/nitrite in liquid extracts produced from specimens of irradiated graphite removed from a sample channel in reactor 1, were similar to the level measured from a control specimen of inactive "virgin" graphite. Similarly, the nitrate/nitrite levels measured on the surface of steel samples removed from the lower dome and upper dome regions inside the pressure vessel of reactor 1 although measurable, are not significantly different from values recorded on internal steel surfaces during the construction of the Advanced Gas Reactors at Heysham and Torness.

7. Corrosion assessment (B.3.4., B.3.5)

There was no evidence of serious corrosion within the Berkeley plant when limited video inspection was carried out and this is consistent with the results of limited inspection undertaken at Latina. Nevertheless, the proposed corrosion and atmospheric monitoring scheme has been maintained at Berkeley PS and a similar approach is being considered for both Latina and Hunterston A.

In order to monitor general corrosion and possible stress corrosion cracking, samples have been positioned above and below the core in the reactor pressure vessel(s) at Berkeley. Temperature and gas sampling points are also positioned at five different vertical positions in the reactor vessel. These are monitored from above an empty charge standpipe and fuel channel using instrumentation incorporating a six-way valve, connected to a data logger. Similarly the temperature and atmosphere inside the lower gas duct and within insulation surrounding the duct are being recorded.

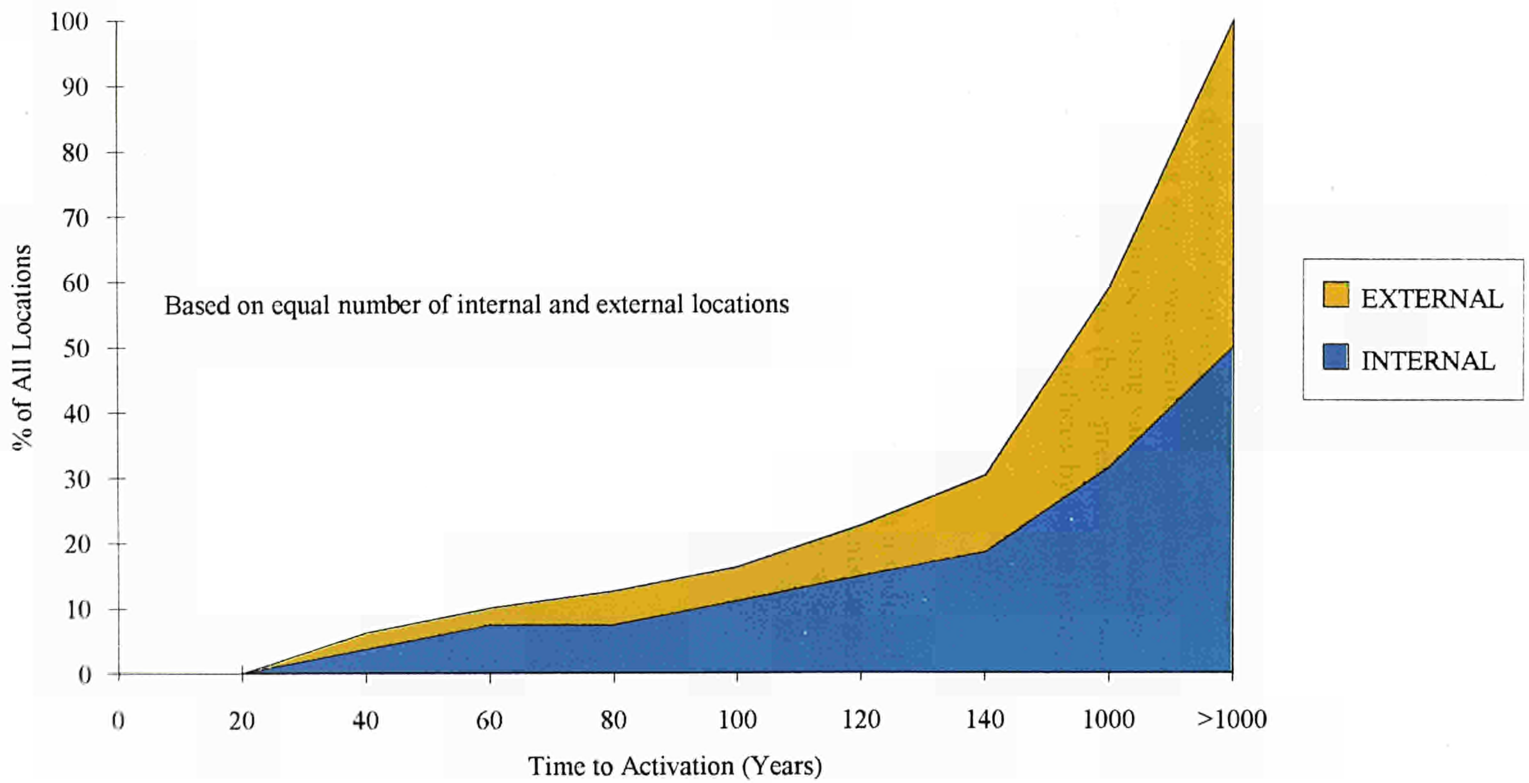

FIGURE 1 - MINIMUM PREDICTED TIME TO DEPASSIVATION BY CARBONATION - EIGHT STATIONS [To (kmax min cover)]

2. AREA No. 2: DECONTAMINATION FOR DECOMMISSIONING PURPOSES

A. Objective

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

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

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

C. 1989-1993 Programme

Research in Area No. 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 1993, the six research contracts relating to Area No. 2 were completed.

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 - October 1993	
Project Manager:	F BREGANI	
	Phone: 39/2/7224 30 46	Fax: 39/2/7224 39 15 (or 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; contract FI1D-0023, report EUR 13255).

B. WORK PROGRAMME

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

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

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

B.2.2. Decontamination tests on contaminated specimens.

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

B.2.4. Data analysis.

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

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

B.3.2. Initial radioactivity characterisation.

B.3.3. Process design and configuration.

B.3.4. Decontamination.

B.3.5. Dismantling and final radioactivity measurements.

B.3.6. Evaluation of secondary wastes.

C. Progress of work and results obtained

Summary of main issues

The research activity of this year has been focused on the decontamination and dismantling of a carbon steel line at the Garigliano BWR power plant (B3) in order to demonstrate the effectiveness of the decontamination process under study. More in detail the selected line is a part of the by-pass at the inlet of the n. °4 preheater which has operated at 150°C and 1.7 MPa for 86000 EFPH. The line includes complex components such as valves, elbows, "T" joints. The hold-up volume is 112 l and the contaminated surface 3.3 m^2. The surface activity is mainly due to Co-60 which ranges from 1.9 to 10.9 Bq/cm^2.
For this decontamination a formic acid 15% + hydrofluoric acid 3% solution has been employed, while the ultrasound (UT) transducers have been applied only on selected components, in order to allow a direct comparison with the action of chemicals and an evaluation of the effects of the ultrasounds. Radiometric measurements indicate that, where UT are applied, the residual activity is a factor 4 to 10 lower than on the components where only chemicals are active; moreover, where UT are applied, the residual activity is always less than 0.3 Bq/cm^2.

Progress and results

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.3.1)

In order to perform the decontamination test the following actions have been implemented:

- removal of the thermal insulation;
- determination of the radius of curvature of pipes and valves for UT transducer application;
- auxiliary loop connection to the test system and UT trasducer installation on the components;
- utilities installation (industrial water, electric power, drains, etc.);
- hydraulic test of the system and instrumentation check.

Due to the hazardous characteristics of the thermal insulation material (which contains asbestos) and to the contemporary presence of a radioactive field, the removal has been performed with special care by a specialized firm using glove bags. The work team was composed by 4 workers and removal operations have been completed in 6 hours. The volume of material removed is 120 dm^3, approximately. Measured dose rates in the working area range between 1.8 and 2.2 uS/h. The total dose absorbed by the personnel is 64 uS*man.

2. Initial radioactivity characterization (B.3.2)

A piece of pipe has been cut downstream the line to be decontaminated in order to obtain 11 square samples (9 cm^2 area) for the initial radiometric characterization. The surface activity ranged from 1.9 to 10.9 Bq/cm^2 for Co-60 and from <0.001 to 0.092 Bq/cm^2 for Cs-137. The corresponding mean values (standard deviation) are: 5.30 (2.79) and 0.039 (0.026), respectively.

3. Process design and configuration (B.3.3)

The line selected for the decontamination is a part of the by-pass line at the inlet of

the n. 4 preheater which has been operated at 150°C and 1.7 MPa for 86000 EFPH (fig. 1). The line include complex components such as valves, elbows, "T" joints; it has an hold-up volume of 112 l and an internal contaminated surface of 3.3 m^2, approximately.
Since all the components are in carbon steel or low-alloyed steels, the chemical solution used in the test is HCOOH 15% + HF 3.0% optimized in the laboratory tests.
In order to allow a specific evaluation of the synergic action of the ultrasounds, the system can be divided in two zones (fig. 2): the zone "A" where 11 transducers are placed at selected points and zone "B" where no transducers have been applied.
The transducers are of the Langevine type and the UT generator is capable to supply 3300 W at a frequency ranging between 20 and 40 kHz.
Chemicals recirculation and process monitoring is insured through the auxiliary loop as shown in fig. 3. Two carbon steel sample pipes (C-1 and C-2) cut downstream elbow 3 are inserted in this loop in order to control process effectiveness.

4. <u>Decontamination</u> (B.3.4)
The decontamination has been performed according to the following operating procedure:
phase 1: pre-washing (industrial water + UT); duration 30 min.
phase 2: decontamination (flowing chemicals + UT); duration 2.5 hours
phase 3: post-washing(industrial water + UT); duration 30 min.
System draining and drying with air follows.
During all the test phases the UT frequency has been set at 20-20.5 kHz, while the supplied power has been 24 W/cm^2. The following on-line measurements have been performed:

- Co-60 activity on pipe C-1 and C-2 by a NaI scintillator (fig. 4);
- temperature of the pipes at selected locations;
- ultrasound frequency and power.

5. <u>Dismantling and final radioactivity measurements</u> (B.3.5)
At the end of the test the sample pipes C-1 and C-2 have been removed from the auxiliary loop and 3*3 cm samples have been cut in order to allow a statistical analysis of the radiometric measurements. Similarly both zone "A" and "B" components have been cut from the system to perform further radiometric measurements. Results of the residual activity distribution on pipe C-1 and C-2 are reported in fig. 5, while all the residual activity measurements are summarized in fig. 6. It can be noted (fig. 5) that the use of UT determine a more effective and uniform removal of the radioactivity; a comparison of the mean values by the "t" test statistic indicate that the C-2 residual activity is significantly lower than the one of pipe C-1, with a probability error less than the 0.5%.
All the radiometric measurements indicate that, where UT are applied, the residual activity is a factor 4 to 10 lower than on the components where only chemicals are active (fig. 6); moreover, where UT are applied, the residual activity is always less than 0.3 Bq/cm^2.
Finally, data on the personnel absorbed doses during all the decontamination phases are summarized in table I.

6. <u>Evaluation of the secondary wastes</u> (B.3.6)
The spent solution arising from decontamination has been stored in a waste tank where precipitation has been obtained raising the solution pH up to 8 by Na(OH) addition. After precipitation the separated sludge has been heated in a hoven at 85°C for two hours in order to obtain the sludge dewatering. Data on spent solution treatment are summarized in table II.

Table I: Personnel absorbed doses during all the decontamination phases.

OPERATION	PEOPLE n°	DOSE RATE uSv/h	WORKING HOURS (TOTAL)	ABS. DOSE uSv*MAN
THERMAL INSULATION REMOVAL	4	2	32	64
UTILITIES INSTALLATION	6	2	180	360
DECONTAMINATION	4	2	48	96
SAMPLE CUT AND MEAS.	4	2	140	280
WASTE TREATMENT	4	2	40	80
TOTAL			440	880

Table II: Summary of the data on the waste treatment.

SPENT SOLUTION		
VOLUME	l	175
ACTIVITY	Bq/cm3	0.17
Fe (dissolved)	g/l	4.25
AFTER PRECIPITATION (*)		
SOLUTION ACTIVITY	Bq/cm3	0.09
SLUDGE WEIGHT (**)	g	1218
SLUDGE ACTIVITY	Bq/g	5.73

(*) NaOH up to pH 8

(**) After 2 h in a hoven at 85 °C

Fig. 1 View of the system selected for the field test at the Garigliano BWR plant

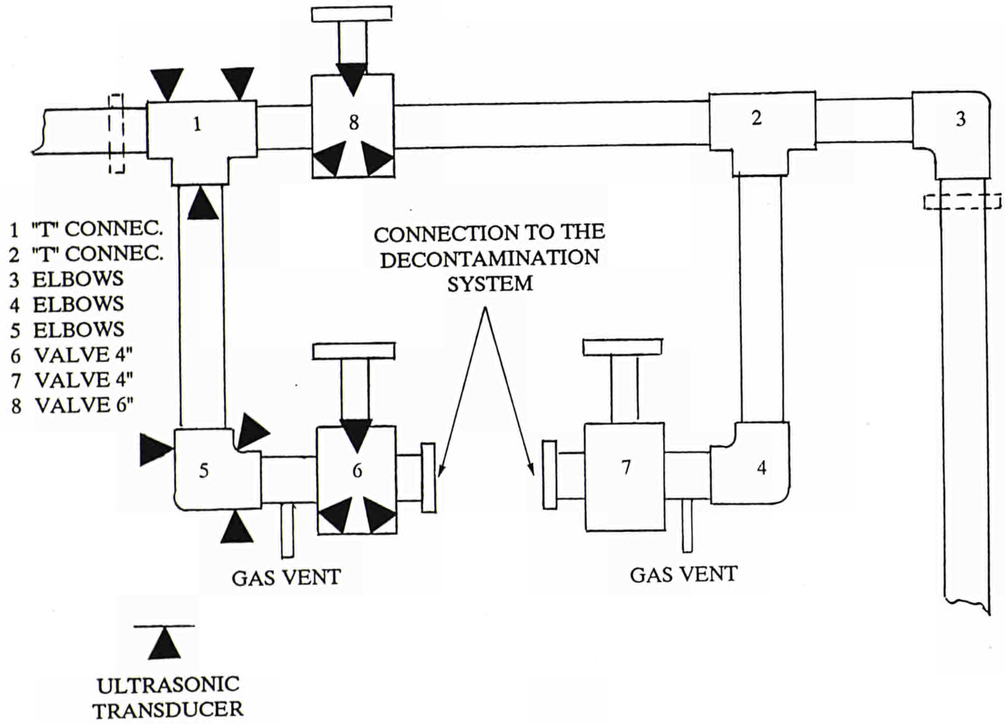

Fig. 2 Scheme of the system selected for the field test at the Garigliano BWR plant

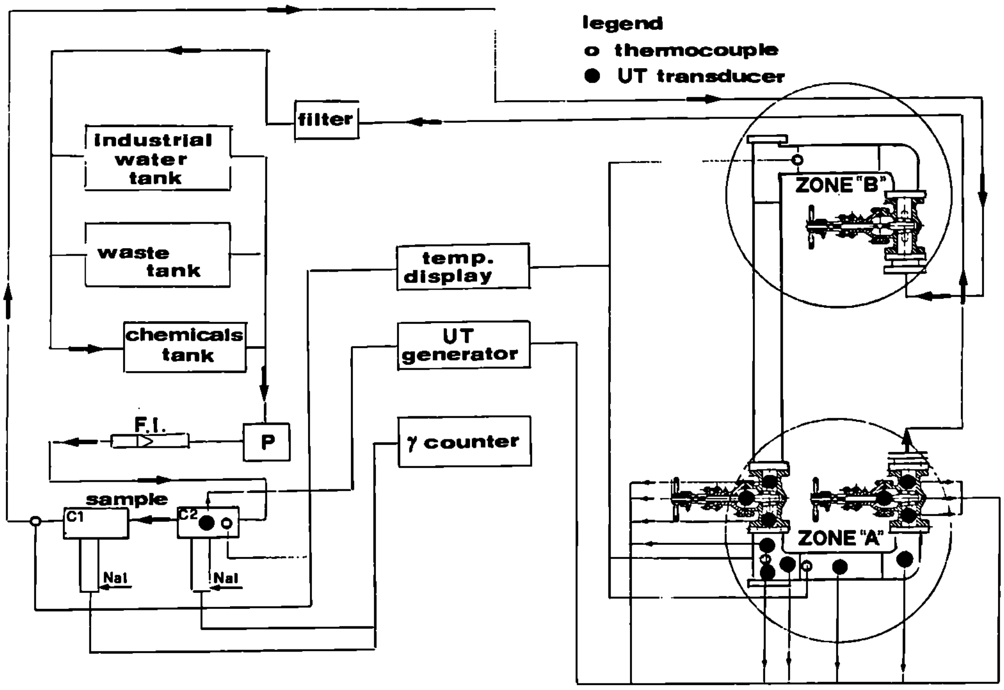

Fig. 3 Operating scheme of the decontamination test at the Garigliano BWR plant

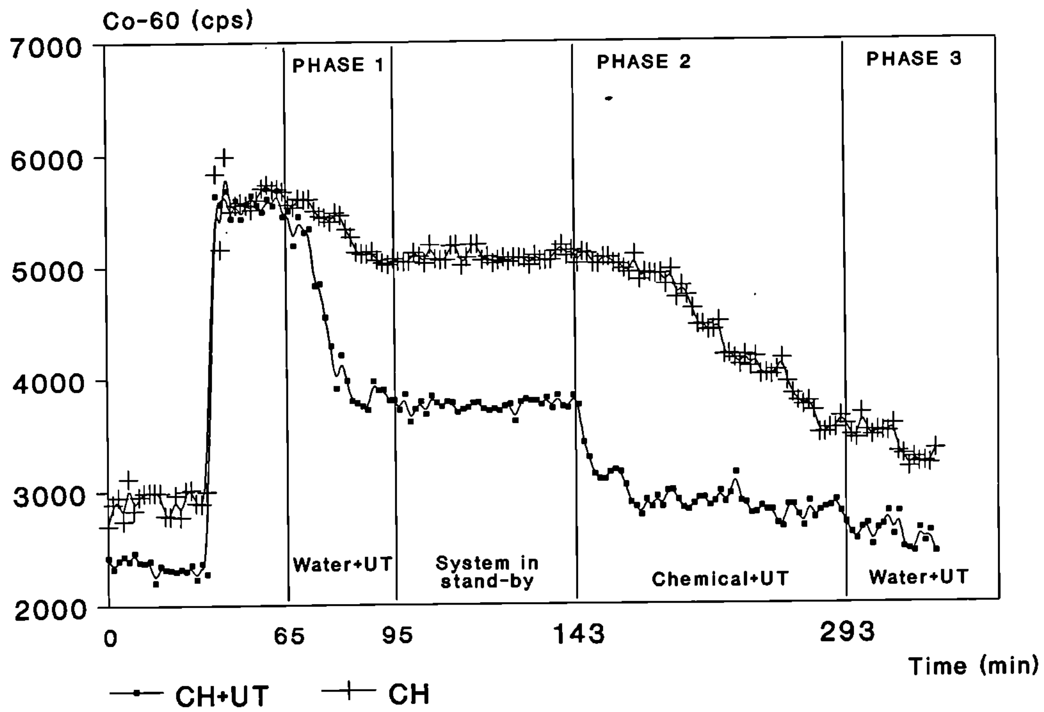

Fig. 4 On-line monitoring of the the activity on pipe C-1 and C-2 during the test

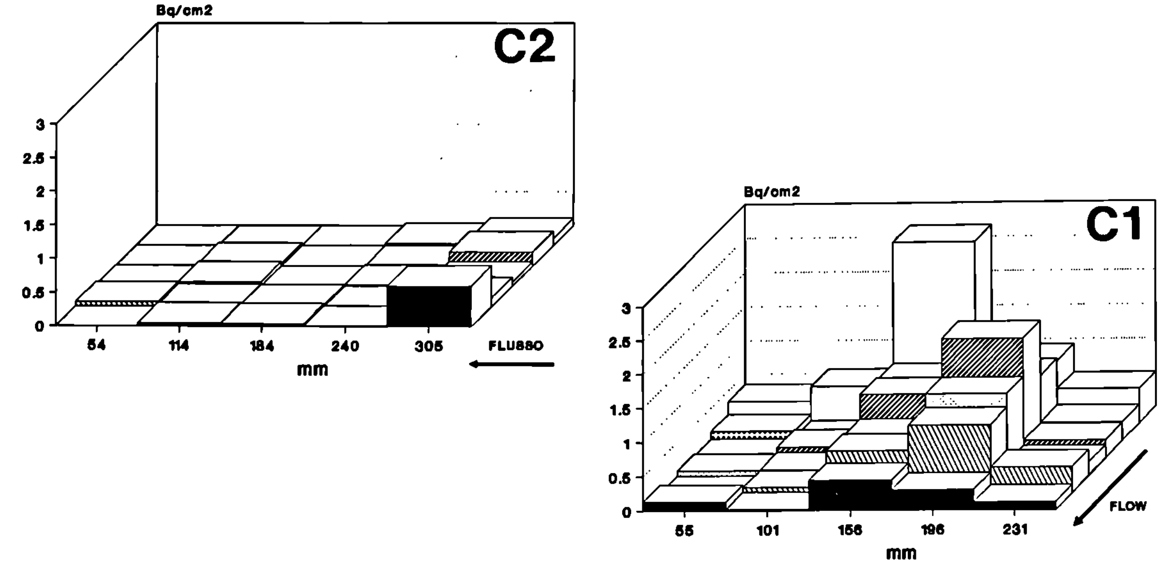

Fig. 5 Residual activity distribution on pipe C-1 and C-2.

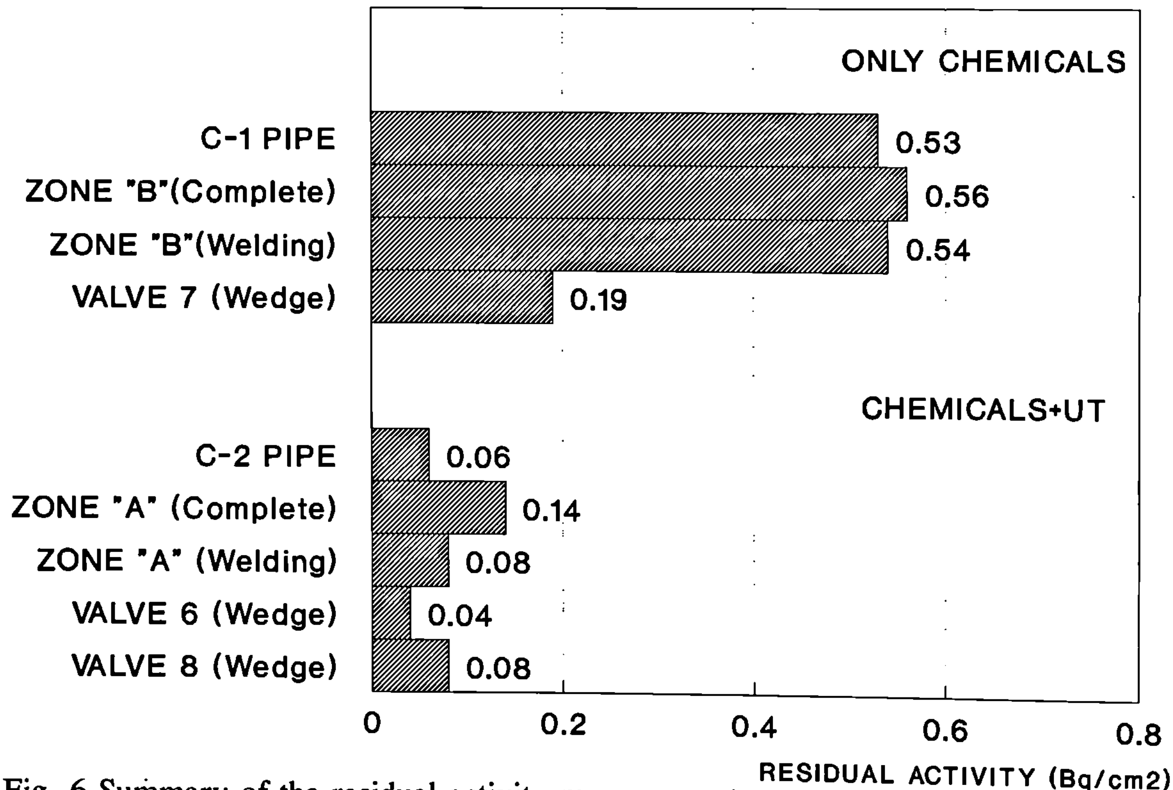

Fig. 6 Summary of the residual activity measurements.

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: 49/6221/94 12 50 Fax: 49/6221/94 17 07

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

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

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1992. The final report is being prepared for publication.

2.3. MICROWAVE SYSTEM TO SCARIFY CONCRETE SURFACES

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

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

B.1. Design and construction of a prototype microwave system
B.2. Design and construction of the support structure.
B.3. Trial operations on a non-radioactive concrete wall.
B.4. Testing of the prototype on a radioactively contaminated concrete surface.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of the main issues

The microwave system, designed and manufactured in 1991, was subjected, after tests performed in 1992, to several modifications in order to improve the system: the previous rigid wave guides were replaced with flexible ones and the adherence of the suction bell's skirt to the surface during operation was increased. In 1993, tests were resumed to establish the performance of the upgraded system and to determine the temperature distribution in the irradiated concrete and the influence of the concrete's characteristics on the scarification efficiency.

The tests on the radioactively contaminated surfaces were carried out eventually.

Progress and results

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

The main modification consisted (see Fig. 1) of three 6 m flexible wave guides, which replace the previous rigid ones connecting the microwave generators to the irradiation heads.

Consequently, the box containing the magnetrons is not fixed to the irradiation heads and can be positioned independently, far from the suction bell containing the irradiation heads, and also outside the radioactively contaminated area if desired; moreover, the mobility of the suction bell is greatly enhanced. The flexible wave guides have a rectangular section (95 mm x 45 mm), a minimum bending radius of 300 mm and a loss in the transmitted power that does not exceed 0.085 dB per m.

Another modification implemented to improve the transfer of the debris to the vacuum system envisaged dividing the suction bell's skirt in vertical mobile sectors kept in contact by means of springs with the possibly irregular concrete surface.

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

A special mobile table was constructed, connected to the existing carriage of the support structure. This table, to which the suction bell with the irradiation heads is fixed, can be positioned transversally with respect to the direction of the carriage's movement; the lateral excursion obtainable is 600 mm, and allows the production of three contiguous furrows without moving the support structure. Employing the table, also, the irradiation heads and the bell can be rotated by 90°, to test the application of microwaves with the heads in series instead of in parallel. Furthermore, the table can be moved vertically to attain irradiation head stand-offs up to 100 mm. Lastly, an indicator for the stand-off is foreseen.

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

All tests on the microwave system were carried out at ENEA's Trisaia Research Centre; those on non-contaminated concrete were performed in the Technological Hall.

Efficiency of the flexible wave guides - Fifty tests were performed with the flexible wave guides; an examination of the data (s. Table I) showed no significant difference on the scarification efficiency due to the flexible wave guides. As Figure 2 shows, the furrows obtained were continuous and well-defined, their area and depth did not change and also the quantity and the removal rate of the debris were unvaried, reaching respectively 9 kg and 6 cm^3/s. Slight losses of microwaves and correspondent increases in temperature were found in localized spots along the wave guides; these spots depended mainly upon the layout of the wave guides and not on the microwave power being transferred.

Characteristics of the concrete. The concrete characteristics were examined (density, water content and water/aggregate ratio, porosity, permeability to air, compression strength, superficial hardness, etc.) to determine their influence on the results of the scarification. Other parameters considered were the size distribution of the debris particles and the modification of the cement structure after microwave irradiation.

For this purpose, in addition to the two large slabs (3 m x 2.5 m x 0.17 m) having the same characteristics (as regards composition, additives, rebars etc.) of the concrete used in the building of the Italian power stations, several small slabs (0.8 m x 1 m x 0.2 m) were built, using different types of cement and varying amounts of water, sand and aggregates.

The analyses showed that the main factors affecting scarification are: the pore dimensions and the evaporable water in the cement. Small pores favour the shattering of the cement, probably diminishing the resistance of the cement structure to the sudden pressure rise due to the irradiated water vaporizing within the pores.

The sclerometric (superficial) hardness which is also indicative of the pore dimensions is an important parameter to consider in evaluating the liability to scarification of cement slabs.

Evaporable water in the cement includes that bonded to the cement as hydrogel and the free water present in closed and open pores; this water, measurable by the loss of weight after 24h in an oven at 105°C, must be above a certain limit (about 2%) to allow an efficient scarification.

As regards the other parameters considered, the debris particles do not present significant differences in size distribution for various compositions of the irradiated material; and no change appears in the cement structure measured by means of X-rays before and after scarification.

Temperature measurements - The purpose is to determine the temperature distribution on the surface and inside the slabs during scarification. For the surface measurements, an infrared videothermographic camera was used; for the temperature inside the slabs, thermocouples were positioned around the slabs at various depths (3.5, 7, 10.5 cm). Data was collected at predetermined time intervals, memorized on a floppy disk and reproduced separately in graphical form.

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

The system was tested on radioactively contaminated surfaces in the Hot Cell for Mechanical Operations, named "*Corridor*", of the ITREC Pilot Reprocessing Plant at the Trisaia Centre with satisfactory results both as regards the removal of the contaminated concrete and the containment of radioactivity.

Figure 3 shows a furrow resulting from the scarification in the "*Corridor*".

Table II presents the main data relevant to the activity performed in the radioactive area.

In conclusion, it may be stated that the microwave scarification system is a reliable apparatus capable of continuous scarification rates, efficient containment and removal of debris.

Nevertheless, the system in its present state could be further developed to improve its flexibility and facility of operation.

TABLE I: COLD TESTS ON CONCRETE SLABS

test N° of	Day	Slab	Humidity %	Power kW	Stand off mm	Advancement rate mm/sec	Reflected Power Kw	Energy leakage cm 0-50-100 (slab) mW/cmq	Energy leakage cm 0-50-100 (guides) mW/cmq	Temperature flexible wave guides (n°1-2-3) °C	Dimension of the furrows l x l x h cm	Debris Kg	N° of explosions	Note
121	12	OPC 325 0,6*	2,80	16	15	1,68	4,5			54-88-169	75x21x3	6,100	63	photo 13-14-15/13- t1:1' - stop:7'50"
122	12	325 F/B	1,20	16	15	1,68	3,5			62-101-231	90x21x3	9,200	30	t1:28" - stop:10'04"-
123	12	325 F/B	1,20	16	15+30	1,68	4,4			62-101-231	92x23x6,5	3,200	9	over test 122 - t1:3' - stop:11'35"
124a	12	OPC 325 0,6	2,80	16	15	0	3						2	t1:0 - stop:1'03" - test for temperature measurement
124b	12	OPC 325 0,6	2,80	16	15	0	3,5						5	over test 124a - t1:0 - stop:1'30" - test temp measurement.
124	12	OPC 325 0,6	2,80	16	15	1,68	4					4,800		over test 124b e 124a - t1:20" - stop:9'45"
125a	12	OPC 425 0,6	2,60	16	15	0	4,5				8x16x1		3	t1:0 - stop:40"
125b	12	OPC 425 0,6	2,60	16	15	0	4,8				9x18x2		2	photo 19-20/13 - over test 125a - t1:0 - stop: 2'30"
125	12	OPC 425 0,6	2,60	16	15	1,68	4			66-120-167	64x23x3x	4,900	60	photo 21/13 - over test125b e 125a - t1:35" - stop: 6'
126a	12	OPC 425 S*	2,90	16	15	0	3,5	100-20-1	60-1,8-0,2	55-66-61			4	t1: 0 - stop: 1'24"
126	12	OPC 425 S	2,90	16	15	1,68	4	100-20-1	20-2-0,6	·38-55-175	60x20x3	6,000	6	photo 23-24/13 -over test 126a- t1:40" - stop:6'58"
127a	13	OPC 525 0,6	3,20	16	15	0	3,8	100-18-1	60-2-0,2	68-45-130			3	photo 25-26/13 - t1:0 - stop:2'30"
127	13	OPC 525 0,6	3,20	16	15	1,68	4	100-18-1	60-2-0,2	35-50-135	62x22x2,5	5,800	45	over test 127a - t1:2' - stop:8'20"
128	13	OPC 525 0,6	3,20	16	15	1,68	4	100-18-1	25-2-0,8		62x22x2	4,400	35	photo 27-28/13 - t1:40" - stop:8'40"
129	13	OPC 425 S	2,90	16	15	1,68	3,5		20-1-0,7		64x17x2,5	4,900		photo 28/13 - t1:20"- stop:6'54"
130a	13	OPC 425 0,8*	2,80	16	15	1,68								
130	14	OPC 425 0,8	2,80	16	15	1,68		8-0,2-0,1	65-2-0,3	50-70-205				over test 130a - t1: - stop:
131	14	OPC 425 0,8	2,80	16	15	1,68	3,5		80- -0,2	63-80-250	66x18x2,5	3,250	49	t1:45" - stop:7'
132	14	OPC 425 0,8	2,80	16	15+2,5	1,68	5	60-3-0,4	80-1-0,2	50-45-61	70x18x5,5	1,400	45	photo 29-30/13 - over test 131-t1: 36" - stop: 6'50"
133a	14	OPC 425 0,4*	2,10	16	15	0	3,5						2	t1: 0 - stop:40'
133	14	OPC 425 0,4	2,10	16	15	1,68	3,7	60-2-0,4	80-3-0,6		65x18x2	3,000	51	photo 32/13 - t1: 30" -;stop:7' discontinuous farrow
134	14	CA-S	0,40	16	15	1,68	3,5						0	t1: 2' - stop: 7'
135	14	425A 0,4*	0,80	16	15	1,68	3,7			50-90-315				t1: 3' - stop:11'

LEGEND : - the data of the reflected power, of the energy leakage and of the temperature is the maximum one found

- t1 = time elapsed between carriage onset and magnetron start-up
- stop = time elapsed from magnetron start-up

* Initial water/cement ratio

Table II: Radioactive area scarification results

Tests N°	Humidity %	Power kW	Stand off mm	Advancement Rate mm/s	Reflected Power kw	Dimensions of the furrow l x l x h cm	Weight kg	Activity before Bq/g	Activity after Bq/g	Decontamination factor* %	Activity in drums (Cs^{137}) 10^3Bq
142	1,3	15,2	15	1,68	3,3	60x20x1,5	2,340	2,43	0,14	94	187,2
143	1,3	16,2	15	1,26	3,5	25x20x1,5	2,058	3,18	0,22	93	191,4
144	1,3	16,2	15	1,68	2,6	40x20x2	2,528	119,4	6,1	94,8	252,8

* determined with respect to activity before scarification

Figure 1 : Flexible wave guides

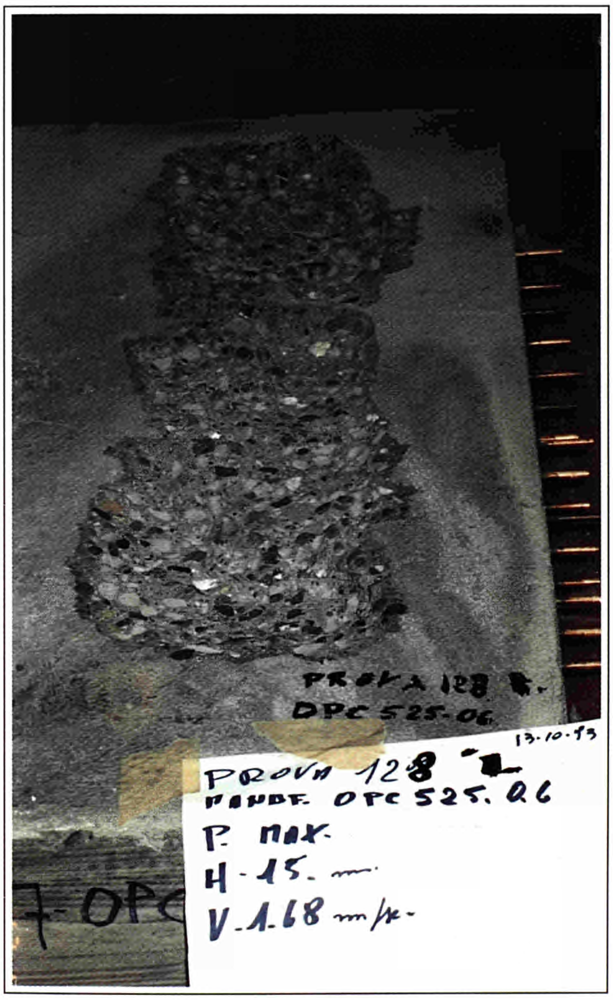

Figure 2 : Test 128

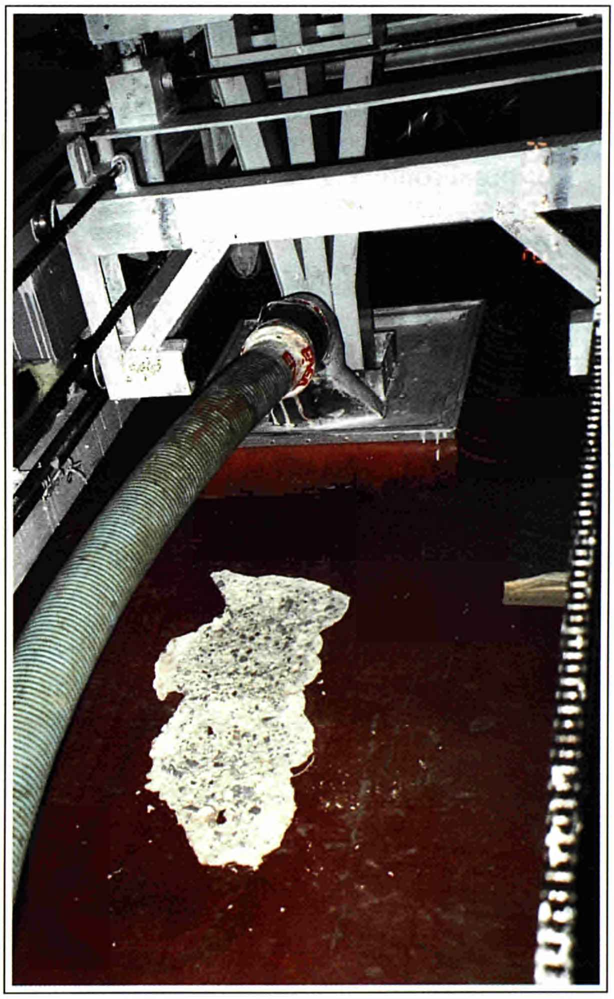

Figure 3 : "Corridor" scarification

2.4. DECONTAMINATION OF LARGE-VOLUME NUCLEAR COMPONENTS USING FOAMS

Contractors: CEA-Cad, AEA Winf
Contract No.: FI2D-0035
Work Period: October 1990 - December 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 (CEA)
B.2. Foam production and development of a circulation system (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 (CEA)
B.6. Industrial application: decontamination of a 1000 m^2 graphite-gas cooler from the decommissioning site of G2/G3 (CEA)

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

Feasibility tests were conducted on a pilot installation at Cadarache, proving that the foam process could be implemented on an industrial scale. The operating conditions suited to the decontamination of the Winfrith deaerator were established. Nevertheless, the annoying withdrawal of the AEA partners obliged CEA-Cad to find another suitable nuclear component to demonstrate the effectiveness of the foam decontamination technique on an industrial scale. After preliminary tests, CEA-Cad successfully conducted the decontamination of a gas-cooler from the decommissioning site of G2/G3 at Marcoule.

Progress and results

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

The initial sulphuric formulation planned for Winfrith was no longer adequate since the materials involved in this new project were different. The gas-cooler was made of ferritic steel and brass. It was therefore necessary to modify the descaling foam to have comparable erosion levels for the brass and the ferritic steel. A sulphonitric mixture gave the best results.

No rinsing foam was planed for Winfrith. Instead, a water spraying was proposed. Nevertheless, it seemed pitiful to loose all the benefit in terms of minimising the secondary wastes during this last treatment. Therefore additional tests were conducted to finalise a rinsing foam formulation.

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

Feasibility tests conducted on a 2,1m3 pilot unit proved that the circulation of the fluids was mastered and that the process could be implemented on an industrial scale.

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

Decontamination tests were conducted on samples cut from a similar gas-cooler being dismantled. The treatment retained was applied in three steps: a degreasing, a descaling and a rinsing step according to the modified foam formulations. The results obtained confirmed the effectiveness of the foam technique in terms of decontamination.

4. Secondary waste treatment (B.4.)

The treatment of the effluent was our partner's responsibility (AEA, Winfrith). No progress report has been communicated this year.

5. Design, construction and operation of a prototype foam production and circulation rig: non radioactive demonstration (B.5.)

The decontamination device is represented in figure 1. The design has been featured according to the pilot installation. This installation was directly used for the decontamination demonstration. The only non radioactive tests performed on this installation concerned the checking for leaks during the assembly of the industrial pilot.

The decontamination installation was composed of :
- two tanks: CO1 and CO2
- three units: one foam generation unit and two filtration units.

The stock solutions were prepared in a $3m^3$ stainless steel reactor (C02). There, a homogenisation was performed by means of a pneumatic mixer. The chemical solutions were then transferred to the foam generator unit.

The static foam generator was composed of a cylinder made of PVC filled with a knitted fabric. The generator was supplied with the chemical solution by means of pump 1 linked to a shock absorber and with air. The respective flows were monitored with an electromagnetic and a thermal flowmeter.

The foam thus produced entered from the bottom of the gas-cooler expanded and entirely filled the component acting on all the contaminated surfaces. The foam exiting from the top was destroyed into liquid by addition of alcohol in the intermediate tank (CO1). The alcohol (2 pentanol or 4-methyl-2-pentanol depending on the foam treatment) was sprayed by means of a cyclic pneumatic pump. Had any inopportune foaming occurred in the intermediate tank, an additional spraying of alcohol would have been performed by means of a pump monitored by hand.

During decontamination, the liquid resulting from drainage and coalescence was recovered at the bottom of the component, collected and transferred to the first filtration system (F1). The liquid was then coarsely filtered trough a filter (150 microns). Large contaminated particles of grease or oxide were thus retained. Then, the solution was transferred to the intermediate tank by means of pump 2.

To sum up, the function of the intermediate tank was to collect:
- the liquid resulting from drainage and previously filtered
- the liquid resulting from the destabilisation of the foam exiting from the component.

The resulting liquid was then transferred to the second filtration system by means of pump 3. There, it was thoroughly filtered (100, 50 or 20 microns). The recycled solution was then returned to the mixer and the treatment lasted for several hours until decontamination was achieved.

For safety reasons, each filtration system was composed of two filters wired up in parallel. The pluggage of the filters was controlled by means of two manometers fixed on the terminals.

6. Industrial application by radioactive tests on a G3 contaminated graphite-gas cooler (B.6.)

a) The gas cooler

The objective was to conclude the contract with a demonstration of the foam decontamination process on a representative nuclear component.

This industrial demonstration was performed on a gas-cooler from the primary circuit of a graphite-dioxide carbon reactor. The gas-cooler presented a contaminated internal volume of $13m^3$ corresponding to a contaminated surface of about $1000m^2$. It was essentially composed of two materials: the sheath was made of carbon steel and contained 1950 tubes made of brass.

The activity deposited on the tubes was not uniform and was mainly due to ^{60}Co. The contamination level of the component did not seem very high (about 20Bq/cm2 considering the radiological evaluation). Anyhow, there was no need to have higher contamination level, for the main purpose was to study the hydrodynamic features of the foam and to see what residual contamination could be achieved with this process.

As a matter of fact, this gas-cooler was a most appropriate component to prove the effectiveness of the foam technique. The use of foams is specially relevant to large closed volumes, presenting a highly developed surface and a complex internal geometry.

b)The foam process

The treatment was applied in 3 steps as described in Table I.

The circulation of the fluids in the decontamination loop was very well mastered and the quality of the foams proved to be excellent. Only $6m^3$ chemical solutions were needed to treat the 1000m2 contaminated surfaces. Moreover this figure can be reduced to 3 or $4m^3$ since there were always $1m^3$ stock solution remaining in the reactor for each treatment. It is to stress that the volume of the effluents thus produced was reduced by ten in comparison with other usual decontamination methods.

The component has entirely been cut and dismantled by now. The radiological analysis of the materials showed that the residual contamination on the brass tubes and the ferritic steel did not exceed the background noise (1Bq/cm2).

The analysis of the effluents and the filters is being performed. Chemical analysis are also realised to determinate the concentration of copper, zinc, and ferric cations. In a first estimation, about 10^8Bq have been removed (corresponding to ~$20Bq/cm^2$).

As a conclusion, this foam decontamination process was validated on an industrial scale and proved to be very effective. A promising future should rise for such a technique especially in decommissioning fields.

Table I : Foam process and operating conditions

	degreasing foam	descaling foam	rinsing foam
Formulation	NaOH: 2 mol/l Oramix: 0,4% Amonyl:0,3% 2 pentanol: 0,2% to 0,5%	H_2SO_4: 1,5 mol/l ; HNO_3: 0,09 mol/l Oramix: 0,8% Amonyl: 0,3% 4-methyl-2-pentanol: 0,2% to 0,6%	H_2SO_4: 0,02 mol/l; Na_2SO_4: 0,48mol/l Oramix: 0,8% Amonyl: 0,3% 4-methyl-2-pentanol: 0,2% to 0,4%
Foam production rate	70m3/h (to start) 50m3/h (steady conditions)	70m3/h (to start) 50m3/h (steady conditions)	70m3/h (to start) 50m3/h (steady conditions)
Theoretical bulk factor	12,5	14	14
Experimental bulk factor	11,5	13 to 15	11
Bubble diameter	0,5 to 1 mm	1 to 2 mm	no measurement
Δp (generator terminals)	0,75 to 1,5 bars	0,9 to 1,2 bars	around 1 bars
treatment	1 hour under steady conditions	5 hours under steady conditions	2 hours under steady conditions
volume	2,2m3	2m3	2m3
Remarks	before recycling, the solution is dirty with greases, oil and graphite particles	- formation of copper chips, - copper deposits on the carbon steel - the brass tubes present a very yellow shining look	difficulties to render soluble the bisulphate salt

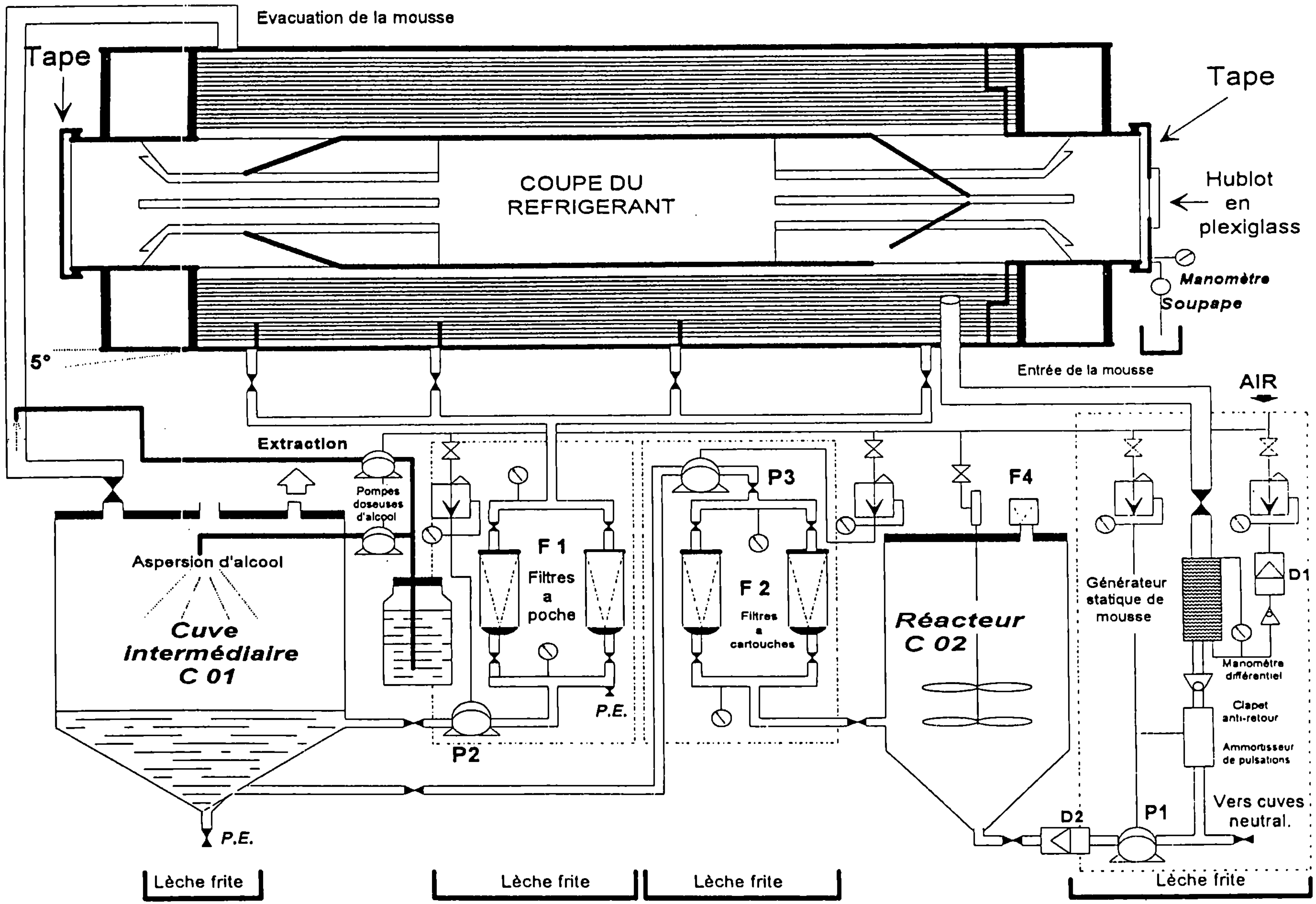

FIG. 1: DECONTAMINATION DEVICE

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 1993
Coordinator: V CALI, ENEA-EUREX
Phone: 39/11/48 32 25 Fax: 39/11/48 33 49

A. OBJECTIVE AND SCOPE

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

The programme includes:

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

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

B. WORK PROGRAMME

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

C. Progress of work and results obtained

Summary of main issues

The contractual activities developed in 1993 were focused in three main directions:
- the improvement of the knowledge of the decontamination mechanism in solution, evaluating the decontaminating action in the presence and in the absence of foam, operating in various experimental conditions.
- the simulation of a decontamination experiment in a mock-up of a vessel of the reprocessing plant, in order to evidence the most significant operational problems and to establish a correct approach to the decontamination of a hot plant.
- the evaluation of the overall set of experimental results obtained in order to correctly design the last stage of the contractual work, that is the feasibility study and the operational decontamination of the vessel of the reprocessing plant.

Progress and results

1. Laboratory tests (B.2.)

The experimental work was continued utilising a new foam formulation:

FOR > - Natrium myrystylethoxysulphate: 0,8 %
- Alkyldiethanolamide: 0,1 %

Two main items were investigated:
- the transport of alpha-activity by an homogeneous surfactant-containing solution through the foam displacement.
- the decontamination of inox specimens by foam contact under different conditions.

In the first case the mobilisation of the alpha-active species was assumed to occur after a step performed in the liquid phase, whereas in the second set of experiments the specimen decontamination and transport take place together in the foam phase.

1.1. FOAM TRANSPORT

1.1.1. Experimental set-up

A small amount of Plutonium in a nitric acid solution, was put inside a two litre container C (Figure. 1), together with a foam-forming solution FOR.
Bubbling with air results in foam formation that is collected in another container B. The alpha-activity associated with the foam (ACT_{tr}), was determined with an alpha-counter, together with the activity that still remains in solution (ACT_{res}).

1.1.2. Results

The percentage of the total activity that is transferred with the foam utilising three different decontamination systems is shown on Figure 2 :

- FOR in HNO_3 0,2 M
- FOR in HNO_3 0,2 M + SDS 0,1 % (SDS = Sodium dodecyl sulphate)
- FOR in HNO_3 0,2 M + EDTA0,1 M (EDTA = Ethylenediaminotetraacetic acid)

The aim of adding SDS was mainly directed to favour the formation of the ion-pairs (Pu ion), thus facilitating the transport of more hydrophilic species towards the water-air interface. The results seem to indicate only a very small favourable kinetic effect (in particular for lower treatment times).

Alternatively the presence of EDTA (in quite high concentration) acts as a competitor, due to the probable formation of complex with Plutonium, thus the Plutonium available for interaction with the foaming agent (particularly Natrium myrystylethoxysulphate) is lower.

1.2. DECONTAMINATION OF TEST SPECIMENS

1.2.1. Experimental set-up

The apparatus and procedures were described in previous Reports .

The effect of the acidity (HNO_3 0,2-2 M) on Plutonium removal from the specimens is, as expected, important. After about three hours contact between the contaminated specimen and the foam (or foam-forming solution) at pH ca. 5, a modest decrease of the activity (ca. 75 - 80%) was observed, whereas in the presence of nitric acid 0,2 - 2 M, after the same contact time the specimen activity drops to ca. 10 - 20 % (Figure 3).The foam can facilitate the acid action, presumably through the reduction of the solid - liquid interface tension, thus improving the penetration of the acid within the deposited solid layer.

The possibility of formation of ion-pairs between Pu cationic species and SDS (0,1% weight), was also investigated. The observed effect on the foam activity was not significant, thus indicating that the removal of deposited contaminants from the specimens is mainly due to the acid component.

Experiments were also performed in the presence of EDTA 0,1 M, in order to test the decontamination feasibility under mild acidic conditions (pH 3,5). About 30% of the residual activity has been measured after two hours treatment with ligand-containing foam (Figure 3). This value, although lower, is not far from that obtained using strong acid foams. For example, ca. 18 % of the residual activity remains after treatment with FOR - HNO_3 2 M foam.

Finally a set of decontamination runs using acid foams were conducted in the presence of the TOPO 0,1 % (w/w) as additive. The aim of these experiments was to test the ability of such hydrophobic ligand (usually applied in strong acid media) to improve the decontamination performances.

The decontamination results, shown in Figure 4, indicate that the ligand effect does not depend on acidity in the examined range (HNO_3 0,1-1 M) and that foam mixtures containing FOR-TOPO 0,1 % in HNO_3 0,2 M exhibit nearly the same performances as FOR-HNO_3 2 M foam.

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

The cold mock-up, extensively described in previous Reports, was contaminated with a natural Uranium solution, following a typical evaporation cycle.

The tank was then washed and decontaminated with a foam-producing solution.

A typical experiment is reported in Table I

Two decontamination strategies are compared:

A - The contaminated tank was fully filled with a 2 M nitric acid solution, which after being mixed for four hours, was evacuated.

B - Two batches of foam-producing solution in HNO_3 2 M were introduced in the contaminated tank. The foam, produced by means of air bubbling in the solution, was collected and measured; the contaminated liquid remaining at the bottom of the tank was discarded.

Some important features can be evidenced:

1 - With the above-mentioned experimental conditions, the removed contamination is similar for A and B (45 ~ 50 %).

2 - Using the foam system the amount of the resulting waste is only 1/10.

3 - The activity transferred with the foam shows a regular pattern (see Figure 5), but, with this experimental set-up, it is only a small percentage of the total activity (0,4 %).

3. Discussion of the results.

The work done, including laboratory and mock-up tests, demonstrates the feasibility of the decontamination of components of nuclear installations by this method and has provided a large amount of experimental data on this technique.

TABLE I - Mock-up decontamination

Decontaminating agent	LIQUID		FOAM	
	ACT_{res}	*Volume*	*ACT_{tr}*	*$Volume_{tr}$*
	%	*[l]*	*%*	*[l]*
A-HNO_3 2 M solution	45	20		
B-FOR in HNO_3 2 M	50	2	0,4	0,1

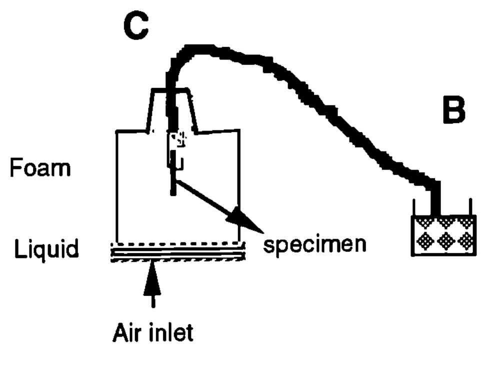

Fig. 1 - Apparatus for foam decontamination

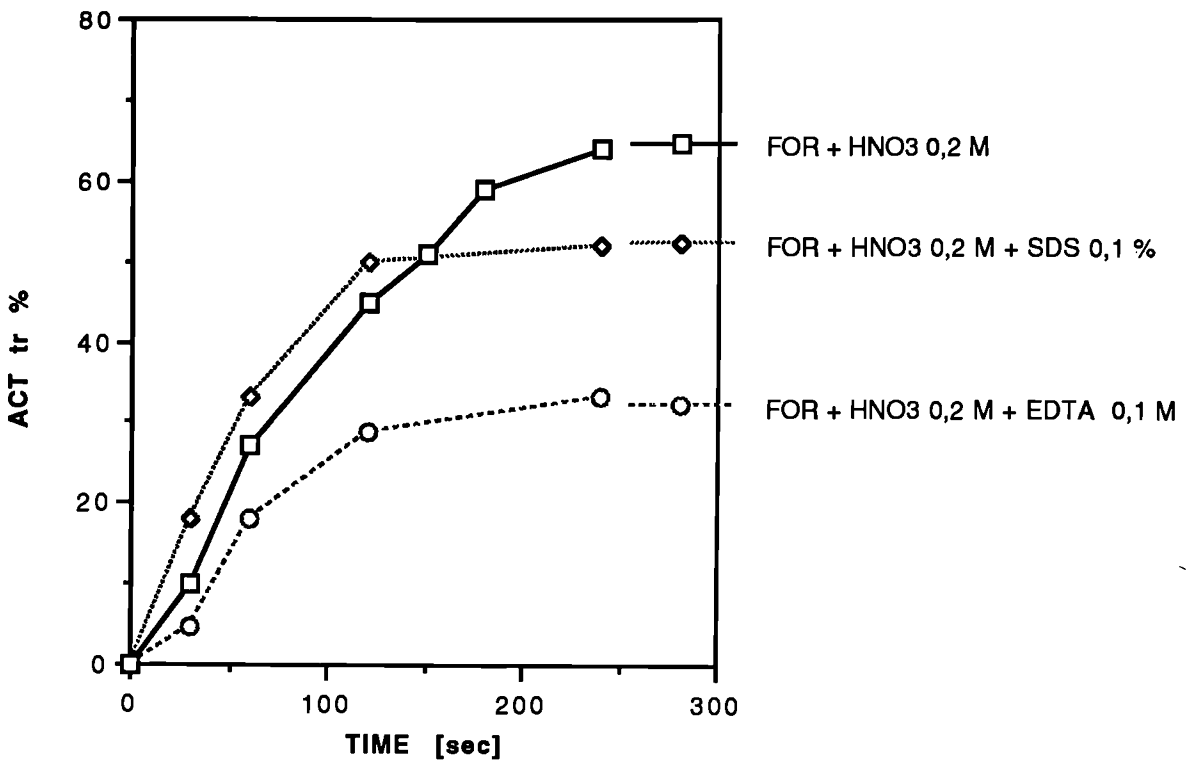

Figure 2 - Foam transfer

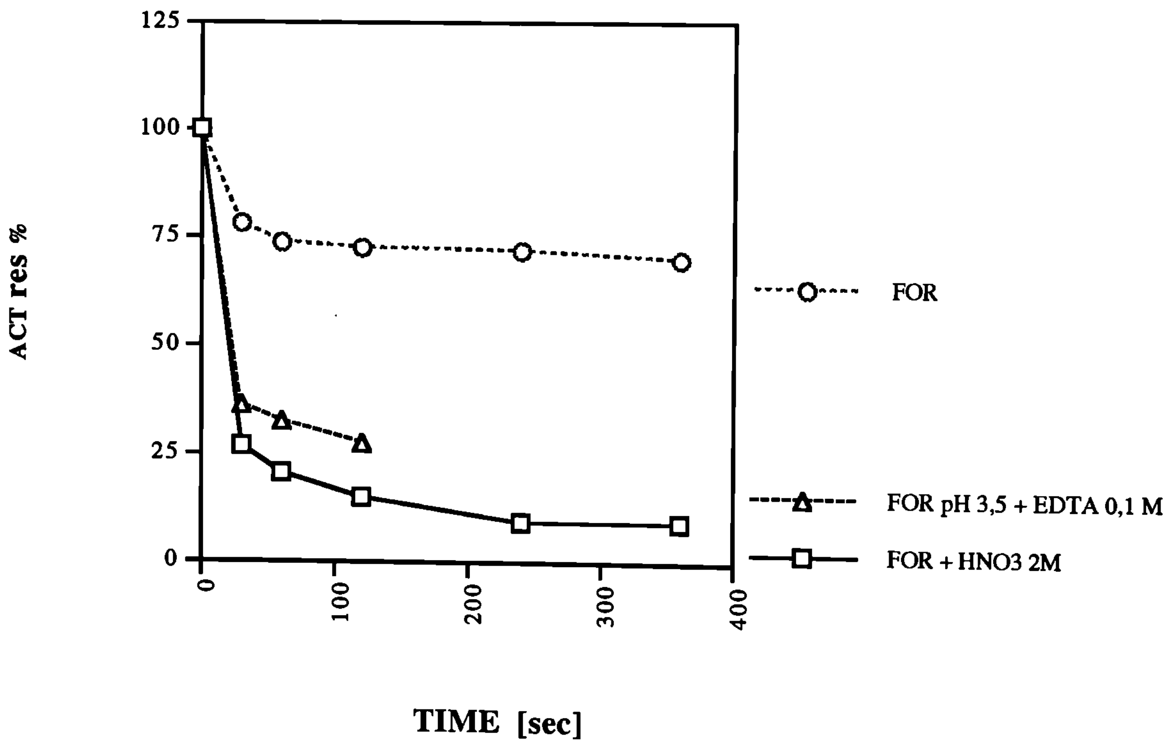

Figure 3 - Specimen decontamination

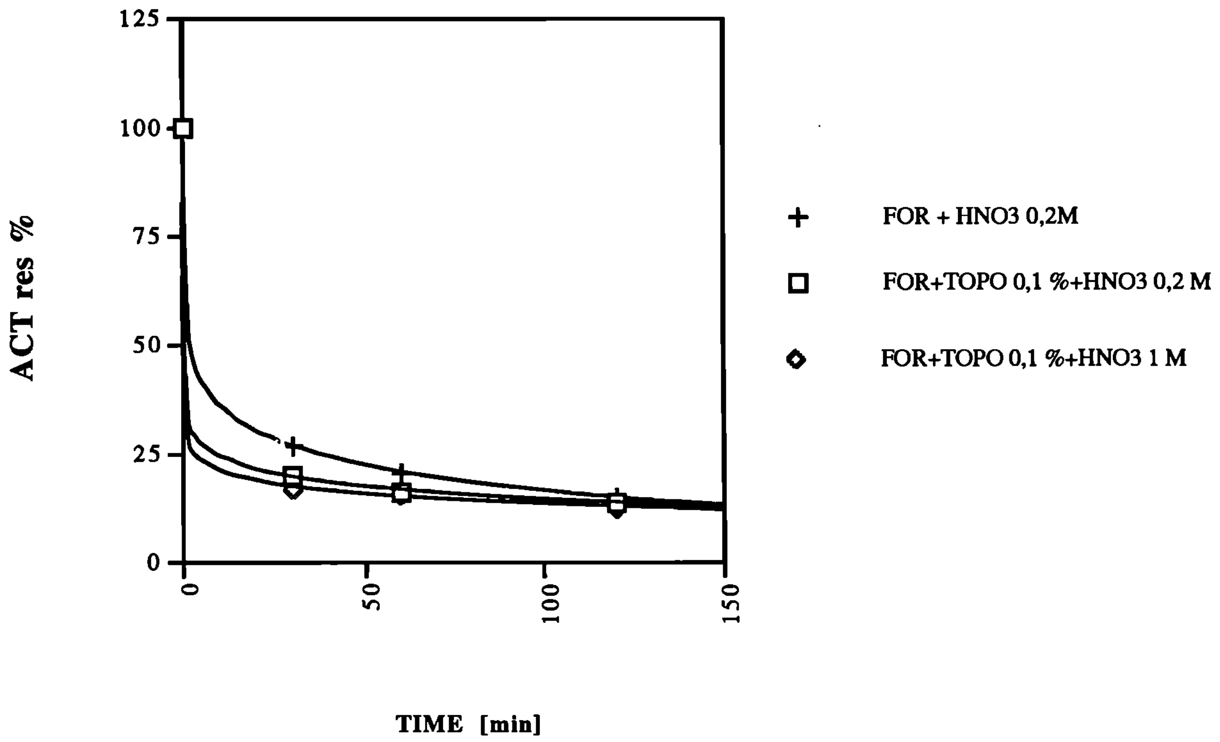

Figure 4 - Specimen decontamination TOPO 0,1 %

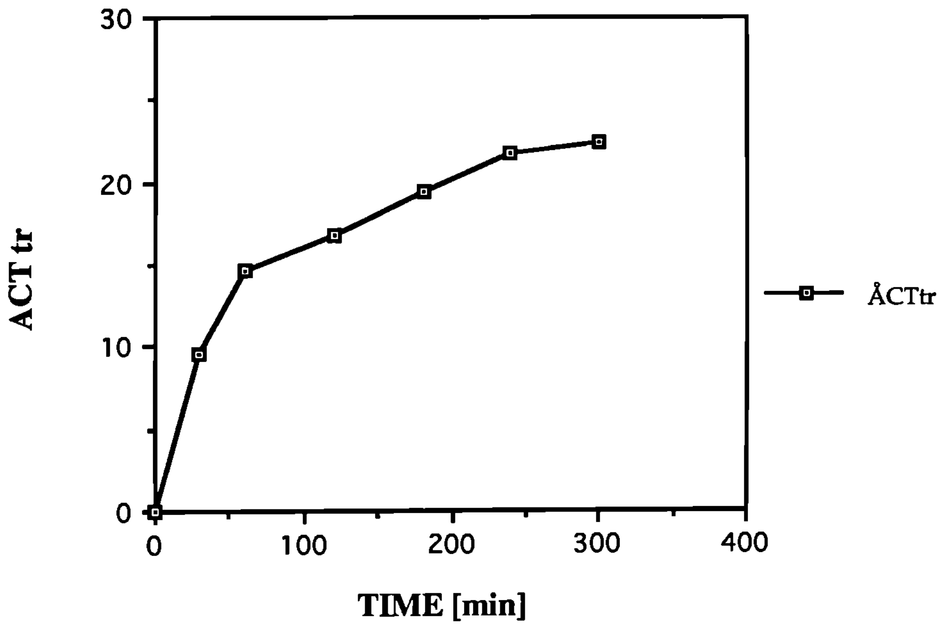

Figure 5 - Activity transferred with the foam

2.6. DECONTAMINATION TECHNIQUE USING A DISPERSED CHEMICAL AGENT

Contractors: BATTELLE Ingenieurtechnik GmbH
Contract No.: FI2D-0054
Work Period: September 1991 - March 1993
Coordinator: G POß
Phone: 49/6196/936 422 Fax: 49/6196/936 499 or 199

A. OBJECTIVE AND SCOPE

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

The programme essentially includes:

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

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

B. WORK PROGRAMME

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

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

In the first quarter of 1993, further experiments to complete working phase B.4. were performed. After demonstrating the capabilities of film etching applying ultrafine fogs of low viscous agents as well as a thicker layer of special mixed, higher viscous etching in 1992, some final experiments with the gel-like HNO_3/Blacid TS-B/Tylose mixture [1] confirmed the decontamination results achieved in previous investigations: applying the mixture as a 1 - 2 mm uniform layer on austenitic steel targets deliver reductions of total activity of typically 30% for a single process cycle. The samples investigated originated from a feed water pipe of the Gundremmingen KRB-A with overall activities in the range of 1000 to 12,000 Bq.

Progress and results

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

Like the previous investigations, the final etching experiments with radioactive contaminated samples and the measurements of the initial and residual target activities were performed in a KGB laboratory. The samples originated from a feed water pipe of the KRB-A. The targets were nearly cubic with approximately 5 cm edge length, carrying total initial activities in the 300 Bq/cm² range. The standard HNO_3/Blacid TS-B/Tylose mixture developed was applied to the samples with approximately 0.2 ml/cm², corresponding to a 2 mm film thickness. After 30 minutes of induction time, the liquefied gel was removed. The initial and the final target activities, as well as the gel-bounded activities were measured separately. The results of previous experiments could be confirmed showing that the complete acid attack with the 2 mm gel film needs less than 30 minutes, and a total reduction of approximately 30% of the initial sample activity is achieved per treating cycle (see Tables 1 and 2). The removed activity was found as gel-bounded activity in the rinsed liquefied gel. Radionuclide analyses prior to and after the etching process show that the main contribution refers to Co-60 and Cs-137.

Reference

[1] Blasberg Oberflächentechnik, Solingen, Germany; Gebrauchsanweisung für "BLACIT TS-B" Beisalz BN 25 610.

Tab. 1 **Gel etching experiment, Sept. 3, 1992**

sample* no.	surface area [cm^2]	initial sample activity [Bq]	reaction time [min]	gel bounded activity [Bq]	reduction sample activity [%]
1	25	19200	30	6207	32,3
2	25	849	60	204	24,0
3	25	2890	90	968	33,5

* Samples from a feed water pipe, Gundremmingen KRB A

Tab. 2 **Gel etching experiment, Febr. 8, 1993**

sample* no.	surface area [cm^2]	initial sample activity [Bq]	reaction time [min]	gel bounded activity [Bq]	reduction sample activity [%]
1	42,3	11844 ±47	22	3910 ±147	33
2	23,0	7245 ±63	30	2030 ±136	28
3	42,5	11857 ±47	30	3550 ±33	30

* Samples from a feed water pipe, Gundremmingen KRB A

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 (eg reactor pressure vessel) and reinforced-concrete structures (eg reactor shielding) of redundant nuclear installations, account being taken of the particular requirements due to radioactivity.

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

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

C. Programme 1989 to 1993

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

D. Programme implementation

Eleven research contracts relating to Area No. 3 were concluded, of which seven were completed at the end of 1993.

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

Contractor: TÜV Bayern
Contract No.: FI2D-0007
Work Period: October 1990 - June 1993
Project Manager: P BÖHM, TÜV Bayern
Phone: 49/89/51903 165 Fax: 49/89/51903 191

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

This project was completed in June 1993. The final report is available as EUR Report No. 15463.

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

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

A. OBJECTIVE AND SCOPE

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

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

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

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

B. WORK PROGRAMME

B.1. **Definition of cutting parameters for decommissioning purposes** (UH-IW)
B.2. **Development of controlling systems for processes parameters and the cutting result** (UH-IW)
B.2.1. Preparation of a two-dimensional feeding mechanism for underwater cutting tests.
B.2.2. Development of an on-line controlling system to detect the state of wear inside the cutting head.
B.2.3. Development and adaptation of controlling methods to verify the cutting result during or just after cutting.
B.2.4. Design of a cutting head which includes controlling systems, cutting tests to qualify the sensor systems.
B.3. **Development of methods to remotely replace worn parts of the cutting head under water** (UH-IW)
B.4. **Characterisation and handling of secondary waste**
B.4.1. Preparation of test facilities for measuring aerosols and suspended particles when cutting in air and under water (UH-IW).
B.4.2. Measurement and characterisation of the secondary emissions when cutting or kerfing in air or under water (CEA).
B.4.3. Development of methods to lower the spreading out of emissions in air or under water (UH-IW).
B.4.4. Cutting tests to determine the efficiency of measures to lower the emissions and to determine the filtration systems (UH-IW, CEA).

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in December 1992. The final report is available as EUR Report No. 15241.

3.3. STEEL CUTTING USING LINEAR-SHAPED CHARGES

Contractor: OTO MELARA
Contract No.: FI2D-0010
Work Period: July 1990 - June 1994
Project Manager: G PAZIENZA
Phone: 39/187/582 785 Fax: 39/187/582 669

A. OBJECTIVE AND SCOPE

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

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

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

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

B. WORK PROGRAMME

B.1. Determination of basic charge parameters

B.1.1. Theoretical assessment to characterise high performance cutting charges.
B.1.2. Manufacture of charges and execution of tests.
B.1.3. Analyses of the experimental data compared with the theoretical results, conclusions on first phase.

B.2. Optimisation of the cutting charges

B.2.1. Theoretical assessment to further optimise important parameters.
B.2.2. Manufacture of charges and execution of tests with measurements of blast effects in the air, of ground vibrations, photographs from an ultra-rapid framing camera and of flash X-ray tubes.
B.2.3. Analyses of the experimental data with a view to specify high-performance charges.
B.2.4. Study of the fracture phenomena.

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

B.3.1. Theoretical assessment of the final configuration of high-performance cutting charges and specification of 8 tests (in order to determine the scaling law).
B.3.2. Manufacture of charges and execution of tests against a vessel steel target.
B.3.3. Manufacture of charges and execution of large-scale tests in special areas allowing large amounts of explosives.

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

C. Progress of Work and Results Obtained

Summary of main issues

In the framework of the final optmisation of the linear cutting charge, we carried out No.2 instrumented tests with the **NC1 type charge** (20% less explosive than the reference one) against a larger target of mild steel with the dimensions of **200x230x1000 mm**.
The aim was to study the influence of target size on the appearance of the fracture plane.
For comparison purposes No.1 instrumented test against the standard small target having the dimensions of **200x230x150 mm** was also carried out [B.3.1.].

Some numerical simulations were also conducted in order to assist the experimental study. The theoretical cutting process in the two cases (small and big target) was studied in particular. The aim of this work was to minimise the number of the required experimental tests.

No. 5 new NC0 reference charges are going to be manufactured for future tests.
For these tests the **ASTM A533-B Vessel steel** was supplied for the targets that will have the following dimensions :

No.2 ....140x230x150 mm (small)
No.2........140x230x300 mm (medium)
No.1........140x230x1000 mm (large)

This material is heat treated in order to simulate the embrittling effect than can be expected, for the common vessel steels, at the end of their operating life due to the neutron irradiation (fluence>$2x10^{19}$ neutrons/cm^2) .
It is expected that, due to its brittle nature, the probability of crack propagation increases during the cutting process, [B.3.2.].

The study of the fracture phenomenon was continued in conjunction with the **Department of Mechanical and Nuclear Constructions of the University of Pisa** about the evaluation of the **stress intensity factor (K_I)** of the material when subjected to the dynamic conditions of the cutting process (numerical calculations) [B.2.4.].

Progress and Results

1. Burst tests against long target

Such tests, carried out against a 200x230x1000 mm mild steel target, were instrumented so as to measure the level of longitudinal acceleration induced in the target, as well as the amount of deformation along the thickness, i.e. along the plane of cutting.
The above-mentioned parameters are in fact useful indicators in case of plane of fracture creation, as they describe both the kinematics and the stress state of the piece during the first instants of the cutting process.
The second of the two tests was conducted using a double initiation, by means of two detonators triggered at the same time (see figure 1). In the latter case the aim was to verify whether such configuration was more effective in terms of both uniformity and depth of cutting.
The results of the measurements conducted on the target of the second tests are given in figure 2 and the maximum cutting depths for the No. 3 tests are summarised in Table I.
In none of the two cases with large target the prosecution of the cutting by means of the development of the plane of fracture was obtained. It can therefore be inferred that the inertia of the target plays an essential role in terms of quantity of energy available for the onset and propagation of fracture.

Moreover comparing the cuts produced on test No.1 (not shown) and test No.2 targets (figure 2), a more uniform and deeper cut can be observed. It is in fact highly probable that in such case the initiation be more effective.
On the basis of the evidences above and of the results obtained previously, it was decided to select charge type NC0 as the one to be employed for the subsequent tests. Charge NC0 has the following characteristics:

mass of explosive (Octol) : 505 $\pm$ 5 g
mass of explosive per unit length : 25.2 g/cm
liner thickness (copper, angle = 100°)
stand-off : 100 mm
double initiation

Such choice was determined by the fact that, contrarily to what it was thought initially, it was observed that it was more convenient to use a charge having a higher explosive content, in order to facilitate the fracture onset.

2. Theoretical study of the fracture phenomenon

The main evidence which came out from the theoretical study of fracture, conducted in parallel with the experimental one, is the one which indicates as highly improbable, especially in the case of large targets, the unstable propagation of cracks for material with high toughness.
In other words, to obtain a prosecution of the cut by means of fracture in case of large targets, with the aim of minimising the amount of explosive needed, sufficiently brittle materials are to be used so that their fracture does not require a very high energy.
As a matter of fact, it has to be taken into account that the Vessel steel to be cut, is in reality embrittled, at the end of its operational life, due to the neutron irradiation, which can be estimated in the order of 2×10^{19} neutrons/cm^2.
On the basis of what is said above the steel ASTM A533-B subjected to a suitable thermal treatment in order to approach the end life conditions for a nuclear plant, was selected.

3. Two-dimensional numerical simulations of the cutting process

In order to time up the experimental tests and to investigate qualitatively the influence of the target dimensions on the conditions needed for the onset of fracture, some simulations of the cutting process were carried out by means of the numerical code AUTODYN-2D.
The parameters used in the simulations were the following ones:

laminar jet (copper)
infinite length (2D translational symmetry)
width : 20 mm
thickness : 2 mm
velocity : 3200 m/s (from experimental evidence)
target
case 1 : dimensions 200x230x150 mm (small)
case 2 : " " 200x230x1000 mm (large)
material : mild steel with HB=160 and UTS=720 MPa

The main conclusion from this study is that the inertia of the target influences mainly its deformation and consequently the cutting depth: 42 mm in case 1 and 38 mm in case 2.

4. Description of the tests envisaged in the continuation of the programme

As anticipated at point 1, No.5 tests were planned, using as target material the fully qualified Nuclear Vessel A533-B steel.
Such material is suitably thermally treated in order to approach its conditions at the end of its operational life, after a neutron irradiation of the order of $2x10^{19}$ neutrons/cm^2.
Such steel will be a lot more brittle than the one employed in the programme to date, it is therefore deemed possible to obtain an after-cutting fracture such as to cut large thicknesses (at least 140 mm), minimising the quantity of explosive needed.
The samples which will be used for the test will have three different dimensions, in order to eventually also assess the influence of the target dimensions:

No.2 ....140x230x150 mm (small)
No.2........140x230x300 mm (medium)
No.1........140x230x1000 mm (large)

The No.5 cutting charges to be employed (already under construction) will be of the NC0 (reference) type with double initiation, for the above mentioned reasons.
A preventive cooling of the targets at temperatures in the order of -30°/-40°C is also foreseen, by means of tanks filled with dry ice (solid CO_2).
Such provision has the aim to increase the brittleness of the material (see figure 3.1) and the possibility cannot be excluded of adopting it also in operational conditions, utilising pre-existing cooling systems of the nuclear reactor.

Table I : Results of the experimental tests

test No.	type of charge	dimensions of target (mm)	maximum cutting depth (mm)
01	NC1	200x230x1000	50 no fracture
02	NC1(double initiation)	200x230x1000	55 more uniform cutting, no fracture
03 (reference)	NC0	200x230x150	40 with partial fracture

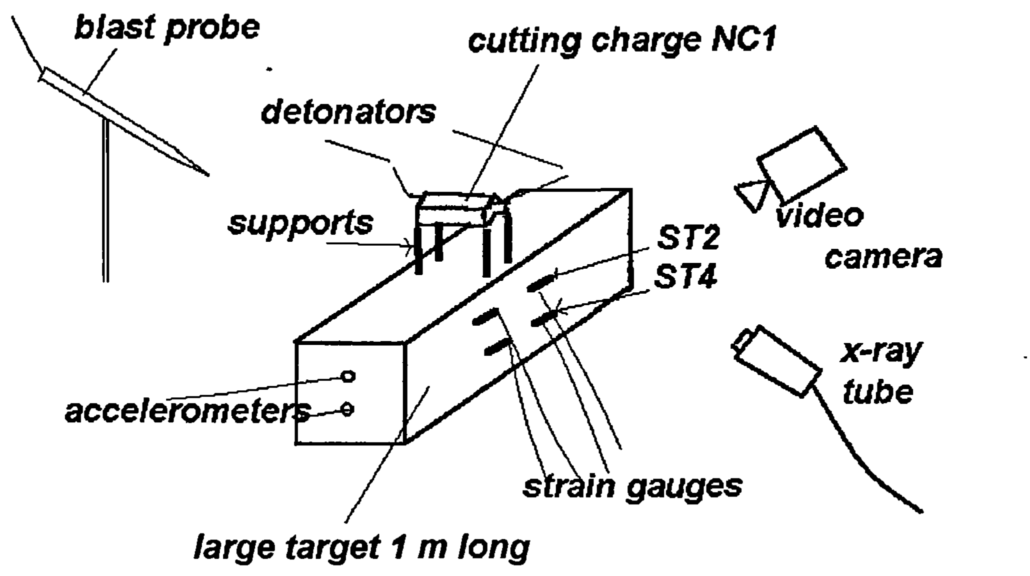

Figure 1 : Experimental set-up of the burst tests

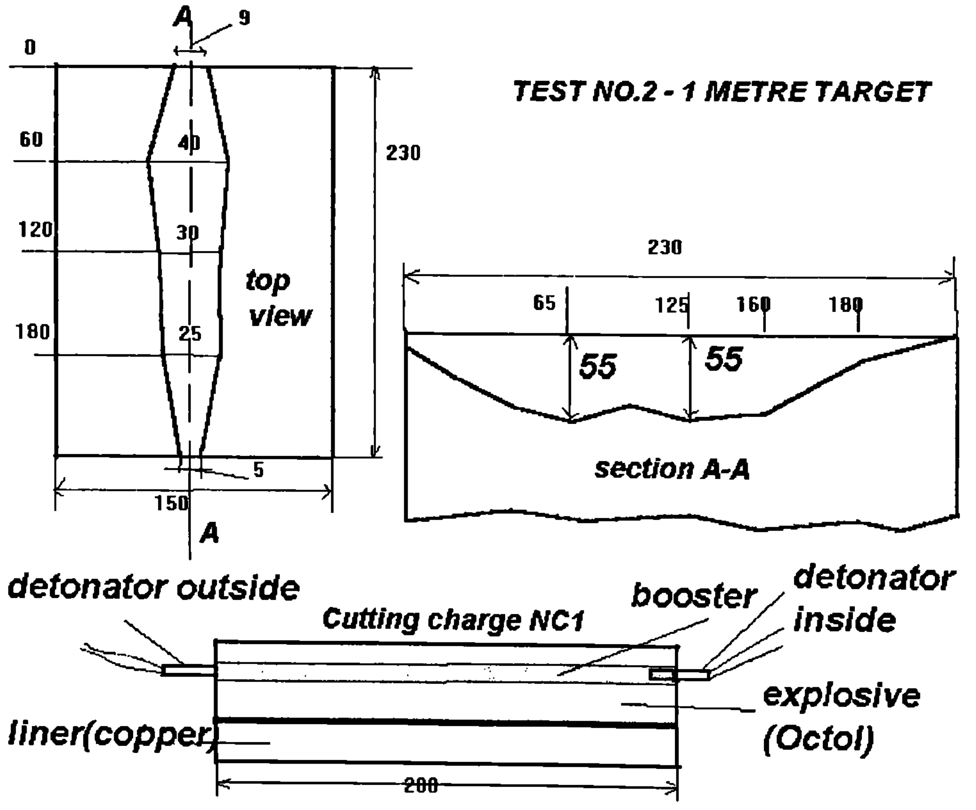

Figure 2 : Results of the second burst test

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

Contractors: CEA Valrhô, CEA-Sac
Contract No.: FI2D-0013
Work Period: October 1990 - June 1994
Coordinator: J R COSTES, CEA/DCC/UDIN
Phone: 33/66 79 63 13 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

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

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

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

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

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

B. WORK PROGRAMME

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

C. Progress of work and obtained results

Summary of main issues

The facility where the cuts take place, the analytical techniques, the test procedures, the tools and first provisional results have been described in the previous reports.

All the measurements relating to the five tools (alternating saw, arc-air, plasma torch, arc saw and grinder) for the cutting of mild steel and stainless steel plates with thicknesses of 10, 30 and 50 mm are practically finished. The comparisons concern the using range of the tested tools, the wear of the most exposed parts, the cutting speed, the production of secondary emissions with their distribution (sedimented dross, attached slag, deposits on the walls of the cell, aerosols in the exhaust duct), the chemical analysis of the aerosols on the sampling filters, the gaseous production and the operational costs.

Progress and results

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

The preparation of the testing cell has been described in the first annual report.

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

The selected tools and materials and also the test procedures have been detailed in the second and the third annual reports.

3. Simulated conditions (B.3.)

The composition of the cut plates and of the worn parts of the tools are known and chemical analyses on the aerosols of the sampling filters have been carried out.

These chemical analyses on stable isotopes (Fe, Ni, Cr, Mn particularly) give information about the behaviour of the radioisotopes of the same elements (^{55}Fe, ^{63}Ni, ^{51}Cr,^{54}Mn).

By examining the ratios Cr/Fe, Ni/Fe and Mn/Fe in the aerosols and by comparing these ratios to the same ratios of the cut plates, it can be concluded that:

- the ratio Cr/Fe remains constant,
- the nickel is enriched in the aerosols by a factor from 1.4 to 4,
- the manganese is also enriched in the aerosols (factor of 9 for grinder and arc saw and about 5 for the other tools),
- the presence of copper (6% of the analysed elements) in the aerosols for arc-air (due to the sheathing of the electrode) and the presence of titanium (1.7% of the analysed elements) for arc saw (from the wheel),
- the ratio aerosol total mass/mass of analysed elements varies between 1.57 and 1.72 for grinder, arc-air and plasma torch and is about 2.5 for arc saw, what emphasizes an oxidation of the aerosols.

4. Secondary waste analysis (B.4.)

The analytical techniques and the results of the measurements of the secondary wastes have been described in the previous report.

5. Final evaluation (B.5.)

The main results are as follows:

5.1. Cutting speed

The fastest tool is the plasma and the slowest the alternating saw. The aerosols produced by unit of cut length are decreasing with the cutting speed, so in order to minimise the aerosols, a relatively fast cutting speed has to be used.

5.2. Kerf

The kerf is narrower for the alternating saw and the plasma. This parameter is important because it is directly linked to the by-products induced. Except for the arc saw, the kerf is increasing with the thickness.

5.3. Wear of the tool

The wear of the tool is particularly sensible for the arc-air (graphite electrode), the grinder (wheel) and mainly the arc saw (wheel). The wear of the nozzle of the plasma torch is negligible.

5.4. Solid secondary emissions

The arc saw and the arc-air are the tools which produce the most important mass of wastes (Figure 1).

- The sedimented dross represent the major part of these secondary emissions (from 90 to 99.9% of the total mass collected).
- The deposits on the walls of the cell are significant for the arc-air, the arc saw and the grinder. The wheels of the two last tools induce showers of sparks that are ejected towards the walls.
- The alternating saw produces much less aerosols than the other tools (factor > 100). The cut of mild steel plates induces more aerosols than the cut of stainless steel plates. The arc saw (for stainless steel) and the arc-air (for mild steel) are the tools which produce more aerosols (Figures 2 and 3).

The thermal tools, like plasma torch, arc saw and arc-air, induce more submicronic aerosols than the two other tools.

The size distribution of aerosols is often bimodal and sometimes trimodal emphasizing the several phenomena of formation. The main mode, for the thermal tools, is often around 0.8 μm.

5.5. Gaseous secondary emissions

During the cuts, NO, NOx, CO, CO_2 and O_3 were measured.

The productions of nitrogen oxides are about 0.3 - 0.5 l/min for the cutting by arc-air, plasma torch and arc saw. The production can be increased in case of use of nitrogen as plasma gas.

The production of carbon oxides (CO) is important for arc-air (1.8 l/min).

5.6. Operational costs

An assessment based on the surface cutting speed (m^2/h), the price of the tool and its supply, the wear of the tool and the spare parts, the electric consumption, the production of wastes (worn parts, secondary wastes, tool and supply) and their removal show that the operational cost, as mentioned above, is relatively similar for plasma torch, grinder, alternating saw and arc-air but is more important for arc saw due in major part to the price of the wheel.

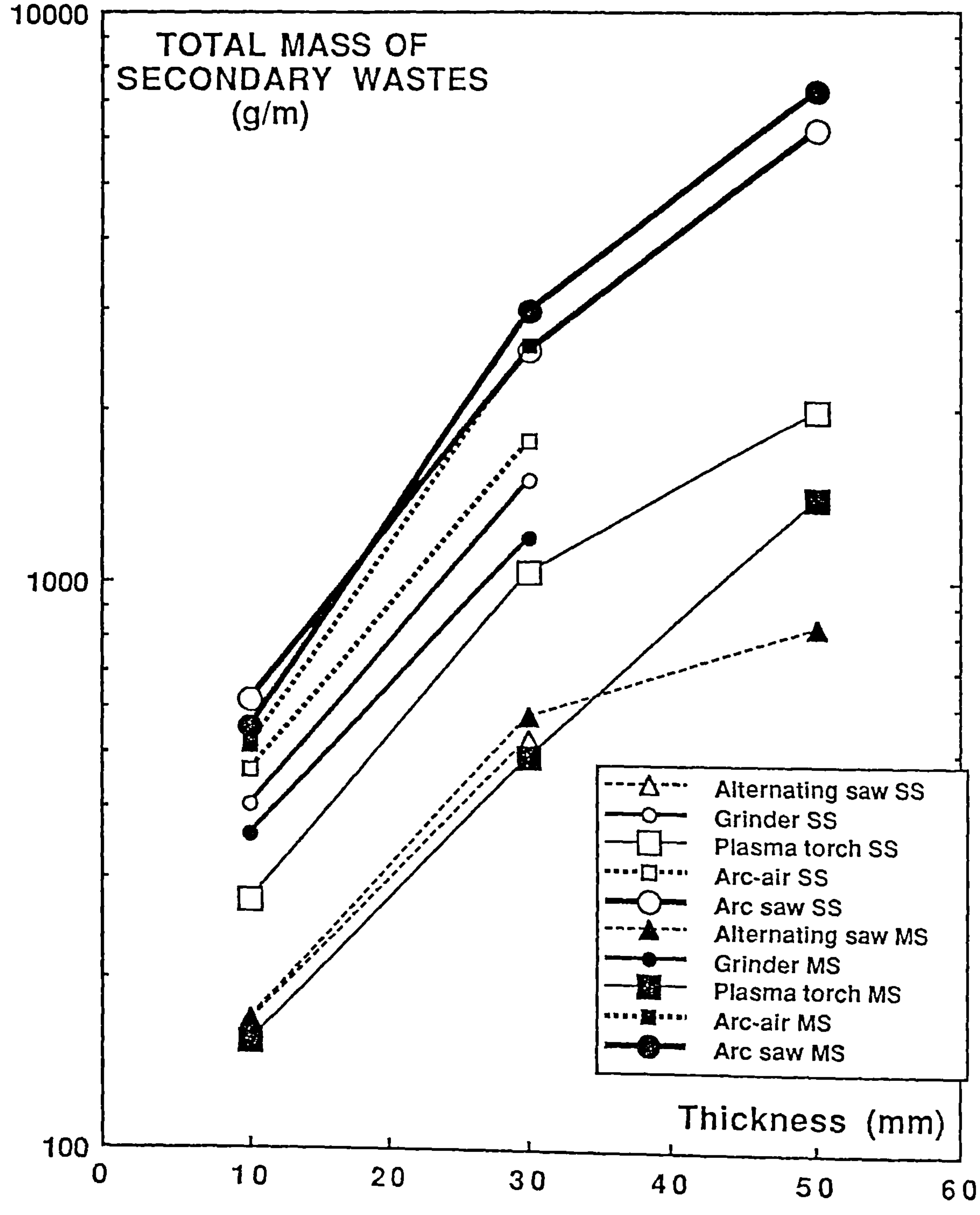

Figure 1: Total mass of secondary wastes produced by five cutting tools versus the thickness of the stainless (SS) and mild steel (MS)plates

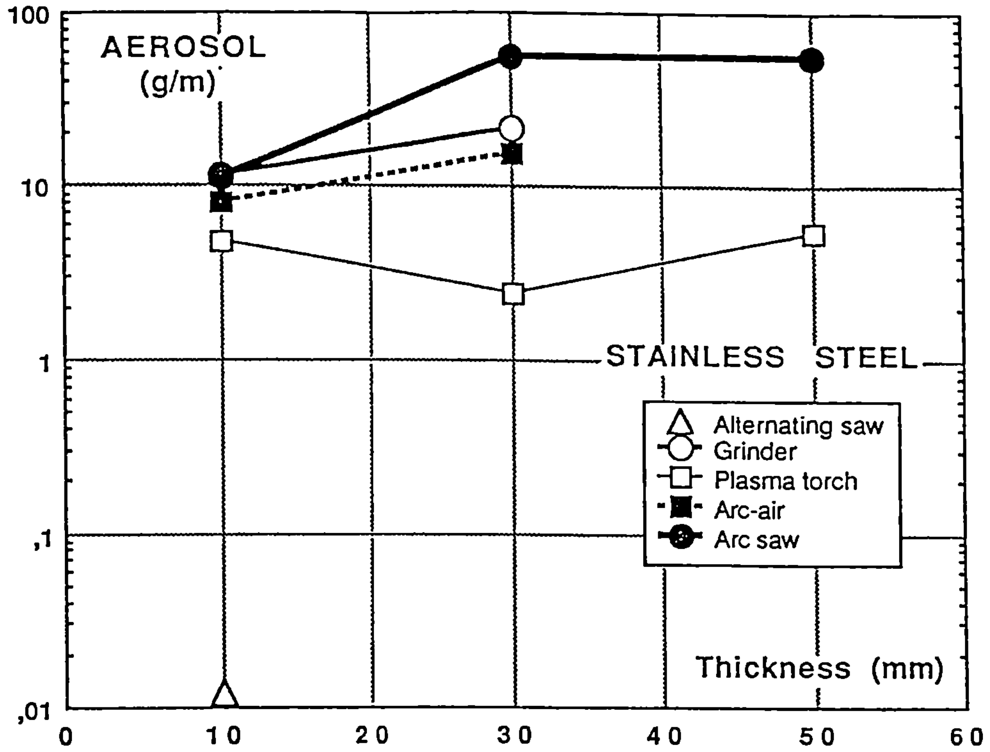

Figure 2: Aerosol production for five cutting tools versus the thickness of the stainless steel plate

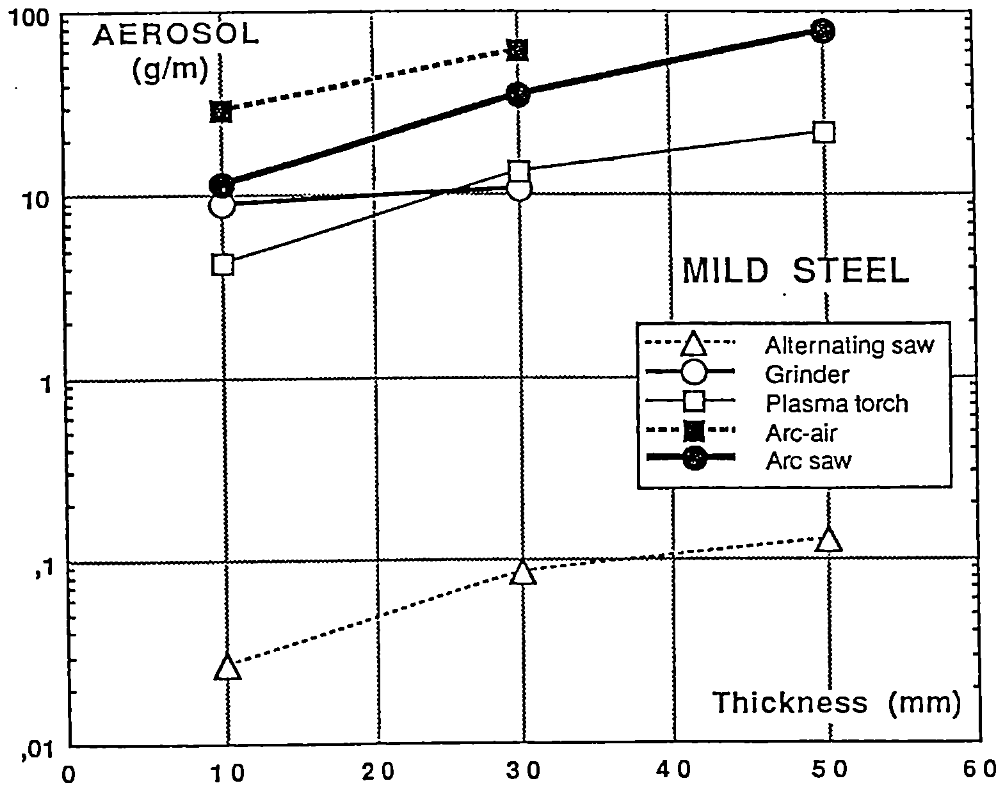

Figure 3: Aerosol production for five cutting tools versus the thickness of the mild steel plate

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 - March 1994
Coordinator: J P DUFAYET, CEA Cad, SST/SEMO
Phone: 33/42 25 72 11 Fax: 33/42 25 47 57

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, consumable electrode and CAMC
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

In 1993 CEA-Cad and IW finished the development of the cutting tools and APS finished the final adjustments to the manipulator control module.

In this framework, the plasma cutting tests at Cadarache were continued with the new L'TEC torch with a capacity of 1000 A, which allowed a maximum 125mm thickness to be reached for a cut in pure argon. At Hanover, the new CAMC tool was developed and allowed the objective of a 200mm thickness to be cut at a speed of 25mm/min with two parallel sources.

Effort was then focused on the preparation of different cutting apparatus and tools: the plasma torch, the consumable electrode, the CAMC, with a view to final trials in the Pegase facility. With this aim, IW notably succeeded in using the same source of current (a DC welding rectifier) for both the consumable electrode and the CAMC tool.

The facility was then equipped with devices allowing sampling of the secondary emissions and certain direct measurements. Tests were made on 80mm thick stainless steel plates, then on radioactive plates from the G2 reactor at Marcoule which is being dismantled.

Finally, the Pegase equipment was disassembled with a view to its transfer to Aix la Chapelle in January 1994. The equipment is scheduled to be installed in the new European Centre for Mechatronics so as to complete testing of the automatic control module on representative models.

Progress and Results

1. Development of the Plasma Torch

We attempted to use the SAF TD 600N torch at high voltage several times, but we noted that it was very severely damaged when the current exceeded 900 Amperes. We finally abandoned this torch and preferred the L'TEC torch that could run with no damage from 1000 to 1100 Amperes. With this new torch (see Figure 1), we were able to cut stainless steel plates at a depth of 30cm underwater up to a thickness of 125mm, using argon as the plasma-producing gas.

However, we observed a loss in efficiency under 2m of water, mainly due to the fact that the removal of kerf dross was much more difficult. The dross had a tendency to adhere to the workpiece near the kerf, which eventually hindered torch operation. Using an argon-hydrogen mixture as the plasma-producing gas would probably enable these results to be improved, as hydrogen would provide greater power at the plasma level. These tests were not performed at Cadarache because of the risk attached to the use of this gas. As a result, the limit of the cutting thickness capacity under 2m of water is about 100mm.

2. Development of Other Tools

Contact Arc Metal Cutting (CAMC)

The CAMC cutting technique was used in the present project as a new technique from the former Soviet Union that works with a non-standard AC power source. The first tests showed promising results and led to the conclusion that this technique should replace plasma saw cutting, which was found to be unacceptable for remote-controlled use.

The prototype of the CAMC cutting tool was improved by changing the size of the water jet nozzles and their arrangement vis-à-vis the kerf. The cutting speed could therefore be increased from less than 10mm/min to more than 100mm/min for a workpiece thickness of 80mm.

The CAMC technique cannot compete with plasma for standard workpiece thicknesses up to 80mm, but in the case of greater thicknesses, considerable water depths or intricate workpiece shapes such as tubes, CAMC may have definite advantages.

The result of CAMC cutting for 150mm thick stainless steel can be seen in Figure 2. The wear of the electrode proved to be very dependent on the latter's material. The advantage of rigid electrodes reinforced with carbon fibers conflicted with their low density and therefore low energy transfer to the workpiece. Any increase of the cutting speed led to considerable corresponding wear of the electrode.

The width of an electrode is limited to 120mm because of the holder and the water jets. This means that a cutting length of up to 1000mm should be possible for a workpiece of 100mm with a single electrode. Changing the electrode is very simple and could easily be automated.

Consumable Electrode Tool

The consumable electrode tool is shown in Figure 3. The unit is mounted on the manipulator and electrically isolated by means of a plastic block. The housings of the motor and of the gear are under continuous slight air pressure to prevent water entering. When CAMC and water-cooled cables were used in the present project, it became possible to reduce the amount of cables to 2x2 cables with a cross-section of $90mm^2$ each. As the same power source could be used for both tools at the same time, obviously the same water-cooled cables could be used too. The consumable electrode tool was altered for this purpose, allowing the water-cooled cables to be connected to the tool.

The tests proved that it was possible to cut the "tube" model using a single power source and a 4mm wire. In this case the cutting speed is about 75mm/min. The cutting result is shown in Figure 4. Although the model can be cut this way, the wire feed cannot be optimized, as is possible for standard workpieces. This means that the wire is not completely consumed within the kerf and remnants of the wire fall away from the workpiece. This will not only damage the tank bottom but will also lead to a large amount of additional secondary waste.

The maximum plate thickness that can be cut by a single power source and a 3mm diameter wire is 100mm. This is the upper limit. Cutting more than 100mm requires a second power source, connected in parallel and a 4mm diameter wire.

3. Final Tests at Cadarache

Setting up the Cutting Tools in the Pegase Facility

Setting up the CAMC and consumable electrode tools from Hanover required transporting a power supply station, a pump and a control unit to Cadarache. In order to reduce tool changing times, the equipment was optimized by using, on the one hand, a common direct current supply with water-cooled cables in order to reduce their cross-section, and, on the other hand, a special flange to fix the tools onto the manipulator.

Setting-up the Apparatus for Secondary Emission Measurement

Let us recall that the purpose of these measurements is to make a material balance of the solid emissions (sedimented drosses, suspended particles, attached slags, aerosols) and the gaseous emissions resulting from the underwater cutting of workpieces representative of nuclear installation dismantling. The apparatus is illustrated in Figure 5.

Progress of the tests

Ten experiments have been carried out. Their main characteristics are given in Table I. Seven non-radioactive plates and three radioactive plates have been cut.

Table I : Main features of the experiments

N° of experiment	Tool	Place	Material	Material thichness (mm)	Radioactivity	Water depth (m)	Cutting speed (mm/min)
1	Consumable electrode	underwater	stainless steel	80	NO	0.6-0.85	110
2	Consumable electrode	underwater	stainless steel	80	NO	1.65-1.90	110
3	CAMC	underwater	stainless steel	80	NO	0.4-0.9	45
4	CAMC	underwater	stainless steel	80	NO	1.4-1.8	45
5	Plasma torch	underwater	stainless steel	80	NO	0.47	150
5 Bis	Plasma torch	underwater	stainless steel	80	NO	0.56	150
6	Plasma torch	underwater	stainless steel	80	NO	1.93	150
7	CAMC	underwater	mild steel	32	YES	1.50-2.17	50-100
8	CAMC	underwater	mild steel	32	YES	1.03-1.93	50-100
9	Plasma torch	underwater	mild steel	32	YES	2.05	300

The radioactive plates were taken from the outlet pipe of the primary coolant circuit of the G2 reactor at Marcoule. This gas-cooled type, 40 MWe reactor was put into operation in 1958 and shut down on the 31/01/1980. Dismantling of the reactor is still in progress. The first results of the analyses made on the plates revealed an activity of about 70Bq/g of Co 60 and 2Bq/g of Cs 137; other analyses are in progress.

Preliminary Test Results

Although all the measurements have not yet been processed, the following indications can already be given:

Wear of the Tool: this phenomenon led, on the one hand, to worn parts having to be replaced after a certain period of use, and, on the other hand, to additional waste being generated. For the plasma torch, very slight wear of the cutting nozzle was noted (less than 0.1g/m of cutting length). For the CAMC tool, the graphite electrode wore down more quickly; this had also been noted in the Hanover laboratory tests. It therefore seems necessary to find means of slowing down this wear in order to cut acceptable lengths with the same electrode. For the consumable electrode, the consumption of the 3mm diameter wire was considerable, about 15kg/m of cutting length. This represents an additional production of waste due to the tool that is a drawback for dismantling operations.

Waste from the Workpiece: the consumable electrode generates about 3 times less waste than the other tools, as the kerf is considerably narrower. Finally, the plasma torch generates 15 to 20% more waste than the CAMC tool.

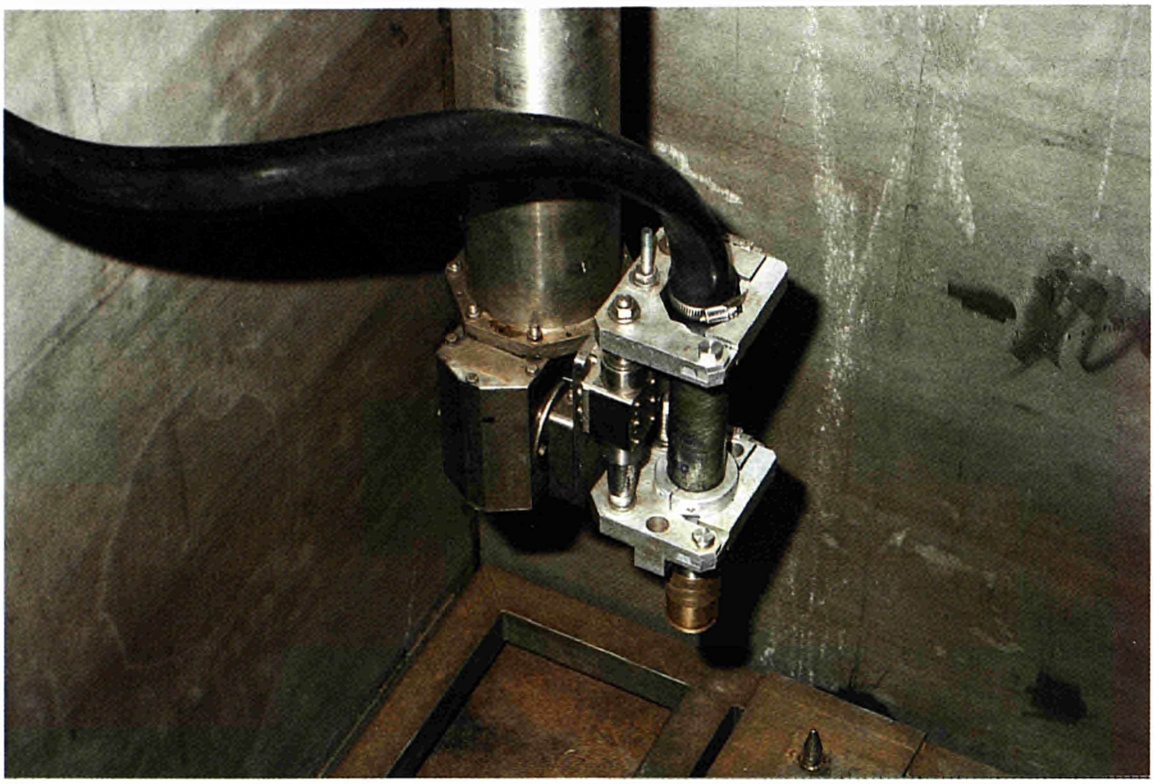

Figure 1 : The L'TEC plasma torch in the empty tank

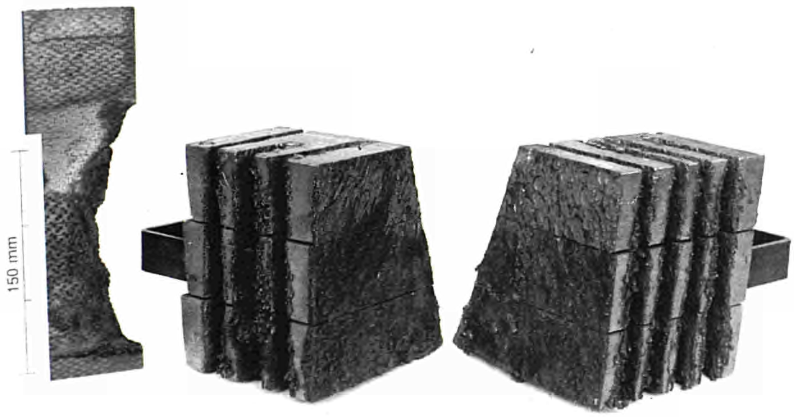

Figure 2 : Result of cutting 150mm of stainless steel plate with the CAMC tool

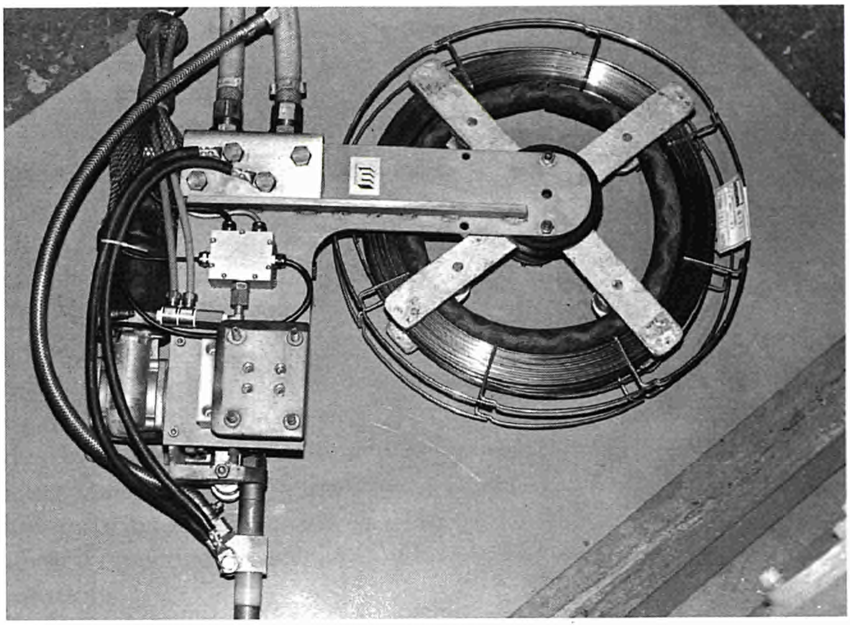

Figure 3 : Consumable electrode tool ready for use in the Pegase facility

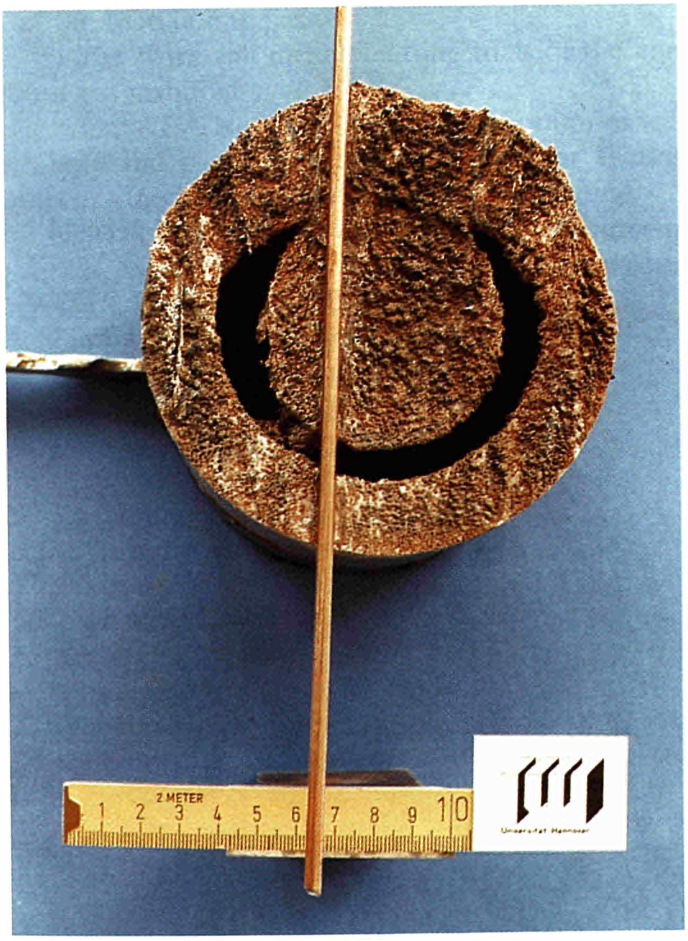

Figure 4 : Result of cutting the "tube" model with the consumable electrode tool

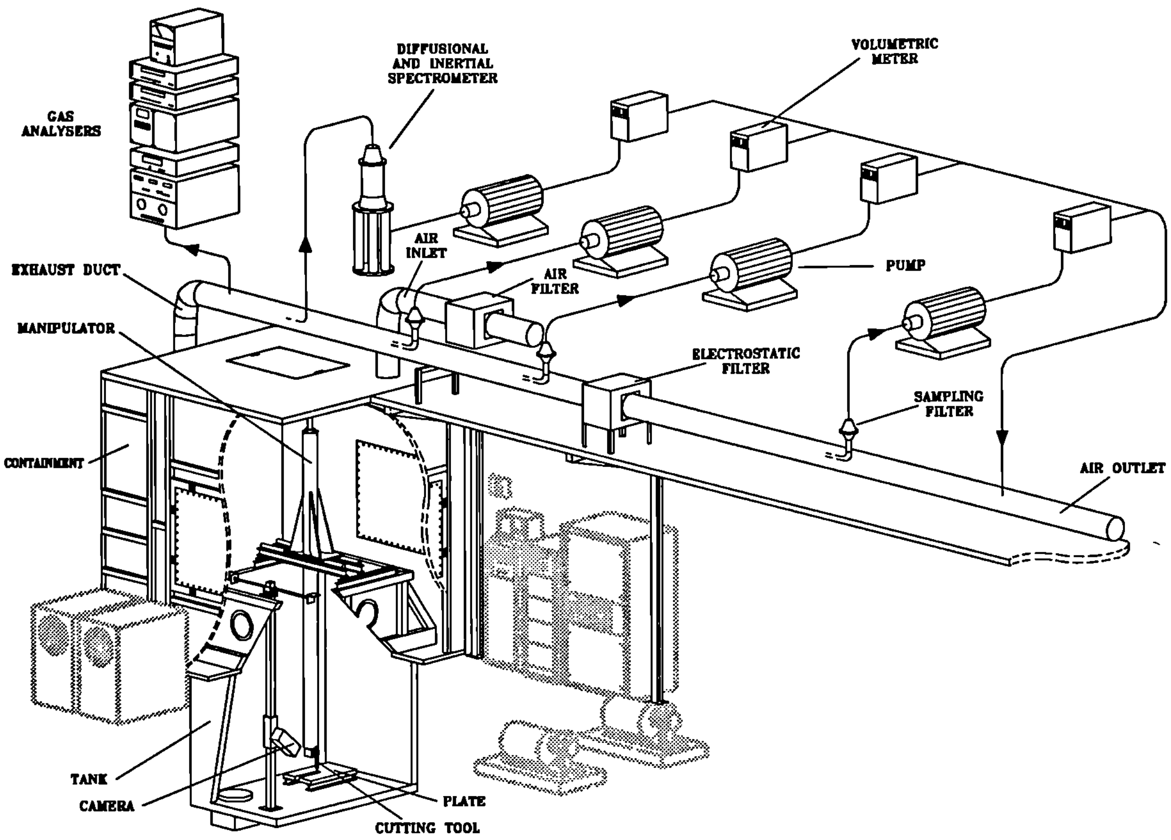

Figure 5 : Overall view of the experimental device for gas analysis and aerosol measurement.

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

Contractors:	AEA Wind
Contract No.:	FI2D-0026
Work Period:	July 1990 - June 1993
Project Manager:	S J WHITE, AEA Wind.
	Phone: 44/9467/72432 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 RESULTS OBTAINED

Summary of main issues

During 1993, the work has been primarily directed towards the development of the algorithm and the recording of additional cutting sequences in a non–active environment. The version 1.0 algorithm was due to be tested during this period, however, technical difficulties prevented extensive testing to be undertaken.

Progress and results

1. Transducer development (B.4)

The second plasma cutting torch, complete with built–in transducer has been to perform a series of additional cuts. The data from these cuts have been captured using a digital system compared to the previous analogue method. The advantages of the digital system is that the sound quality is improved at source capture, also with the digital system data can be manipulated easily for detailed analysis. A third advantage is that the data can be replayed in full 16 bit sound quality, reducing the number of real cuts required. Work is being undertaken to record and achieve acoustic signals for a range of cutting conditions. These signals can then be used for subsequent analysis if required following the termination of the existing contract.

2. Algorithm development (B4.3)

As already stated due to technical problems in setting the equipment up at Windscale, it has not been possible to undertake any trials with the first version of the algorithm. It is currently expected that such trials may take place during the first quarter of 1994. Due to major reorganisations within the company, difficulties have also been encountered in retaining the expertise on the project; although not a major disruption to this project, it has been a contributory factor to the delays.

3. Control system implementation (B4.4)

The industrial PC used to host the system has been extensively tested and proved extremely effective. Trials relating to HF interface have also been undertaken and proved satisfactory. Due to the timescales, it will now not be possible to complete the final phase of this work under the EC framework.

4. Preliminary tests of the deployment system (B.5)

Preliminary trials on the deployment system have been completed and found to perform satisfactorily. Tests were undertaken on the 4–axis deployment system together with the remote viewing system. These tests confirmed the success of the changes undertaken to reduce HF interference on the control system.

5. Full–size testing of the deployment system (B.6)

Full scale mock–up trials were undertaken in the HERO non–active test facility at Windscale. A number of remote cuts were performed on full scale mock–ups of the stay–tubes, complete with stainless steel insulation. Arc initiation and auto reverse features were tested; this is a system to avoid excessive strain on cables through a constant coiling action.

6. Final evaluation including specific data (B.7)

Data recordings were taken for a number of costs and the acoustic signal analysed for a signature frequency. It was soon established that the 300Hz signal could be used to monitor the success or not of the cut. A number of additional costs were undertaken for a range of cutting parameters including unsuccessful penetration. The findings of the tests will be reported in the final report due in January 1994.

D CONCLUSIONS

The final stage of this EC project has been subjected to a number of setbacks both in technical and non-technical areas. 1993 has proved to be a particularly difficult period in respect to maintaining the exceptional rate of progress demonstrated in previous reporting periods. It should, however, be stated that the overall project has been considered by AEA to be a success. The project successfully developed a small bore plasma torch complete with remote deployment system. In addition, several fundamental problems were overcome in the process, particularly those relating to interface and HF shielding.

Having successfully developed the cutting system and its associated deployment system, the equipment was used on actual decommissioning operations on the Windscale Advanced Gas Cooled Reactor. Operational changes resulted in the system deployment taking place some 18 months ahead of schedule. Extensive operational data was collected during this period together with additional cutting data and made available to EC decommissioning data bases. Modifications and improvements were undertaken as a result. The overall stay-tube cutting programme was completed in August 1991 with a total of 100 stay-tubes and 30 flux scanning tubes without major problems. Significant reductions in dose budget were also achieved with remote deployment and extended nozzle life.

Two remotely deployed inspection cameras were also developed and tested during the programme and used in conjunction with the X-Y-Z deployment system. The EC project was also successful in the development of a robust acoustic transducer which was capable of withstanding the harsh environment and high temperatures associated with plasma arc cutting processes. Using this microphone, it has been possible to identify key acoustic signatures on which to develop the acoustic monitoring algorithm. The developed prototype algorithm for the system would enable (if successful) the remote monitoring of a cut performance and eventually the control of one or more key process parameters.

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

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

A. OBJECTIVE AND SCOPE

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

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

B. WORK PROGRAMME

B.1. Development of a high resistance diamond pearl
B.2. Development of suitable materials for the cable vulcanisation
B.3. Preparation of the test mock-up and of a representative concrete block
B.4. Selection and improvement of a suitable steel cable
B.5. Assembling of cable components (light steel cable, diamond pearls, springs and spacers)
B.6. Cutting tests on non-contaminated concrete structures
B.7. Final evaluation, taking into account cable consumption, costs of the technique, cooling water consumption, secondary waste arisings, radiation exposure, and a comparison with the cutting work carried out in the USSR.

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in July 1992. The final report as EUR-report will not be published.

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

Contractors: AEA Culh., DLR Stuttgart
Contract No.: FI2D-0028
Work Period: September 1990 - November 1992
Coordinator: J H MEGAW, AEA Culham.
Phone: 44/235/464 215 Fax: 44/235/464 138

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

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

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

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

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in November 1992. The final report is available as EUR Report No. 15649.

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

Contractors: CEA Valrhô, COMEX, EPC
Contract No.: FI2D-0036
Work Period: September 1990 - December 1993
Coordinator: G IMBARD, CEA/DCC/UDIN
Phone: 33/66 79 63 10 Fax: 33/66 79 64 22

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 contractual programme was terminated in 1993:

- the safety study concerning radioactive testing of explosive decontamination and cut up methods on the G2/G3 reactor cooling system pipes was completed;
- authorizations were secured to proceed with testing;
- safety equipment (explosion-proof door) and test equipment (handling, ventilation, radiation shielding, firing stand) were installed in a suitable portion of the G2 reactor;
- 37 explosive tests were conducted, including 32 on contaminated pipes.

The test results were very encouraging for cutting 13 mm thick ferritic steel pipes using Nitroroc charges confined within a dihedral double steel and annular PVC envelope.

However, the decontamination factors obtained were variable but extremely limited, ranging from 1.06 to 1.77 under normal operating conditions in G2.

Progress and Results

1. Qualification Tests on G2/G3 Pipes (B.5.)

1.1. Specification of Test Procedure for Approval by Safety Authorities (B.5.1.)

Safety assessments were carried out by CEA/UDIN to secure the necessary approval by safety authorities for testing in the G2 reactor building at Marcoule. The following aspects were investigated:

- the test site: for both technical and safety reasons, the interim fuel storage room on the underground level of G2 was selected;
- calculation of the mechanical strength of the test room and its environment: the maximum permissible pressure in the room was 389 mbars, limiting the explosive charge to 530 g of TNT equivalent;
- risk inventory during testing (explosion, noise, blast effect, consequences on structures, toxic fumes, radiological hazard, etc.) and preventive measures required;
- installation of mobile equipment in the test room (position, mounting);
- test procedure and work schedule.

Approval by the safety authorities to proceed with testing was granted on September 16, 1993.

1.2. Preparation of the Test Area (B.5.2.)

After installing the equipment required for the explosive tests (explosion-proof door, handling and hoisting equipment, firing stand), five preliminary tests were conducted on a non-radioactive pipe section using Pentrite detonating fuses with increasing charges from 50 to 390 grams (equivalent to 530 g of TNT).

The resulting overpressure in the test zone corresponded to the predicted values and the tests were completed without difficulty. However, because a high-frequency (1000–2000 Hz) vibration was measured between 8 and 30 msec after the explosion (see test recording in Figure 1), it was decided to limit the test charges to 220 g of Pentrite during the initial radioactive decontamination tests (i.e. one full turn of a 70 g/m fuse around a 1000 mm diameter pipe).

1.3. Qualification Testing (B.5.3.)

Qualification tests were conducted on 500, 800 and 1000 mm diameter G2/G3 pipes contaminated internally with ^{60}Co to between 100 and 200 Bq/cm^2.

Thirty-two tests were carried out under radioactive conditions: 19 decontamination tests using Pentrite detonating fuses, and 13 cutting tests using moulded, confined Nitroroc charges.

Decontamination Tests

These tests were performed on pipes of the three diameters mentioned above, with Pentrite detonating fuses containing charges of 40 g/m and 70 g/m (see photo in Figure 2).

The decontamination results by the mechanical action of the shock wave in the G2/G3 reactor cooling pipes were very disappointing. With identical operating conditions (i.e. same initial activity, same linear explosive charge, same procedure), variable decontamination factors were obtained (ranging from 1.03 to 1.48) for the first test. The decontamination effectiveness of the succeeding tests diminished rapidly, and was nil for the 5th test on the same pipe. The final decontamination factors were very low in every case (not exceeding 1.77).

Cutting Tests

Cutting tests were performed only on 13 mm thick, 800 mm diameter pipes with 450 g Nitroroc charges moulded to match the outside diameter (see photo in Figure 3).

Good cutting results were obtained: the cut was clean and uniform, with a limited cutting width and relatively little projection of particles inside the pipe (see photo in Figure 4).

The decontamination factors were comparable to those obtained using Pentrite fuses (1.14 to 1.70).

1.4. Final Evaluation (B.6.)

The use of explosive cutting techniques to dismantle G2/G3 reactor cooling pipes has a number of advantages:

- the short time required to install the charges (a few minutes) limits the occupational doses (the charges may be detonated from outside the irradiation zone);
- operators are protected from contamination by the barriers constituted by the pipe itself (the charges are placed on the outer circumference).
- the process generates only very limited quantities of secondary waste.

However, these advantages are largely offset by major drawbacks for in situ operation:

- damaging mechanical effects on the environment, requiring protective shielding;
- propagation of a shock wave throughout the structures and the pipes themselves, with possible detrimental consequences on the supporting structure;
- major safety precautions are required.

In any event, the decontamination effect of the explosions may be considered negligible, at least in the G2/G3 cooling pipes.

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SCALE:  CATEGORY AXIS   -->  1 TICK MARK = 5 ms
        VALVE AXIS      -->  1 TICK MARK = 200 mbar
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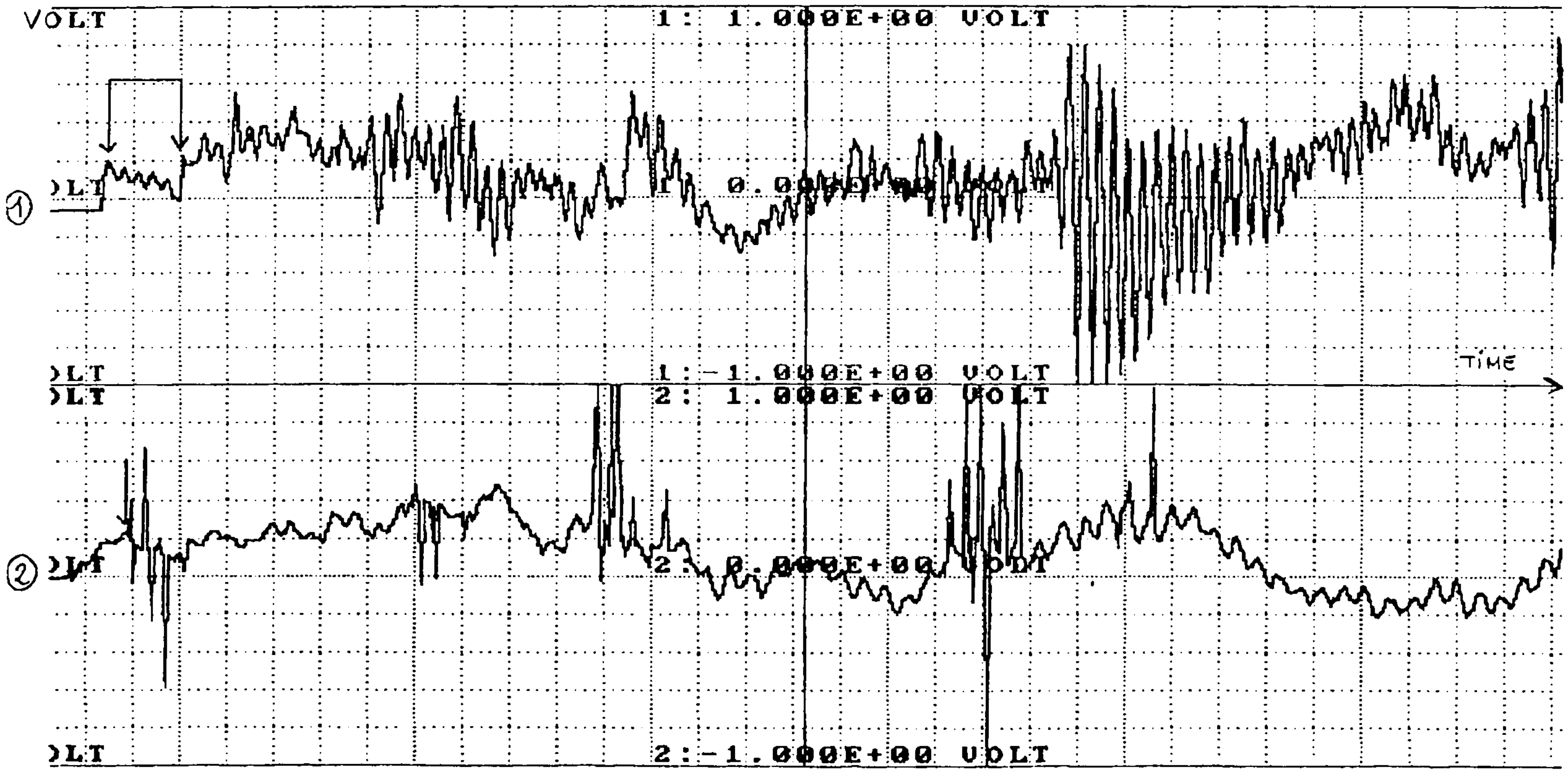

Figure 1 : Pressure measurements during G2/G3 preliminary tests

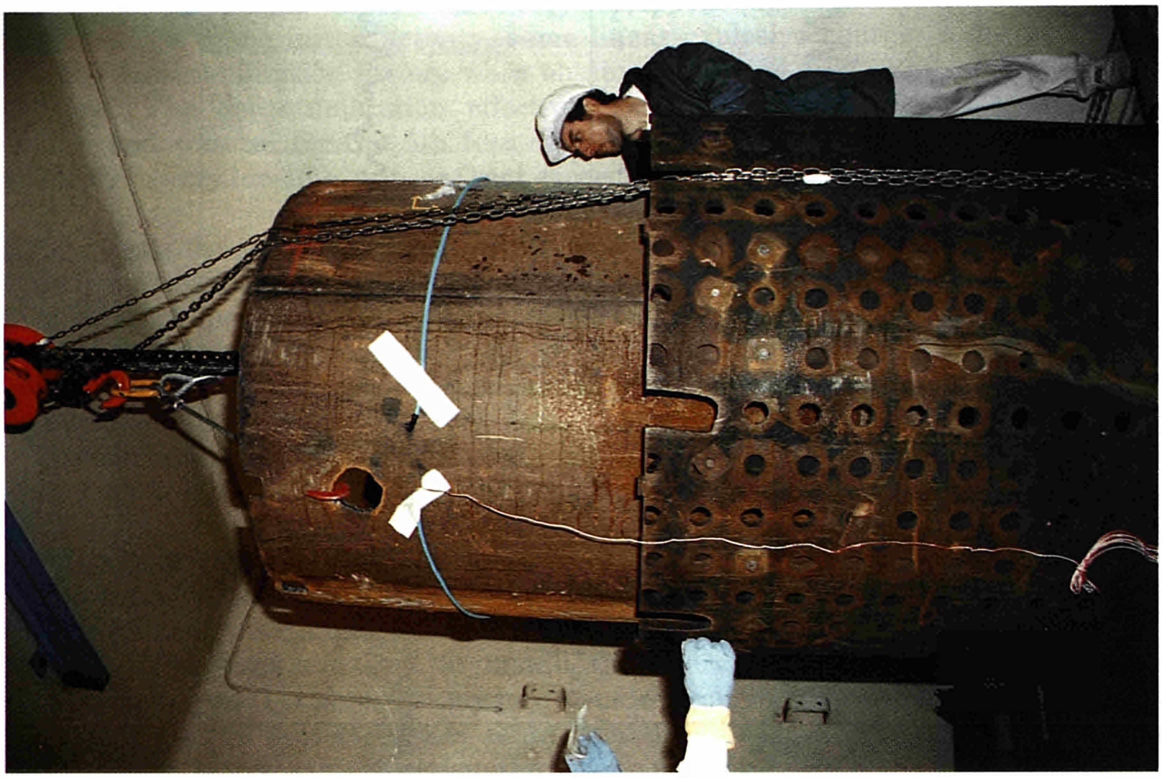

Figure 2 : Decontamination test with pentrite explosive

Figure 3 : Cutting tests with Nitroroc explosive

Figure 4 : Pipe after cutting

3.10. UNDERWATER LASER CUTTING OF METAL STRUCTURES

Contractors: CEA-FAR, Radius
Contract No.: FI2D-0047
Work Period: January 1991 - June 1994
Coordinator: D de PRUNELE, CEA-FAR
Phone: 33/1/69 08 23 51 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) in air and under various water depths on non-active stainless steel plates, and evaluations of the effluents' generation.

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. Determination of the parameters for cutting in air (10 to 40 mm thicknesses)
B.7. Determination of the parameters for underwater cutting (10 to 30 mm thickness)
B.8. Evaluation of effluents' generation (cutting in air)
B.9. Evaluation of effluents' generation (underwater cutting, 0.5 to 7 metres water)
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

Some stainless steel plates (304L) with various thicknesses (10 to 31 mm) were cut underwater and in air with a CO_2 laser (2.5 kW on sample). The secondary emissions in gaseous form (O_2, H_2, NO, NO_x,O_3) and in solid form (sedimented drosses, suspended particles, aerosols) were quantified and characterised. The influence of some parameters like cutting place, water depth and cutting speed was studied.

PROGRESS AND RESULTS

Evaluation of effluents' generation in air and underwater (B.8.,B.9.)

The 5 kW CO_2 laser of STA laboratory was used with a power of 2.5 kW. A 0.37 m^3 experimental vessel with a column of water, a ventilation system with associated samplings (Fig.1) were installed in order to achieve the cuts and to measure the secondary emissions on stainless steel plates (304L).

Table I indicates all the experiments carried out. The cutting parameters during the tests were the following :

- Laser power on the sample : 2.5 kW,
- Focal length : 250 mm,
- Focus position relative to the workpiece : + 13 mm,
- Nozzle geometry : 1 x 4 mm^2,
- Oxygen mass flow rate and pressure : 80 l/min and 4 bar,
- Distance between nozzle and workpiece : 1 mm.

1. Aerosols results

The aerosol mass which represents 0.08 % to 1 % of the total mass collected decreases according to the water depth (Fig.2) and increases with the plate thickness and particularly for e = 31 mm, which is the maximal thickness for the power used (2.5 kW) (Fig.3).

The underwater cutting produces less aerosols than the cutting in air by a factor of about 2.5.

The study of the aerosol size distribution shows that there are two cases :

- the mass mean aerodynamic diameter (MMAD) is about 0.45 μm,
- the size distribution is bimodal with a first mode around 0.07-0.15 μm and a second one, the major part around 0.45 μm.

There is no apparent influence of the water depth and of the thickness on the MMAD.

The aerosols are enriched in chromium (x 2 to 4.5), in nickel (x 4.3) and in manganese (x 2.7).

2. Gas analysis results

The production of gases like NO, NOx and O_3 is relatively small (4.10^{-4}, 6.10^{-4} and 10^{-5} l/min).

The production of hydrogen is not detectable in underwater cutting.

3. Suspended particles results

The mass of suspended particles in water represents 1.3 to 4.5 % of the total mass collected, they increase with the plate thickness (Fig.4) and are relatively homogeneous in the water column.

4. Sedimented dross results

The sedimented dross mass represents 95 % of the total mass collected in the underwater experiments and up to 99 % in the experiments in air. This mass is decreasing for underwater cutting and increases with the plate thickness (Fig.5).

Thus mass-mean diameter of the sedimented dross varies from 0.4 mm to 1 mm for the studied thickness (Fig.6).

There is no significative influence of the water depth.

TEST NUMBER	CUTTING PLACE	PLATE THICKNESS (mm)	WATER DEPTH (m)	CUTTING SPEED (m/min)
1	Underwater	20	7	0.1
2	Underwater	20	0.5	0.1
3	Underwater	20	2	0.1
4	Underwater	20	4	0.1
5	Underwater	20	0.5	0.1
6	Underwater	20	2	0.1
7	Underwater	20	4	0.1
8	Underwater	20	7	0.1
9	Underwater	20	4	0.1
10	Underwater	10	0.5	0.5
11	Underwater	10	7	0.4
12	Underwater	10	0 5	0.4
13	Underwater	31	0.5	0.03
14	Underwater	31	7	0.03
15	Underwater	31	7	0.03
16	In air	20	–	0.3
17	In air	20	–	0.3
18	In air	20	–	0.1
19	In air	31	–	0.13
20	In air	10	–	0.85
21	In air	20	–	0.05

Table I : Summary of cutting conditions

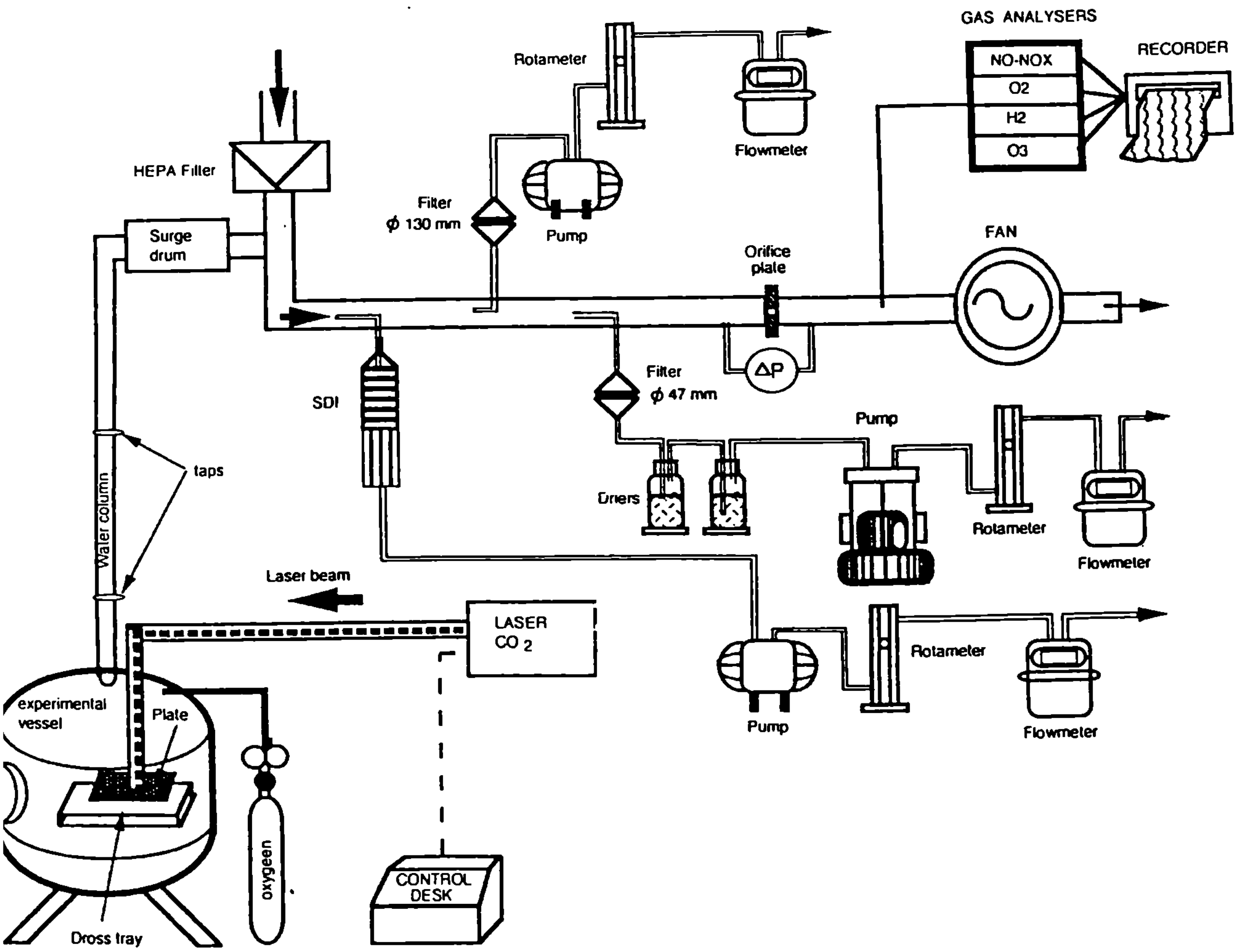

Figure 1 : Diagram of the ventilation and by products measurements

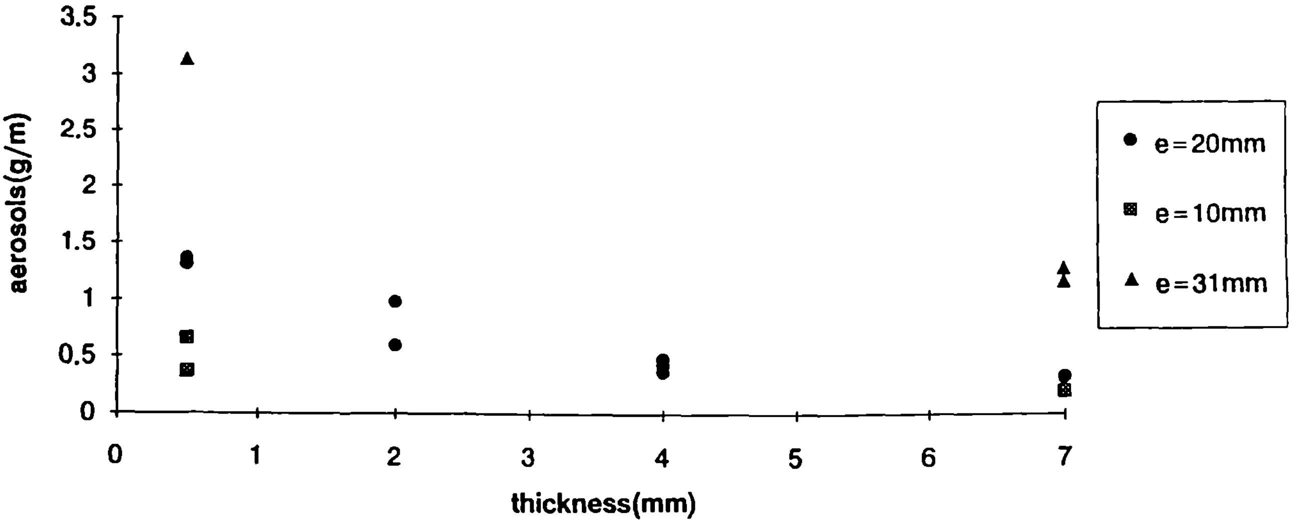

Figure 2 : Variation of the aerosols mass v.s. water depth

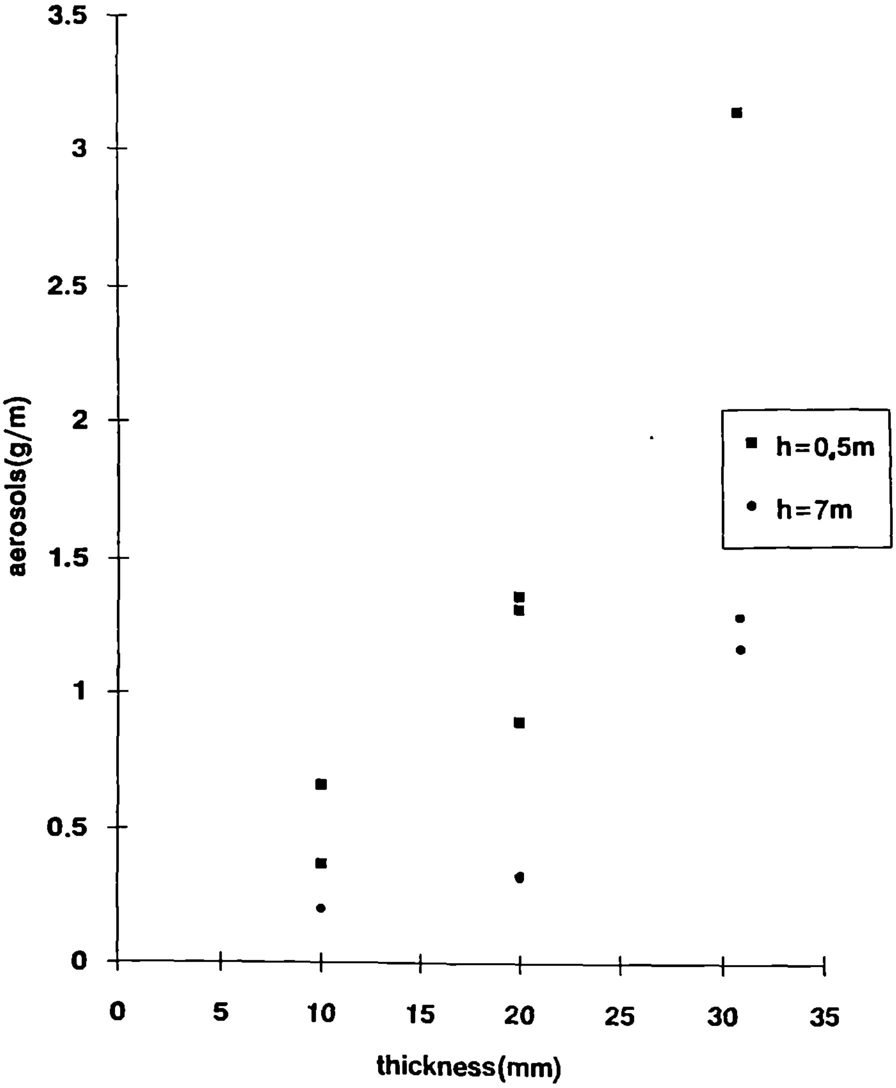

Figure 3 : Variation of the aerosols mass v.s. plate thickness

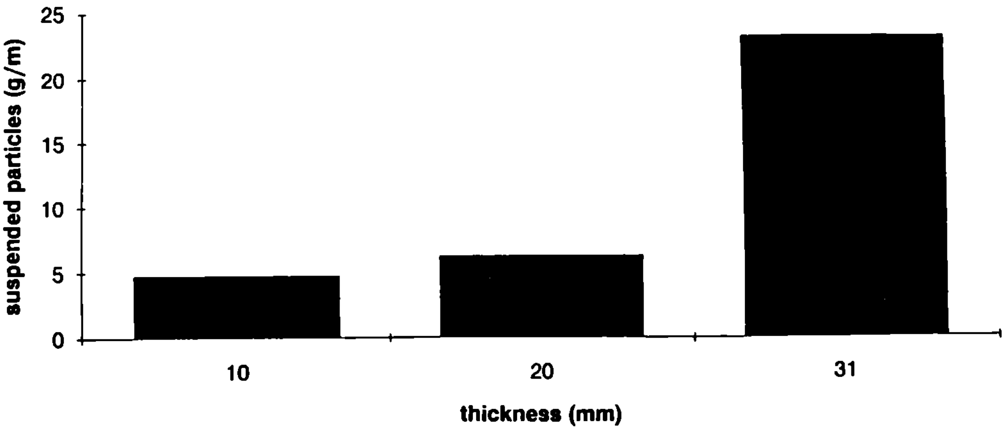

Figure 4 : Variation of the suspended particles v.s. sample thickness

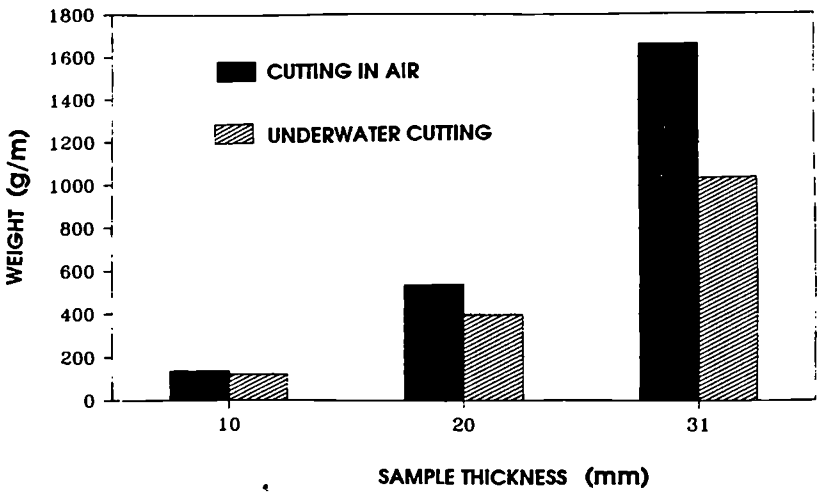

Figure 5 : Sedimented dross mass v.s. plate thickness

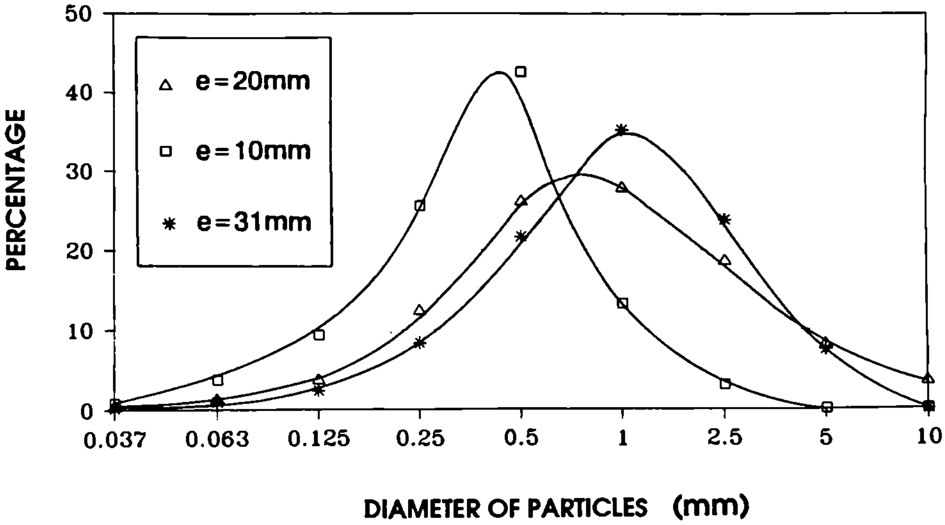

Figure 6 : Sedimented dross size distribution for 10, 20 and 31 mm thickness

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 - September 1993
Coordinator: R S G PARKER, AEA Culham
Phone: 44/235/463 685 Fax: 44/235/464 300

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

B.1. Literature survey to obtain current information on plasma arc torch design, with particular reference to electrode life enhancement.
B.2. Requirements study to identify the necessary or desirable properties and features of the electrode material and the scope of the theoretical work to be carried out.
B.3. Identification of potential electrode materials with specialists of the Harwell Laboratory to select the materials to be included into the test programme.
B.4. Plasma arc process modelling and design of electrode/nozzle.
B.5. Testing existing methods and electrodes in order to allow for comparison with the tests performed on the new electrodes (B.6.).
B.6. Manufacture and testing of selected new electrodes to determine their performance, and comparison with those on the existing electrodes (B.5.).
B.7. Review of test results on the existing and new electrodes, comparison with the theoretical work and modifications to the electrodes to further improve their performance.
B.8. Manufacture and testing of revised electrodes as defined in B.7..
B.9. Final evaluation of new electrode designs with respect to the potential benefits to the dismantling of nuclear facilities, including specific data on costs, waste arisings, working time and related radiation exposure of workforce and working area.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues during 1993.

During the last working period (the contract ended on September 30, 1993), the manufacture and testing of existing and new electrodes continued (B.6., B.8). A review of the test results (B.7) with respect to the final evaluation of new electrodes and nozzle designs (B.9) has been carried out and a draft final report issued.

D. CONCLUSION - FINAL EVALUATION

The research project which looked at ways of improving the life of plasma cutting torches was undertaken by AEA Technology, TWI, Arc Kinetics and Goodwin Air Plasma. The aims of the project were to examine existing and new plasma torch designs and materials, to use plasma modelling to optimize the electrode performance, to test the performance of revised designs and to make recommendations that would lead to the enhancement of the electrode life of plasma cutting torch. The recommendations which can be made at the end of the contract are as follows:

: A review of plasma modelling data had indicated that modelling the physics of the electrode/plasma interface is extremely complex and would require more time than was available for the project. It would be possible to develop simpler models but the model would have no practical value unless it was related to experimental and operational observations.

: The spread of the test results suggested that reproducibility and production methods for the electrode and holder should be examined further. This problem should be resolved before undertaking any further work on the design or alternative materials. The spread also suggests that it should be possible, using optimised production methods, to increase the electrode life by a factor of about two. "*Conditioning*" of the electrode with a low current arc prior to use in production cutting would be worth investigation.

Laboratory tests on electrodes manufactured by an alternative process are recommended, viz pressing holder and insert blanks, drawing and machining the electrode from the bimetallic bar.

: Inconsistency between the test results from individual electrode assemblies implies that the materials were not tested as separate variables but the following conclusions can nevertheless be drawn from the test results:

- ZrB_2 is not a suitable electrode material in the present design of torch.
- Extended inserts protruding directly into the coolant had marginally longer average lives than the standard 6 mm inserts and would be worth further examination.
- Ceramic sprayed electrodes with a bond coat did not perform as well as the standard Hf electrodes and should be eliminated as a candidate design.
- Ceramic sprayed electrodes without "*bond coat*" performed marginally better than the standard electrode if "*first arc*" failures were ignored. The manufacturing methods need to be controlled to eliminate the introduction of other variables and failure modes.
- Ceramic spraying might provide a means of extending the life of a plasma cutting electrode but more rigorous laboratory testing is required to verify this.
- Ru + Y_2O_3 electrodes had average cutting lives about 50 % better than the standard Hf electrodes.

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 (eg 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, eg alpha-contaminated material, and non-ferrous scrap.

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

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

D. Programme implementation

Five research contracts relating to Area No. 4 were concluded, of which four were completed at the end of 1993.

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

Contractors: SG, NIS
Contract No.: FI2D-0014
Work Period: July 1990 - May 1993
Coordinator: D HOLLAND, SG
Phone: 49/2151/894 290 Fax: 49/2151/894 444

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 OBTAINED

This project was completed in 1993. The final report is being prepared for publication.

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 - June 1994
Coordinator: H A W CORNELISSEN, KEMA, Arnhem
Phone: 31/85/56 61 04 Fax: 31/85/51 58 35

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

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

C PROGRESS OF WORK RESULTS OBTAINED
SUMMARY OF MAIN ISSUES

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

KEMA has developed, designed and constructed a test installation for separation of aggregates from contaminated concrete on pilot plant scale. Verification tests at VAK in Kahl (Germany) with contaminated concrete showed a volume reduction of 55%. For contaminated mortar a volume reduction of only 37% was realised mainly because the installation was designed for separating concrete. For mortar also the sand fraction had to be removed.

The work and safety procedures were adequate. The installation and operations were approved by TÜV-Bayern (Germany). No faults, maloperation and dust dispersion did occur during the verification tests. The transport of the installation did not require special permits before and after the verification tests at VAK.

Most of the contamination (80%) that was collected consisted in fine "clean" material (< 1mm). The activity measured on the coarse material (> 1mm) was due to bounded dust and clustered cement stone.

In the laboratories of Taywood Engineering the preliminary immobilisation trials were continued to determine the most suitable formulations for large scale immobilisation testing of active cement waste generated by the KEMA test installation. The results indicate that the most practical volume reduction of waste in each mix is 55-60%.

Preliminary experiments to compare recycled aggregates with standard aggregates in concrete and the increase of volume reduction by cleaning and sieving were performed. No significant decrease in mechanical properties of the concrete prepared with the recycled aggregates was found. Further, the contamination level of the coarse fraction can be reduced substantially by removing the cement dust layer by cleaning.

Progress and results

1 Verification tests with "clean" concrete from VAK (B.5.)

Based on the results of the test runs with the installation, first the parameters for thermal treatment were determined on "clean" (not-contaminated) concrete from VAK. The concrete, from the disposal of VAK, was characterized in the KEMA laboratory.

The visual determination result of concrete debris showed that the concrete components partly consist of porous aggregate. These were characterized as red coloured sandstone. The sandstone aggregate has a higher porosity compared with quartz aggregate and has a lower strength and specific gravity which explains the relatively low compressive strength.

Series of tests with "clean" concrete from VAK were performed. The results from the testing are given in table 1.

Table 1 Results of testing of "clean" concrete from VAK

input concrete	210 kg	
after heating	198 kg	
after sieving: < 1 mm	96 kg	46%
> 1 mm	102 kg	48%
loss of moisture	12 kg	6%

The results from the tests indicated a somewhat lower separation efficiency with VAK concrete [1]. It was concluded that by separating concrete with sandstone aggregates, the produced waste concrete was approximately 46% (fine material). The higher rate (46% instead of 35% for standard concrete) can be explained by the presence of porous aggregates in the VAK concrete.

2 Verification tests with contaminated concrete at VAK (B.6.)

Five test runs were performed for the verification at VAK in Kahl, Germany. Two concrete and three mortar samples were processed in the installation. At the end of the process the dense coarse aggregate ("clean", > 1 mm) was separated from the fine porous cement stone (radwaste, < 1 mm) and both were examined on activity and mass.

The fine fraction is a mixture of cement and small sand particles. This fraction is contaminated and has to be conditioned and subsequently stored. The coarse fraction largely consists of "clean" dense aggregates, like sand and gravel. This material is released and can be re-used. The tables 2 and 3 show the mass and activity levels of the output material (coarse and fine fraction) of each run. In table 2, the results of the separation of the original concrete (run 1 and 2) and the mortar (run 3, 4 and 5) are presented. In the last column the volume reduction factor (output/input ratio in percentage) is given. From table 2, it can be seen that there is a difference in volume reduction between the first two and the last three runs. This is explained by the composition differences of concrete and mortar. The separation of concretes (run 1 and 2) confirms previous results in which volume reductions between 50 and 65% were realized.

Table 2 Results of concrete separation on contaminated concrete and mortar

test	coarse material (> 1 mm) clean fraction		fine material (< 1 mm) radwaste fraction	
	(kg)	(%)	(kg)	(%)
run 1 concrete	12.70	54.9	10.45	45.1
run 2 concrete	10.00	55.6	8.00	44.4
run 3 mortar	5.95	35.5	10.80	64.5
run 4 mortar	7.35	37.5	12.25	62.5
run 5 mortar	8.95	37.2	15.10	62.8

Remark: * mass reduction in percentage of total sieved fractions

Table 3 shows that the activity of the fine material is higher than in the coarse material. For mortar the activity is higher (test runs 3, 4 and 5) and for concrete lower in comparison with runs 1 and 2. For concrete and mortar 80% of the activity is accumulated in the fine material as well. For concrete most of the activity is accumulated in 45% and for mortar in 63% of the total volume. Further can be seen that still some activity is measured on the coarse material. The following possible reasons are:

- the natural activity of the material
- visual inspection of the coarse material shows a thin layer of dust and some cement stone remained on the surface of the material
- the concrete contained some porous sandstone. As the adherence of cement and sandstone is much stronger than that of cement and quartz sand/gravel, it was more difficult to remove the cementstone
- the process parameters used during the tests were optimized for concrete separation. A further process step for separation of mortar is necessary.

Table 3 Contamination-level in the coarse and the fine fraction after processing

test	coarse material (> 1 mm) coarse fraction		fine material (< 1 mm) fine fraction		fraction of total activity	
	specific activity (Bq/g)	total activity (Bq)	specific activity (Bq/g)	total activity (Bq)	in coarse material (> 1 mm) (%)	in fine material (< 1 mm) (%)
run 1 concrete	0.52	6.60	4.85	50.68	11.5	88.5
run 2 concrete	0.42	4.20	1.87	14.96	21.9	78.1
run 3 mortar	8.28	49.27	32.00	345.60	12.5	87.5
run 4 mortar	4.77	35.06	16.20	198.45	15.0	85.0
run 5 mortar	3.43	30.70	13.30	200.83	13.3	86.7

3 Solidification of the concrete residue (B.7.)

Methods for the solidification of the fine fraction (rad-waste) were studied by Taywood. The immobilisation systems were based either on sodium silicate or cementious grouts [2, 3].

For larger scale tests a series of trial mixtures were made to identify the two most suitable formulations. The mixtures were characterized on the basis of optimal combination as: initial fluid properties, setting behaviour and strength.

The experiments indicated that the most likely practical volume of waste in each mixture was 55 - 60% of the total waste volume (separated material) [3]. Based on these test results from OPC (ordinary portland cement) and water grouts, it can be seen that a grout with a water to cement ration (W/C - ratio) of 0.7 will bind up to ± 60% waste by volume. A mixture of grinder granulated blastfurnace slag (GGBS), OPC, waste and water with a W/C - ratio of 0,8 also will bind up ± 60% waste by volume. The resulting grout flows fairly well and produces a solid mix with satisfactory strength after 28 days.

The available data were used to calculated the actual efficiency of concrete reduction. The calculation is shown in table 4. It can be seen that 1400 kg input concrete produces 490 kg waste (efficiency = 65%). In order to solidify this waste 354 kg cement and 247 kg water is necessary. This resulted in a final volume reduction of about 50%.

Table 4 Efficiency of concrete reduction

	by volume	by mass
Input concrete parts	1000 dm^3	1400 kg
Output waste *	408 dm^3	490 kg
Solidification	408 dm^3	490 kg
OPC-binder	295 dm^3	354 kg
Water	247 dm^3	247 kg
	950 dm^3	1091 kg
After compaction	543 dm^3	1091 kg

Note: * sand and cement < 1 mm

4 Evaluation of the results with respect to equipment (B.8.)

4.1 Radiological risks

A preliminary assessment of the radiological risks to workers and the calculation of the worst consequence to the public was undertaken by AEA Technology [4].

An assessment was made of the radiological consequences using techniques as HAZAP (Hazard and Operability) level 1, FMEA (Failure and Modes Effect Analysis) and an assumed activity level for the concrete of 15 Bq/g with the major nuclide of Co^{60}.

The results indicate that for contaminated concrete up to 15 Bq/g Co^{60} the radiological risks to workers and public is very small during operations with the test installation. This results prove the suitability of the installation for verification with contaminated concrete [5].

The operations at VAK took place in a controlled area in which all necessary provisions and equipments were available.

During processing the workers were equipped with a personal dosimeter and films from VAK and KEMA to measure the radiation doses (in mSv). Masks were used to eliminate any possible hazard due to inhaling of contaminated concrete dust. Continuous aerosol samples were taken from the ventilation out-let and analyzed in the laboratory of VAK. Also, during the thermal treatment of the contaminated concrete the flow extraction of the ventilation system was measured in the furnace by means of a pressure gauge.

The work and safety procedures were adequate [6]. The installation and operation were approved by TÜV-Bayern. No faults, maloperation and dust dispersion did occur during the verification tests. The total personal doses measured after the testing operation and cleaning did not exceed the 0,027 mSv.

4.2 Work step and process control procedures

From the results of identification of possible accidents and maloperations, working experience, dust dispersion measurements and the cleaning (dismantling) of the installation, KEMA has decided to revise the work procedures and instructions [5, 6].

One of the major changes in the work step procedures was the use of the ventilation system. Instead of extracting dust from the components after processing for 5 minutes, the ventilation system was now running during connection and disconnection of the containers. Further process control issues and cleaning procedures were added.

4.3 Transport and release activities

After signing a document which contained the basis of the co-operation between VAK and KEMA, the installation was transported to VAK. Before the installation was transported, a surface contamination measurement was performed at KEMA by the health physics department. The installation was declared free of contamination (below the required Dutch limits of 3,7 Bq/cm^2 for Beta-gamma).

The transport of the installation after the verification tests back to KEMA, was implemented in October, 1993. The installation was cleaned by workers of VAK and contamination measurements were performed by the health physic experts of VAK. No significant higher contamination was found after processing and after cleaning. The installation was found free of contamination in accordance to the German release levels for γ and β radiation (0.5 Bq/cm^2).

4.4 Quality of the concrete with recycled aggregate

To be able to compare concrete properties made of standard aggregate and made of recycled coarse ("clean") aggregate, the two different mixtures were investigated [3].

Table 5 Mechanical properties

type of concrete	compressive strength (N/mm²) 3 days	7 days	28 days	E-modulus (N/mm²) after 28 days	slump value* (mm)	spread* (mm)
reference	28.5	35.5	47.6	34063	80	410
recycled	36.1	43.2	55.9	29069	25	320

Remark: * the slump and spread are criterions to characterise the workability of a concrete mixture

From the results in table 5, it can be seen that a higher compressive strength is realised with the separated "clean" aggregate (recycled). This could be due to small dust/cement stone particles which are attached on the coarse material. Further, it was concluded that there is no significant decrease in mechanical properties of the concrete prepared with recycled aggregates.

4.5 Additional cleaning

To verify the dust influence on the contamination level, the coarse fraction (> 1 mm) was treated in an additional process step. The treatment consisted of wiping off the dry gravel surface with a nylon brush followed by sieving with a sieve mesh size of 1 and 0.5 mm. The results are given in table 6.

Table 6 Results cleaning and sieving coarse-fraction (> 1 mm)

test run 3 mortar	original > 1 mm fraction	after cleaning (wiping) and sieving		
		cleaning* > 1 mm	sieving > 0.5 mm ≤ 1 mm	sieving < 0.5 mm
specific activity	8.28 Bq/g	3.70 Bq/g	7.92 Bq/g	33.80 Bq/g
mass	5.95 kg	770 g	39 g	40 g
total activity	49266 Bq	2849 Bq	309 Bq	1353 Bq
mass fraction	100%	90.7%	4.6%	4.7%
activity fraction	100%	63.1%	6.9%	30.0%

Remark: * the remaining activity was measured after wiping the aggregates > 1 mm on sieve with mesh size 1 mm

The table 6 shows that the contamination levels of the three products differ considerably. A good result is achieved after cleaning and additional sieving of the coarse fraction > 1 mm with a sieve mesh size of 0.5 mm. A small reduction in the mass (9.3%) gives an activity reduction of about 37%. From tables 6 and 7 it could be concluded that the major part of the radioactivity is concentrated in particles smaller than 0.5 mm.

Table 7 Results sieving radwaste-fraction (< 1 mm)

test run 3 mortar	original radwaste-fraction(< 1 mm)	after sieving	
		radwaste > 0.5 mm	radwaste < 0.5 mm
specific activity	32.00 Bq/g	9.48 Bq/g	45.30 Bq/g
mass	10.80 kg	43 g	640 g
total activity	345600 Bq	407.6 Bq	28992 Bq
mass fraction	100%	6.3%	93.7%
activity fraction	100%	1.4%	98.6%

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[2] J.J. BLENKIN, "Immobilisation of Active Concrete Debris generated from Decommissioning Nuclear Power Stations" , Taywood Report 1303/91/5782, TAYWOOD 1991.

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4.3. TREATMENT AND CONDITIONING OF RADIOACTIVE GRAPHITE FROM NUCLEAR INSTALLATIONS

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

A. OBJECTIVE AND SCOPE

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

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

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

B. WORK PROGRAMME

B.1. Removal and/or fixation of radionuclides

B.1.1. Investigations on radioactive and inactive sample structure and texture using different analysing techniques. (all)
B.1.2. Testing of appropriate chemical agents on samples with regard to their possible decontamination and/or immobilisation features. (UDA)
B.1.3. Study of radionuclide removal, mainly titium.
B.1.4. Study of radionuclide fixation.
B.1.5. Characterisation of treated samples using methods from subtask B.1.1.

B.2. Metal coating of graphite by ionic deposition.

B.2.1. Characterisation of samples similar to B.1.1. (all)
B.2.2. Performance of process parameter studies for metal coating applications on inactive samples. (all)
B.2.3. Chemical modification of radioactive surfaces. (CIEMAT)
B.2.4. Metallisation of inactive samples. (all)
B.2.5. Metallisation of radioactive samples. (CIEMAT)
B.2.6. Characterisation of the treated samples concerning chemical properties and thickness of the metal layer, porosity of the surface etc. (all)

B.3. Leaching experiments with the metallised specimen. (CIEMAT)

B.4. Assessment of results and conclusions.

C. Progress of work and results obtained

Summary of main issues

In this year, the characterization of irradiated graphite sleeve from Vandellos 1 reactor, separated from the fuel exposed to air (representative of the core pile), the development of composite copper-nickel coatings and the study of radionuclide leaching in different radioactive uncoated graphite and with metallic and cement coating, have been executed. The graphite sleeves separated from the fuel exposed to air have shown a much higher radioactive content than those separated inside the pool. The most important radionuclides detected, are the same in both cases, but in the latter tritium is the most important. The heterogeneous H-3 and Co-60 distribution indicates that a considerable amount of them comes from contamination inside the core.

The thermical desorption tests have shown that most of the H-3 is linked to carbon atoms from the inside of the graphite. Heating in oxygen atmosphere produces volatile molecules (at temperatures below the burning point) with C-14 and hydrogen.

For metallization, a composite nickel-copper coating and a method to measure the efficiency of the coating, making use of CCl_4 absorption, have been developed.

The leaching tests show that dipping irradiated graphite into certain solutions (chlorides, sulfates, carbonates, etc...), modifies the properties of the graphite, increasing the capability to leaching radionuclides.

In order to test the efficiency of the coatings, tests with various coated graphites, including cement, were carried out following ISO-6961 standards. Using 7 cm cement coatings or 40μm metallic coatings, the leaching rates of Co-60 diminish by 97%, and even in some tests with composite coatings it is impossible to detect tritium.

Progress and results

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

Graphite sleeves separated from the fuel exposed to air (not under water inside the storing pool) were examinated. In this way, representative graphite samples were obtained from the core pile.

The study of the radioactive content shows that in this case the most important radionuclide is H-3, followed by Fe-55, C-14, Co-60 and Cs-134 (Figure 1). The analysis of the axial distribution of these radionuclides in the sleeve, using equidistant transversal cuts, show an homogeneous C-14 distribution, as expected, not being so for the rest. This indicates that they are originated mostly by contamination inside the core. The radial analysis shows that, except for H-3, the radionuclides are concentrated on the surface, specially for C-14 and Co-60 (Figure 2).

In order to locate the radioisotopes inside the graphite, a series of controlled thermic desorptions in oxidating and inert atmospheres have been carried out. 6% of the detected H-3, is desorbed at low temperature from the surface layers, 20% is desorbed between 70 and 400° from the intermediate layers, and the rest is located in internal layers. Nevertheless the most internal layers of the graphite (the remaining 38%) do not contain H-3.

As regards carbon, volatile hydrocarbons liberated in inert atmosphere do not contain C-14. On the other hand, below the burning point, there are liberated hydrocarbons in oxidating atmosphere, possibly coming from non-graphitized carbons, that contain C-14 and 20% of the H-3. Between 500 and 625°C great amounts of CO_2, without C-14, probably coming from the coolant were detected .

To sum up, the study of the radionuclides distribution shows that, except for C-14, the origin of the radionuclides is contamination and therefore it is impossible to make a inventory of radionuclides by calculation.

2. Metallic coating of graphite by ionic deposition

In order to develop an adequate metallization method we have focused the study on the following points:

- Improvement of the surface catalyzation to achieve a more homogeneous coating and to eliminate the initial failures.
- Development of a method to measure the coating efficiency.
- Improvement of the coating porosity.

To improve the surface catalization, the graphite was treated with different products (organic solvents, oxidants and surfactants), trying to eliminate the substances that stop the catalyst deposit and to allow it to enter inside the pores. Most of the products have shown to be efficient, but the best results correspond to water soluble organic solvents with high vapor pressure.

To measure the efficiency, several products that penetrated through the metallic coating pores and reacted with or were absorbed by the graphite, were used. The best method has been the weighing of the absorbed CCl_4.

The improvement study of copper coating impermeability was continued in four different ways:

- diminishing the deposition rate controlling the pH.
- diminishing the generation of H_2 aditioning Brucine
- applying composite coatings of copper/nickel.
- different thermical treatments.

Specimens obtained at low rate and with composite coatings have proved to give the better results, the ones obtained with Brucine were too porous and the ones from thermical treatment suffered cracks.

3. Leaching experiments.

The definitive efficiency of the coatings was tested with leaching tests following ISO-6961 standards.

The leaching tests show that when irradiated graphite is dipped into some solutions (clorures, sulfates, carbonates, etc...) its properties are modified and its leaching rate for radionuclides increases for any leaching agent that should be used afterwards (Figure 3). In this way, it could be shown that graphite stored inside the pool has higher water leaching rates than that separated from the fuel exposed to air.

The leaching tests have been carried out, with water and $NiCl_2$. The specimens were manufactured with graphite obtained by the wet and dry way, uncovered and coated with metal or with cement. The metallic coatings were the ones that proved to be better according to the Cl_4C absorption test.

The higher leaching rate in the graphite obtained by the wet way corresponded to Cs-137 followed by Co-60, C-14 and H-3. For the graphite obtained by the dry way, the higher rate corresponded to C0-60 followed by Cs-134 and H-3. The leaching rate diminishes considerably for cement or metal-coated graphite. With a 40 μm composite coating of copper/nickel the leaching rate of Co-60 is 97% smaller than with non-coated and even in some tests with composite coatings it is impossible to detect tritium.

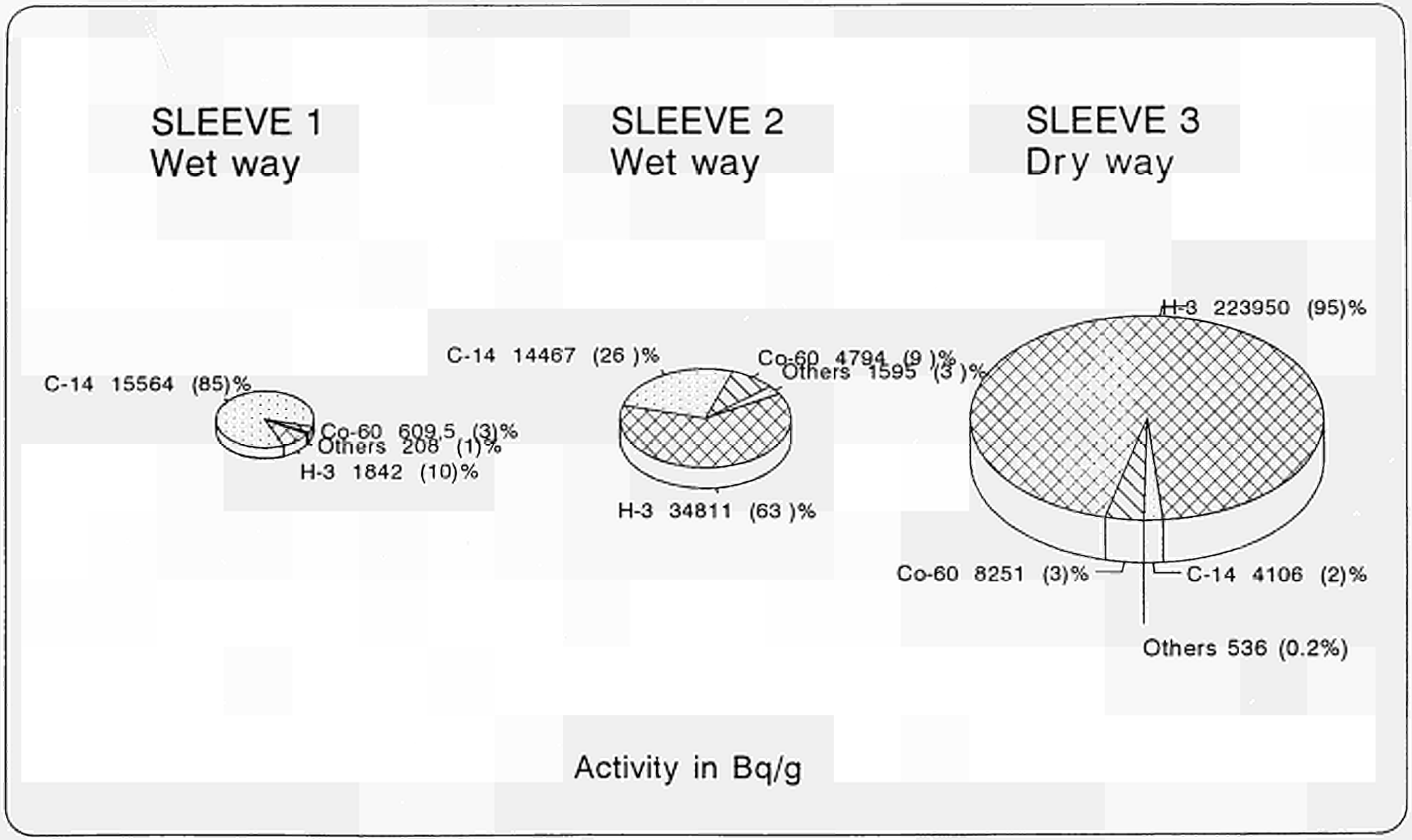
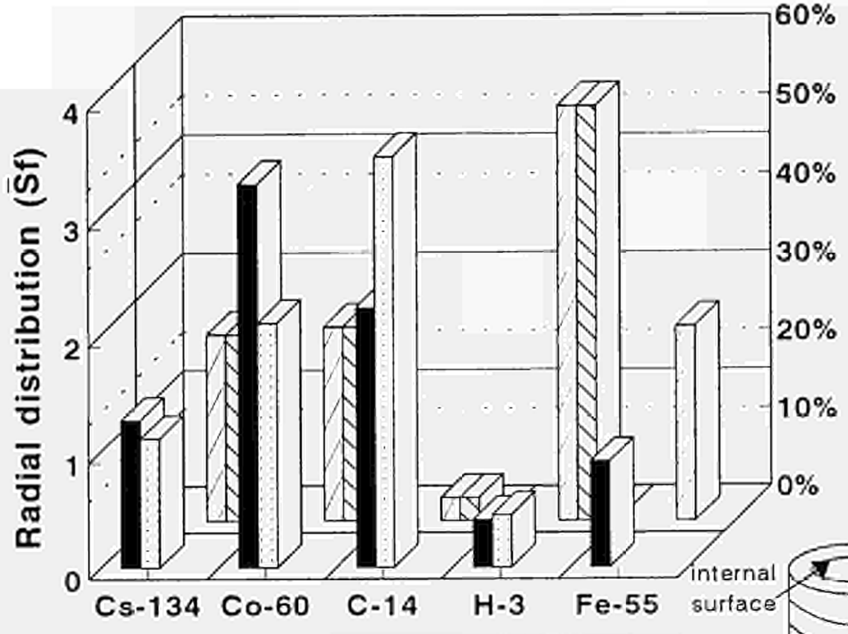


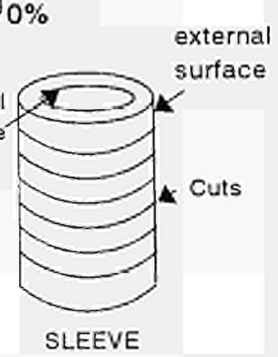
FIGURE 1.- RADIOACTIVE ANALYSIS VANDELLOS 1 GRAPHITE STANDARD ACTIVITY IN SLEEVES N° 1, 2 & 3

$$D = \frac{\sqrt{(Ai/As-\bar{S}f)^2/n}}{\bar{S}f}$$

$$\bar{S}f = \frac{\sum_{i=1}^{n} Ai/As}{n}$$

Where:
D Dispersion coeficient
$\bar{S}f$ Standard surface factor
Ai Activity in cut i
As Activity on surface
i Axial position cut
n Number of cuts





	Cs-134	Co-60	C-14	H-3	Fe-55
Sf in external surface	1,27	3,28	2,23	0,41	0,91
D in external surface	24%	25%	3%	53%	25%
Sf in internal surface	1,11	2,1	3,51	0,45	
D in internal surface	24%	25%	3%	53%	

FIGURE 2.- RADIONUCLIDES AXIAL AND RADIAL DISTRIBUTION SLEEVE Nº3 (DRAY WAY)

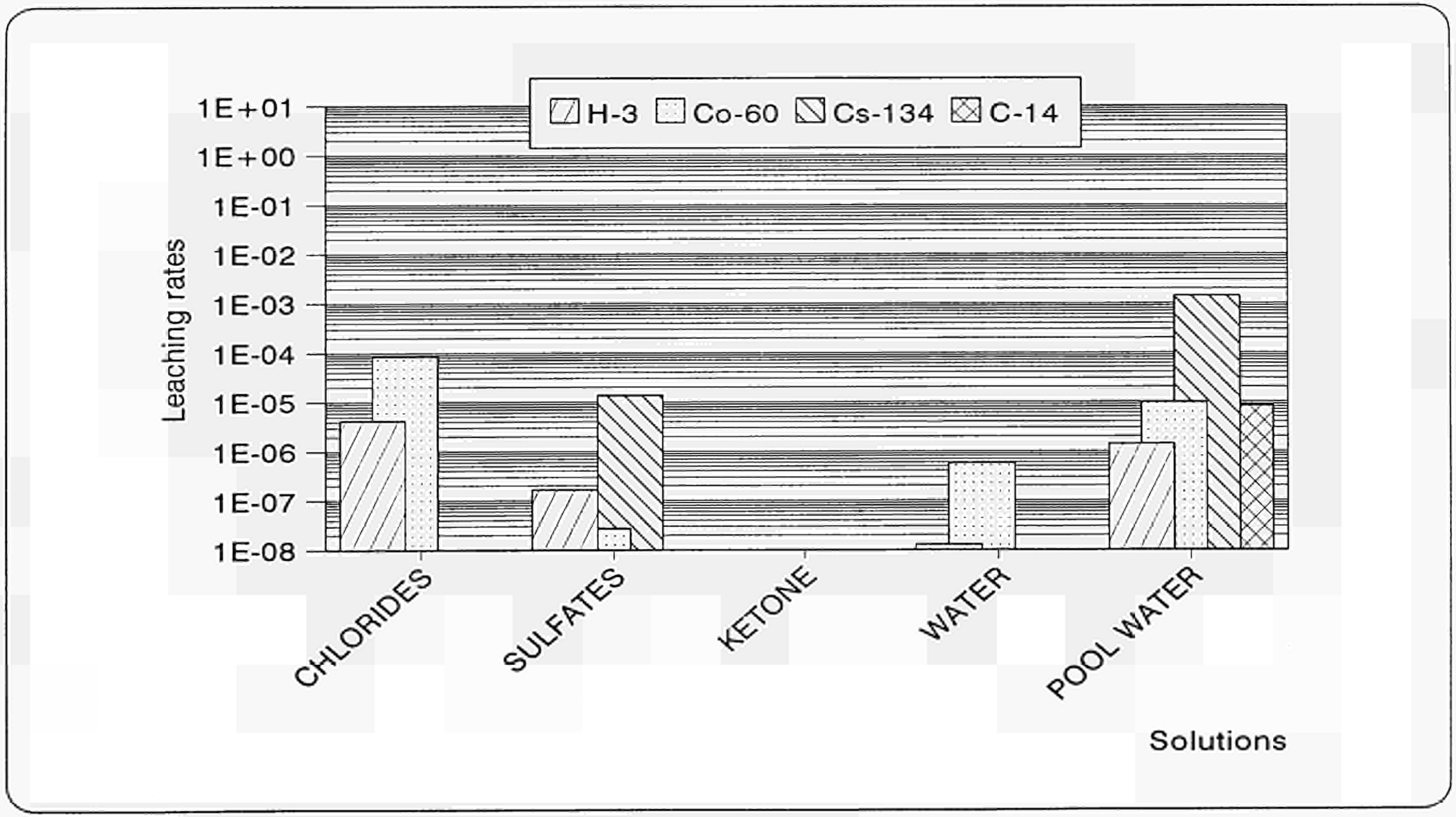


FIGURE 3.- LEACHING INTO SEVERAL LIQUIDS

4.4. RECYCLING OF ACTIVATED/CONTAMINATED REINFORCEMENT METAL IN CONCRETE

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

A. OBJECTIVE AND SCOPE

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

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

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

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

B. WORK PROGRAMME

B.1. Literature study on metal waste types, quantities and activation/contamination levels in order to select potential processes for waste transformation.
B.2. Conduction of a specific test programme on combinations of different metals and metal forms with concrete and mortars.
B.3. Evaluation of the results and survey of possible applications.

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1992. The final report is available as EUR-report 15045.

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

Contractors: Siemens-KWU, SG
Contract No.: FI2D-0037
Work Period: December 1990 - November 1993
Coordinator: K H GRÄBENER, Siemens-KWU
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A. OBJECTIVE AND SCOPE

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

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

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

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

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

B. WORK PROGRAMME

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

C. Progress of work and results obtained

Summary of main issues

In two laboratory experiments carried out using aluminium artificially contaminated with metallic and oxidic uranium, the behaviour of uranium during the melting process was examined. Contrary to cobalt-60, which is easy to remove (decontamination factor = 25), the behaviour of uranium is not yet properly understood.

In industrial-scale melting tests performed by Siempelkamp (SG), 19,390 kg of contaminated aluminium scrap was melted. By adding salts (NaCl, KCl and CaF_2) it was possible to clean the melt of impurities; the radionuclides were fixed in the slag. In the case of approximately 80% of the melted aluminium, the residual activity was < 1 Bq/g (Co-60), meaning that this portion can be released for unrestricted reuse.

In the laboratory, the influence of two different crucible materials - graphite and steel - was also compared. The decontamination efficiency is nearly the same for both materials except that more slag was produced in the steel crucible than in the carbon crucible.

The experimental work has now been completed and the final report is in preparation.

Progress and results

1. Determination of radionuclide distribution by Siemens (B.5.2.)

The scrap used hitherto was mainly contaminated with Co-60 and Cs-137. As it was also desired to study the behaviour of uranium during the melting process, two melting tests were performed using aluminium that had been artificially contaminated with metallic and with oxidic uranium powder. However, most of the powder failed to dissolve in the melt or the slag and formed clumps adhering to the walls of the crucible.

On the basis of the dissolved uranium, it was estimated that a decontamination factor of 1.5 had been achieved. Based on the total amount of uranium that had been added, this would yield a decontamination factor of about 10. More information is therefore needed in order to properly understand the behaviour of uranium during melting.

Table I contains an overview of the results.

2. Melting of aluminium in an industrial furnace by SG (B.6.)

Building on the results of the laboratory tests conducted by Siemens to investigate decontamination of aluminium by melting, Siempelkamp performed corresponding industrial-scale melting tests in 1992. Evaluation of the test results was completed in 1993.

The cleaning agent and purging gas required to ensure optimum cleaning of the melt were slightly changed compared to their use in the laboratory tests. In order to study the effect of different crucible materials on the degree of decontamination, both a crucible made of corundum and one made of steel were employed.

Table III shows the waste inputs, the employed crucible materials and additives, and the activities of the input and output materials.

A corundum crucible was used for melting the first batches. The cleaning efficiency obtained with the purge gases employed was comparable to that of the laboratory tests; namely, an average of approximately 90%.

In order to test the effects of a neutral crucible material and to avoid secondary waste, the rest of the tests were carried out using a steel crucible. Depending on the purge gas employed, cleaning efficiencies of between 90 and 95% were attained.

A total of 19,390 kg (net) were melted. For 15,205 kg of this amount, a gamma activity of less than 1 Bq/g for Co-60 was measured. Other gamma emitters were not detectable.

Table IV presents a summary of all wastes produced by the tests of industrial-scale melting of aluminium. The melting tests of groups 1 to 4, performed with a corundum crucible, produced 1608 kg of removed crucible material having an activity of 3312.48 kBq. This corresponds to 17.07% of the total throughput of groups 1 to 4 (9418 kg). The quantity of secondary waste amounted to 17.3% of the total amount of waste produced. When melting tests are carried out in the future using a steel crucible, this proportion will be reduced to 8.9%.

Related laboratory experiments were performed by Siemens using a graphite crucible. A furnace with a steel liner was used for the industrial-scale tests. To compare the behaviour of these two materials, Siemens also carried out melting tests with a steel crucible. As can be seen from Table II, the main difference was that more slag was formed in the steel crucible. In both the graphite crucible and the steel crucible, most of the activity (80 to 90%) became concentrated in a small fraction of slag (2 to 3% by weight of scrap).

3. Final evaluation (B.7.)

In earlier progress reports, detailed information was provided with regard to activity distribution, working time and personnel doses in both laboratory and industrial-scale melting tests. At the present time, it is not possible to make any statements regarding the cost-benefit relationship of decontamination by melting due to a lack of sufficient experience.

The experimental work has now been completed and we are presently preparing the final report.

Table I: Results of melting experiments with uranium contamination

Experiment No.		S8	S9	S10
Weight in mass Al	g	1000	1000	900
Contamination		U_3O_8	U_{met}	Al from S8; S9
Activity	α Bq ß Bq	3000 n.m.*	1500 n.m.*	170 0.08
Calculated contamination	α Bq/g	3.0	1.5	0.18
Regulus activity	α Bq/g ß Bq/g	0.22 0.10	0.16 0.06	0.11 0.05
Initial Activity of Commercially Used Al in Bq/g: 0.1				

* not measured

Table II: Results of comparison of two crucible materials - graphite and steel

Activity and Mass Distribution					
		Graphite Crucible		Steel Crucible	
		Scrap (%)	Co-60 (%)	Scrap (%)	Co-60 (%)
Weight in:		100	100	100	100
Weight out:	Metal	approx. 100	4	82	1.6
	Soluble salt	-	0.03	-	<0.1
	Insoluble residue:				
	Fraction 1	1.9	92	2.3	88
	Fraction 2	-	-	18	17

Groups	Inputs									Outputs			
	Day in 1992	SG Batch No.	Type of Crucible	Additives Type	Additives kg	No. of Drums	Drum Weight gross kg	Drum Weight net kg	Drum Activity kBq	Qty. kg	Activity kBq	Specific Activity Bq/g	Gamma Nuclides Measured by SG with Pure Germanium Detector
Gruppe 1	22 Sep	1	Lücorma 10E	AL-SM	5	3	745	688	141,00	780	161,00	0,21	Co-60
Gruppe 1	23 Sep	1	Lücorma 10E	AL-SM	5	5	709	614	1251,00	700	78,20	0,11	Co-60
Gruppe 2	24 Sep	1	Lücorma 10E	AL-T42	1	5	714	619	267,00	810	97,70	0,12	Co-60
Gruppe 3	25 Sep	1	Lücorma 10E	AL-C19	1	8	808	656	10375,00	670	303,30	0,45	Co-60
Gruppe 3	28 Sep	1	Lücorma 10E	AL-C19	2	3	640	583	45,00	735	160,00	0,22	Co-60
Gruppe 3	29 Sep	1	Lücorma 10E	AL-C19	1	5	762	668	3375,00	650	335,20	0,52	Co-60
Gruppe 3	30 Sep	1	Lücorma 10E	AL-C19	1	5	803	708	4840,00	855	476,00	0,56	Co-60
Gruppe 3	01 Oct	1	Lücorma 10E	AL-C19	1	5	867	772	2635,00	850	172,40	0,20	Co-60
Gruppe 2	02 Oct	1	Lücorma 10E	AL-T42	0,8	4	855	779	1880,00	830	187,40	0,23	Co-60
Gruppe 3	06 Oct	1	Lücorma 10E	AL-C19	2	5	988	893	2043,00	875	244,00	0,28	Co-60
Gruppe 3	07 Oct	1	Lücorma 10E	AL-C19	2	4	1020	944	60,00	900	298,00	0,33	Co-60
Gruppe 4	08 Oct	1	Lücorma 10E	SM+C19	1+2	4	667	591	785,00	870	160,80	0,19	Co-60
Gruppe 3	09 Oct	1	Lücorma 10E	AL-C19	2	6	1017	903	11130,00	850	1248,50	1,47	Co-60
Gruppe 5	13 Oct	1	Stahltiegel	Al-C19	2	5	729	634	2635,00	835	450,90	0,54	Co-60
Gruppe 5	14 Oct	1	Stahltiegel	AL-C19	2	9	1085	914	17257,00	855	3254,00	3,81	Co-60
Gruppe 5	15 Oct	1	Stahltiegel	AL-C19	2	9	1008	837	6172,00	835	220,30	0,26	Co-60
Gruppe 5	23 Oct	1	Stahltiegel	AL-C19	2	5	840	745	800,00	745	199,40	0,27	Co-60
Gruppe 5	26 Oct	1	Stahltiegel	AL-C19	2	8	1010	858	1732,00	870	245,20	0,28	Co-60
Gruppe 5	27 Oct	1	Stahltiegel	AL-C19	2	9	1005	834	18500,00	790	920,20	1,17	Co-60
Gruppe 5	27 Oct	2	Stahltiegel	AL-C19	2	7	905	772	4825,00	780	410,10	0,53	Co-60
Gruppe 6	29 Oct	1	Stahltiegel	SM+C19	1+2	10	1068	878	25304,67	785	628,00	0,80	Co-60
Gruppe 6	29 Oct	2	Stahltiegel	SM+C19	1+2	11	960	751	27242,24	840	1195,20	1,42	Co-60
Gruppe 6	29 Oct	3	Stahltiegel	SM+C19	1+2	12	1073	845	13969,83	850	1387,10	1,63	Co-60
Gruppe 6	30 Oct	1	Stahltiegel	SM+C19	1+2	16	1186	882	8226,19	830	395,80	0,48	Co-60
	Totals					163	21464	18368	165490,93	19390	13228,70		

Table III: Comparison of inputs and outputs of Siemens aluminium melting tests

Filter Dust								Slag			Other Waste			
Day in 1992	SG Batch No.	Cyclone Filter Dust Drum 1		Cyclone Filter Dust Drum 5		Fabric Filter Dust Drum 6		Slag			Spatter		Removed Crucible Material	
		kg net	kBq	kg net	kBq	kg net	kBq	kg net	kBq	Bq/g	kg net	kBq	kg net	kBq
22 Sep	1													
23 Sep	1													
24 Sep	1													
25 Sep	1													
28 Sep	1							141.00	7031.67	49.87				
29 Sep	1							150.00	2776.50	18.51				
30 Sep	1							154.00	4025.56	26.14				
01 Oct	1													
02 Oct	1													
06 Oct	1													
07 Oct	1							144.00	4625.28	32.12				
08 Oct	1													
09 Oct	1													
13 Oct	1												1608.00	3312.48
14 Oct	1							137.00	8485.78	61.94				
15 Oct	1										134.00	39886.44		
23 Oct	1										136.00	926.16		
26 Oct	1							157.00	6724.31	42.83				
27 Oct														
27 Oct	1 + 2													
29 Oct	1							187.00	4353.17	23.28				
29 Oct														
29 Oct	2 + 3							156.00	14893.32	95.47				
30 Oct	1							122.00	782.39	6.41				
		22.00	632.28	66.00	644.16	23.00 16.00	1978.00 105.76							
Totals		22.00	632.28	66.00	644.16	39.00	2083.76	1348.00	53697.98		270.00	40812.60	1608.00	3312.48

Table IV: Summary of waste from aluminium melting tests

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, eg decontamination and clearance measurements of large surface areas of premises.

D. Programme implementation

Six research contracts relating to Area No. 5 were concluded, of which one was completed at the end of 1993.

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 - March 1994
Project Manager: P MATALONI
Phone: 39/835/97 43 94 Fax: 39/835/97 42 92

A. OBJECTIVE AND SCOPE

Most reprocessing plants, at the end of their lifetime, consist of small shielded cells, inside which the process equipment is installed. The plant philosophy required the operator to enter the cells for any maintenance interventions; the cells are usually accessible from a top corridor through openings closed by shielded plugs.

The present research projects aims at testing a robotic system that can dismantle the equipment of a small cell of this type and remove cut parts from the cell without any direct intervention of the operator. The envisaged robotic system consists of a servomanipulator (MASCOT IV) and a hoist installed inside a containment box; the box has the purpose of avoiding the dispersal of contamination both during the cutting operations and during the transfer of the cut parts to the conditioning cell.

The robotic system will be tested using a mock-up of the dissolution cell of the EUREX plant, built according to the criteria of small shielded cells.

B. WORK PROGRAMME

B.1. Design and construction of a mock-up of the dissolution cell of the EUREX plant
B.2. Design and construction of a containment box and installation of the MASCOT IV servomanipulator
B.3. Non-radioactive testing of the robotic system with dismantling operations, using the cell mock-up.
B.4. Non-radioactive testing of the robotic system with simulated cell decontamination operations, including simulated smear tests.
B.5. Specific data on costs of the system and its radiological impact on work force and working area.

C. Progress of work and results obtained

Summary of main issues

The robotic system consists of a MASCOT IV servomanipulator installed inside a containment box; it has been installed inside room 101 of the ITREC Plant; also the installation of the cell mock-up inside room G43 has been completed.

The dismantling tests were performed after the functional testing of the robotic system; their completion was delayed to March 1994.

Progress and results

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

The mock-up of the Dissolution Cell of the EUREX plant was installed inside Room G43 of the ITREC Plant; the mock-up, its support and the intermediate structure necessary to re-establish the height of 5 m are shown in Figure 1.

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, a two-arm bilateral servomanipulator, installed inside a containment box. A mobile telescopic column that supports the MASCOT and a chain hoist are present inside the box.

The box is positioned above the cell mock-up and communicates with it by means of an opening closed by a double lid system. When the box is positioned over the mock-up, the two lids are connected and removed together. The movement of the two coupled lids is assured by a pneumatic system: four cylinders assure the vertical movement; when the lids are lifted, another cylinder assures the horizontal translation.

The installation of the robotic system inside room 101 of the ITREC Plant has been completed. The operator's post has been installed also; the operator supervises the operations by means of a closed circuit television; two cameras are installed on the MASCOT.

The robotic system is shown in Figure 2; Figure 3 shows the operator's post (the containment box and the cell mock-up have been built without the side walls to allow a continuous control by the operators).

3. Non radioactive testing of the robotic system with dismantling operations, using the cell mock-up (B.3.)

The dismantling tests began after the installation of the robotic system inside room 101; their completion was delayed to March 1994.

Two mechanical cutting tools were considered in the planning stage: a grinder and a reciprocating saw.

The grinder was used to cut the small piping. The tests gave good results, but presented some difficulties: the small diameter of the grinder wheel (115 mm) makes a single cut insufficient if the pipe diameter is not quite small; cuts carried out on different sides were not always aligned perfectly because the friction between the wheel and the pipe sometimes caused little displacements of the grinder; also the body of the grinder sometimes prevented from a careful alignment.

The presence of a two-arm servomanipulator has proved very useful during the cutting of the small piping: one gripper keeps the pipe immobile whilst the other one holds the grinder.

The closed circuit television has given quite good results: the presence of two cameras positioned in two perpendicular directions allows a quite correct positioning of the cutting tool. It was sometimes necessary to work by means of an only camera because an arm of the MASCOT obstructed the view to the other camera.

The cutting of the small piping is shown in Figure 4.

Many problems arose with the reciprocating saw: it is impossible to hold the saw by means of the MASCOT because of the considerable vibrations produced by the saw; the vices designed and built to fix the saw to the equipment to be cut can not be handled by means of the MASCOT because their weight (it is possible to fix the vice to a pipe only if the pipe is on the vertical of the hoist); moreover, tightening the vices is very difficult .

A new reciprocating saw, supplied with a chain clamping device, was bought after discarding the saw-vice system.

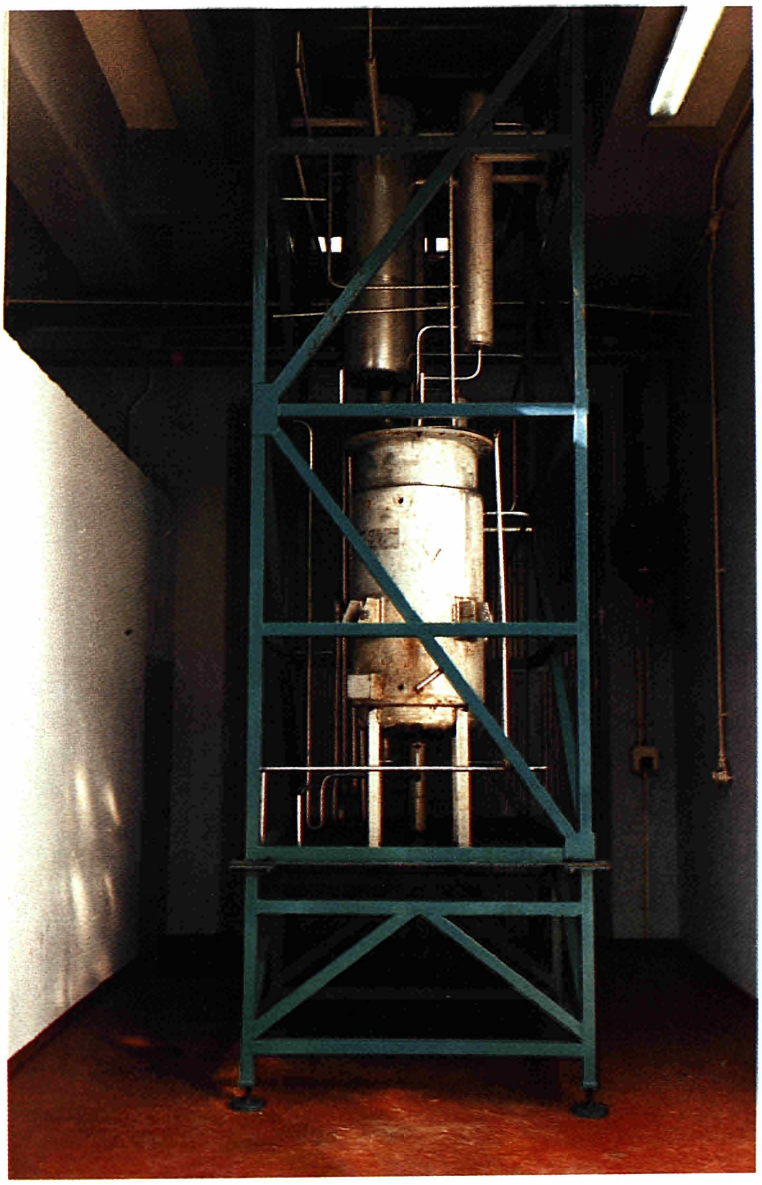


Figure 1 - The cell mock-up

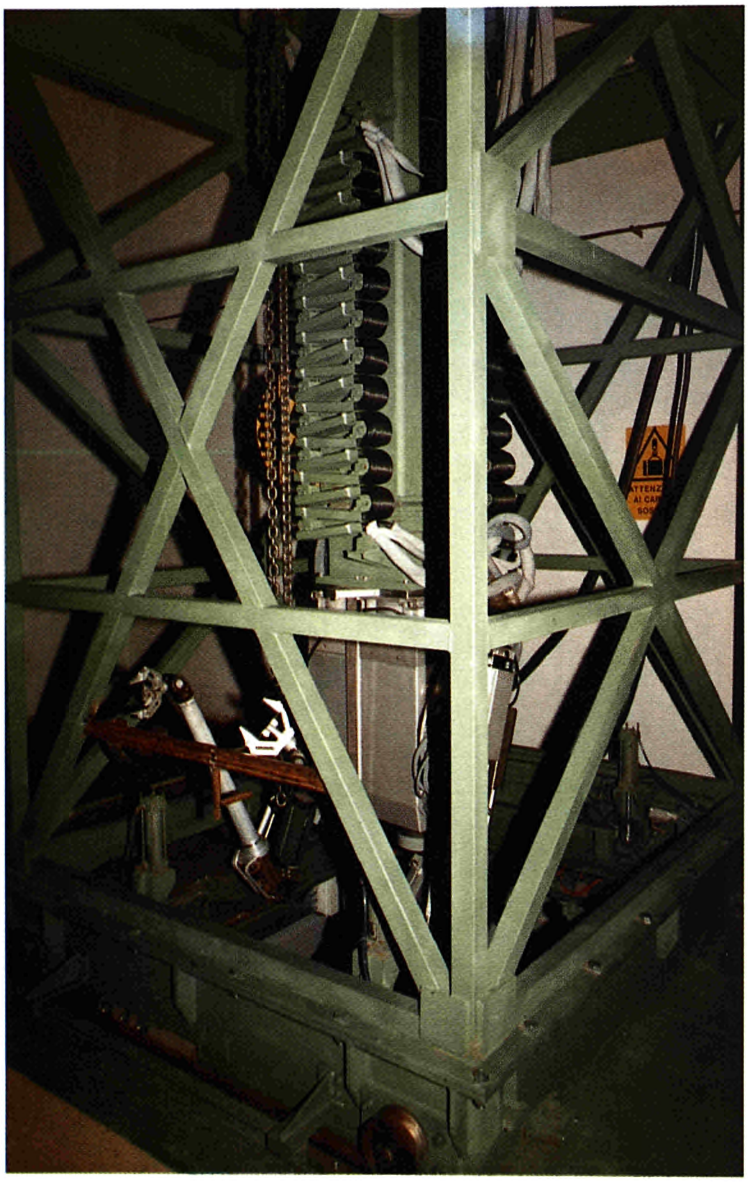


Figure 2 - The robotic system

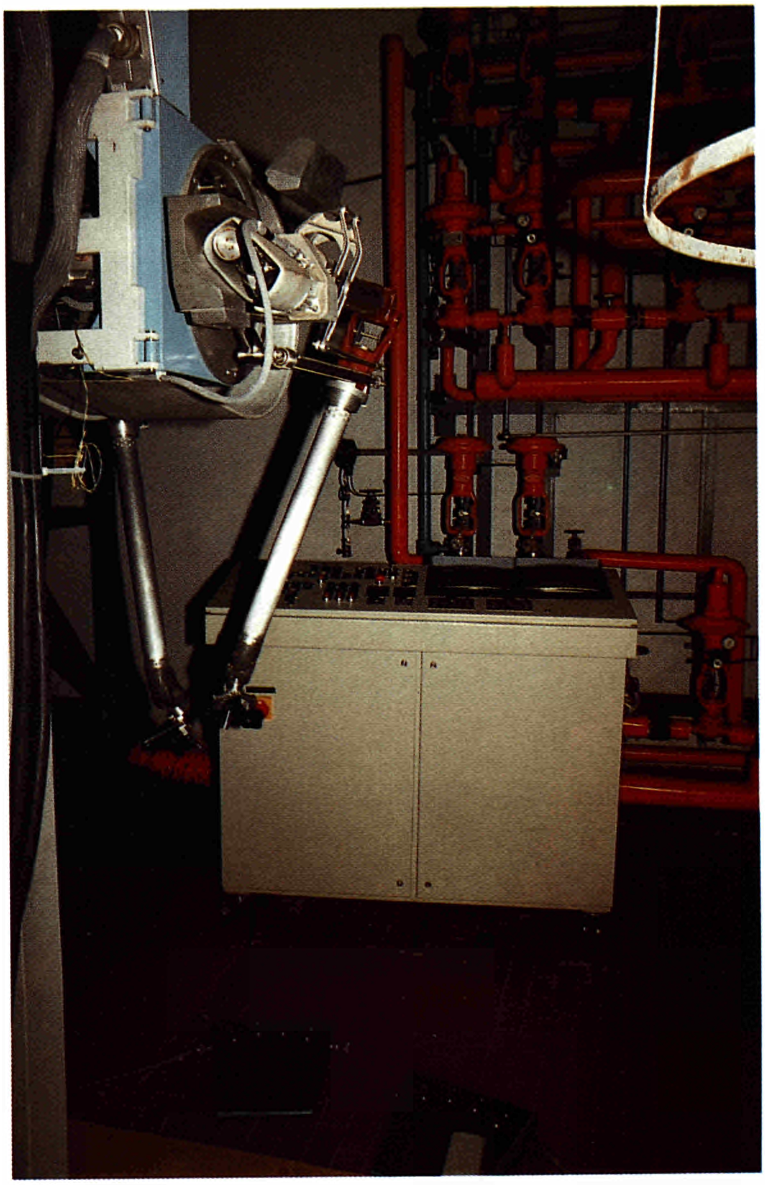


Figure 3 - The operator's post

Figure 4 - The cutting of the small piping

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 - June 1994
Coordinator: M CIARAVOLO, DNU/RTL, ANSALDO S.p.A.
Phone: 39/10/655 88 46 Fax: 39/10/655 87 99

A. OBJECTIVE AND SCOPE

The work consists essentially in the improvement of an existing melting procedure for radioactive materials and mainly relates to:

- a system specification including a preliminary study to identify the most appropriate manipulator system,
- the components design and manufacturing,
- modifications of the existing melting plant for components housing,
- installation of the components and testing of the system.

The expected benefits relate mainly to a reduction of the radiation dose to the melting staff and a reduction of the contamination in the area surrounding the furnace - operations such as slag removing, temperature measuring and samples taking being nowadays carried out completely manually. The manipulator should also increase the efficiency of the melting technique.

The manipulator developed here has thus to:

- keep people away from the furnace while it is open, in order to avoid their radiation/contamination by the melt, in particular through inhalation of radionuclides leaving the melt;
- reduce the contamination of the surroundings of the furnace (the volatile nuclides like caesium leave the open furnace);
- remove dust during melting of zinc-plated metal.

The work is a follow-up of previous EC contracts (FI1D-0016, -0047 and -0059) under which Siempelkamp and KGB Gundremmingen developed the melting facilities TAURUS I, II and CARLA.

B. WORK PROGRAMME

B.1. **System requirements such as basic operations, environmental conditions, interfaces will be specified (Siempelkamp)**

B.2. **System definition (Ansaldo)**

B.2.1. Selection of the basic concept, performing the three operations required, and comparison with a single-purpose device.

B.2.2. Definition of main manipulator operations required, i.e. scumming, sampling, and temperature measurements of the furnace melt bath.

B.3. **Design of the defined system components (Ansaldo)**

B.4. **Manufacturing and shop testing of components (Ansaldo)**

B.5. **Modification of the existing facility (Siempelkamp)**

B.6. **System installation and testing in the Siempelkamp melt shop CARLA (All)**

B.6.1. Cold tests, e.g. tool changing, manipulator working autonomy, at ambient temperature.

B.6.2. Tests at operational thermal conditions.

B.6.3. Tests with radioactive material < 74 Bq/g, i.e. carbon steel, stainless steel, steel plates covered with zinc, brass, copper and aluminium.

B.7. **Final evaluation with regard to costs, melt time, safety, occupational radiation exposure and radioactive emissions to the environment (All)**

C. PROGRESS OF THE WORK AND RESULTS OBTAINED

Summary of main issues

| After the completion of the manufacturing activities, the integration between the electro-mechanical part and the control system was performed allowing the starting of workshop tests to check the functionality of the system for all the various sequences foreseen.

Some minor problems were discovered and the needed corrections made.

On-site modifications to the existing plant were completed to allow the installation of the manipulator, which, after the transportation to Siempelkamp foundry, was initially mounted on a temporary installation where all the functional tests were repeated.

Eventually the manipulator was mounted in CARLA plant and a number of hot test with non active material were performed, showing some problems related in particular to the measure of the bath level and the capability of performing slag removal when, after a certain period, this latter solidifies near the crucible border.

Some modifications are being assessed for the solution of the above problems needing a time extension of the contract for the performance of the additional work.

Progress and Results

1. Manufacturing and shop testing of components (B.4)

The first part of the year has been devoted to the workshop assembly of the manipulator; all the functions and movements have been tested without the control system to verify the smooth running of all the motors and gears as well on board correct cabling.

In parallel the control system was tested by simulating the various operating conditions and sequences, checking also video pages display and the simplified graphic representation on the PLC monitor which allow the operator to control and perform on line monitoring of the operations.

This phase was followed by the integration of the electro-mechanical part and the control system; this was done in the workshop of the supplier of the mechanical part and lasted about two months allowing the acquisition of a useful knowledge of the system, the fixing of some minor misfunctions and eventually the starting of the shop testing of the system.

The manipulator was tested on a specially built frame (see Picture 1) simulating the locations of the crucible, the slag drum, the tools support and the coating tank as in the real environment of CARLA plant.

The initial testing concerned the manual mode operation, where all movements are directly controlled by the operator; proper operation of the safety interlocks was also checked during this phase. All the various operations and sequences related to automatic mode operation were then tested and, in particular:

- Sommer WW180 automatic changing tool device connection with the gripper or with the other tools foreseen
- simulation of movements for slag removal and deposition into the slag drum
- simulation of movements for taking samples from the melting bath
- simulation of movements for bath temperature measurement
- simulation of movements for gripper coating by protective varnish
- gripper hammering for cleaning.

The performance of tests, together with the presence of Siempelkamp personnel, allowed the identification of some aspects to be optimised, such as the need of some modifications to the tools and related support, the opportunity of improving the hammering action for gripper cleaning and the necessity of replacing the wires from the Sommer flange to the on board junction box with a new Rhodium/Platinum thermocouple extension cable.

Some design activities were also necessary in order to implement the above modifications and to bring up to date the as-built documentation.

2. Modification of the existing facility (B.5)

All the modifications necessary for manipulator installation were completed; after the enlargement of the existing foundry platform all the various anchor plates necessary for jib crane and other equipment mounting were installed and welded to the supporting structure.

Control room modifications for the installation of the power rack and the control panel were also completed.

3. System installation and testing in the Siempelkamp CARLA melt shop (B.6)

The manipulator was shipped to the Siempelkamp foundry in Krefeld in June; it was then reassembled and placed on the specially built support in a building near the CARLA plant.

This temporary installation was due to the need of performing all the other integration works, such as the implementation of the bath level measuring device (radar type), which was purchased and installed on top of the manipulator (see Picture 2), and to furnish the necessary training to Siempelkamp personnel without having an impact on the production of CARLA plant

All the functional tests, except positioning over the coating tank, which could not be tested because of space problems in the testing shop, were repeated to gain other useful information on the system.

At the same time a particular procedure for final installation was studied; this had to take into account the fact that the manipulator should be transported and installed in one piece due to the limited maximum elevation of the overhead crane hook.

The manipulator system was installed in CARLA plant (see Picture 3) at the end of September, and after the repetition of functional tests in cold conditions, a campaign of hot tests was performed in the first half of November.

Five slag removal operations were performed (see Pictures 4, 5 and 6) as well as a certain number of temperature measurements.

The tests have shown the existence of the following problems:

- During the insertion of the various tools into the bath molten metal splashes are produced; slag and metal drops hit the internal surface of the telescopic tube remaining attached and in some cases preventing the proper release of the tool on its stand; metal protections have been added to the tools solving the problem.
- The bath level measurement system does not work properly in real conditions due to the unevenness of the melting bath surface (see Picture 7); the accuracy of the measure, which is about +/- 1 cm when measuring a flat metal surface, becomes totally unreliable (+/- 30 cm) for manipulator control.
- Slag removal is effective only within a short period of time when the melting bath is ready and the slag is in a pasty state; after about ten minutes the slag tends to solidify, especially at the crucible border forming a hard annular ring (see Picture 8). This occurrence, which can be due to the strong air flow sucked by the ventilation system, makes very difficult, if not impossible, the completion of the slag removal cycle because the gripper cannot break the slag surface..

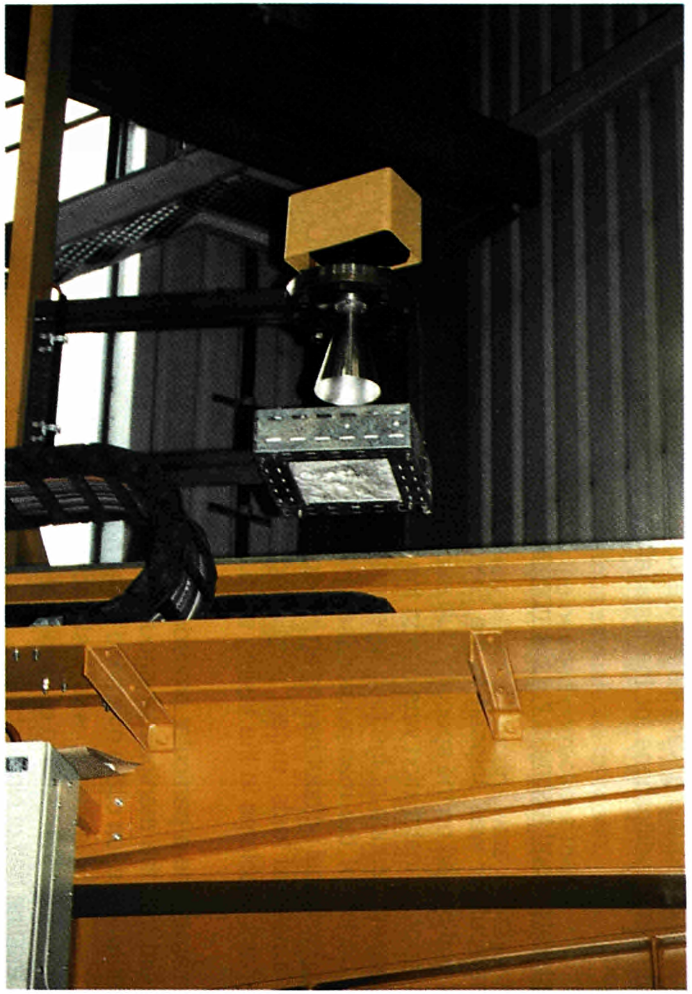
At the end of the tests, due to the need of restarting plant production the manipulator has been removed from the foundry to allow the definition of possible solutions to the problems encountered, and the implementation of the necessary modifications without problems of contamination of the mechanical parts.

The changes envisaged and which are currently analysed are:

- increase of the speed of the various manipulator movements to reduce operative times
- diminution of the ventilation air flow to reduce excessive cooling of the bath surface
- modification of the telescopic mast to allow the application of high loads (1 to 2 tons) on the slag ring (this is not compatible with the system as it is presently designed) without causing damage to the crucible lining.

To allow the performance of the above additional work a time extension of the contract of six months has been asked to the E.C.

Picture 1. Workshop testing of the manipulator

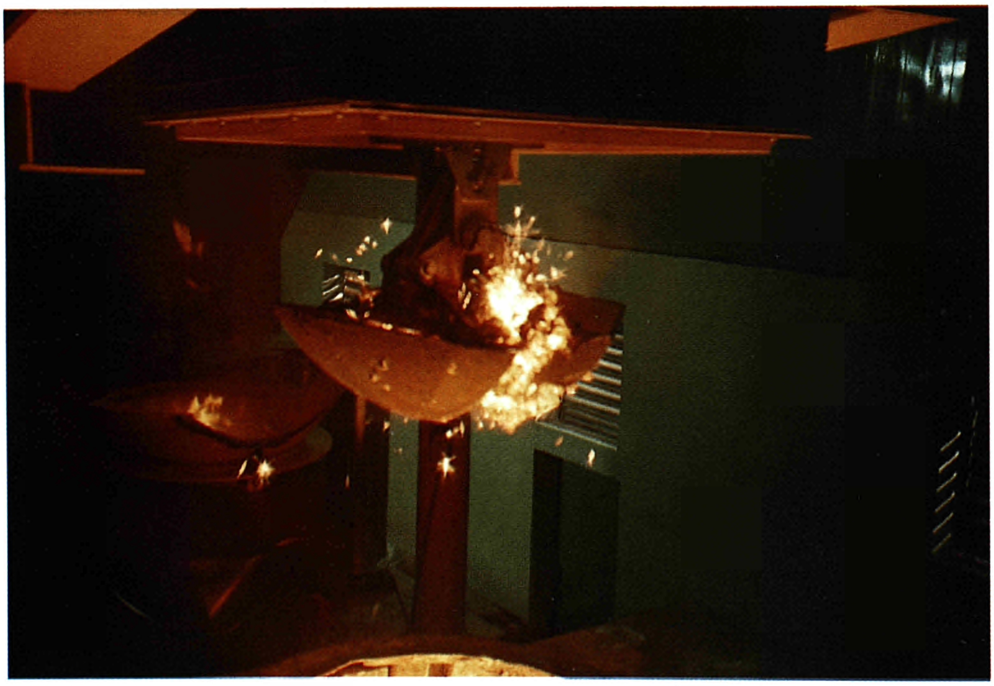


Picture 2. Bath level measuring device

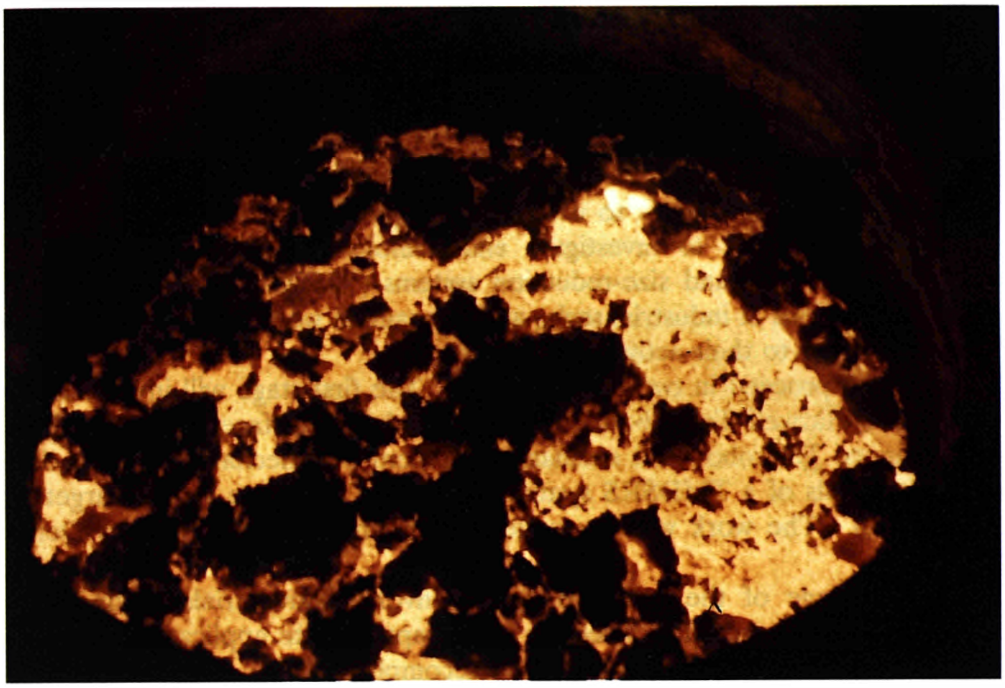
Picture 3. The manipulator installed in CARLA plant

Picture 4. Slag removal sequence (1)

Picture 5. Slag removal sequence (2)



Picture 6. Slag removal sequence (3)



Picture 7. Melting bath surface

Picture 8. Slag ring formed on the crucible edge

5.3. TELEROBOTIC MONITORING, DECONTAMINATION AND SIZE REDUCTION SYSTEM - TMDSRS

Contractors: AEA Harw., SCK/CEN
Contract No.: FI2D-0012
Work Period: July 1990 - April 1994
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

An Electropolishing Head Unit [EHU] plus alternative sub-assemblies and material samples were irradiated and a post irradiation examination carried out. An EHU was also used to demonstrate the decontamination of a flat surface in both teleoperator and robotic modes. Software was developed to demonstrate the feasibility of using a lightweight pole to extend the reach [1.5m] of a Puma 761 robot for clearance monitoring applications. By using kinematic redundancy to avoid joint limits, a radiation monitoring head was scanned over a flat vertical surface of 5m x 5.5m. Active trials of the telerobotic size-reduction system are underway and continuing to meet cost saving targets. Development work was carried out to rectify deficiencies, identified in initial trials, in cutting tools and their deployment. Most of the work packages have been completed and all will be finished by April 1994, the new completion date for F12D-0012.

Progress and results

1. Background

The TMDSRS is based on three main building blocks: a Nuclear Engineered Telerobot [NEATER], a Telerobotic Controller [TRC] and a six axis Bilateral Stewart Platform Joystick [BSPJ] to facilitate man-in-the-loop operation. NEATER is based on Stäubli Unimation's clean room robot, the Puma 762 CR. It is modular for ease of maintenance, fully sealed for prevention of contamination and radiation tolerant to 1MGy. A 66MHz 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 now linked to the VAL II robot controller via a high speed Ethernet link running SLAVE protocols.

2. Electropolishing Head Unit [B2]

A prototype EHU together with sample materials of polymer seals, piping and attachment components were irradiated in 1992. Post irradiation tests, started in 1992, have now been completed. In these tests the operability of sensors, mechanical integrity of the decontamination tool, overall operability of the tool, mechanical efficiency of the seals and chemical resistance to the acid solution were assessed. A report [1], summarising the results, has been issued. It concludes that the radiation tolerance of inductive proximity sensors is sufficiently high to meet the requirements of all three applications. Degradation of Silicone and Vitone 0-rings starts at 10kGy and 100kGy respectively. The weak point in the present EHU design is the attachment of the front seals. These are glued in grooves of square cross-section and the choice of glue is critical. During testing, the seals started to leak at a total integrated dose of only 3kGy. Alternative glues and methods of attachment were studied and solutions recommended. The chemical resistance of the materials irradiated was shown to be independent of the total integrated dose of gamma irradiation.

3. Surface Decontamination [B3]

The aim of this work package is to provide two inactive demonstrations of robotic decontamination using the [EHU] developed in B1. One will operate in robotic mode to demonstrate the decontamination of medium to large flat metal surfaces while the other will operate in teleoperator mode to demonstrate the feasibility of decontaminating small areas in cluttered and unstructured environments.

For both modes of operation the EHU [Figure 1] was attached to a force/torque sensor mounted on the robot wrist. In teleoperator mode end-point forces and torques applied to the EHU are reflected back, by the sensor, to the six-axis input device, a BSPJ. This allows the operator to use the BSPJ to control robot end-point contact forces as well as its position and orientation in the workspace. In the decontamination trial, the BSPJ was used to place the EHU in contact with a vertical perspex surface [for ease of viewing] and ensure that a firm but even pressure was applied to the EHU seal. Once applied, the BSPJ automatically maintained the force on the seal set by the operator. As a further precaution the EHU control unit applied a vacuum through four suction cups mounted on each corner of the EHU operating face. Water was then passed through the EHU and the flow viewed from the other side of the perspex plate. On completion of the electropolishing phase the water was sucked out of the head unit and the vacuum released. Finally, the EHU was moved away from the surface under the control of the BSPJ.

For robotic electropolishing, the contaminated surface is partitioned into an array of cells the size of which is determined by the EHU polishing area. The decontamination process comprises a sequence of electropolishing actions starting at the top left hand corner of the contaminated surface and proceeding a row at a time over the surface. To ensure that an even pressure is applied to the operating surface of the EHU seal, active stiffness techniques are adopted. Force/Torque measurements used by the control algorithm are obtained from a sensor mounted between the robot wrist and the EHU.

Software, developed in the previous year to scan the EHU over a flat surface at a fixed offset distance, was upgraded to allow the EHU to approach and depart from the surface at intervals dictated by the width of the electropolishing area. The upgraded software was validated by a hardware simulation of the robotic electropolishing process. It was carried out under position control only; active stiffness was not used. In parallel to this work, an active stiffness algorithm was developed to control the approach of the EHU to the surface. Early work on the algorithm showed that the achievable level of stiffness was very sensitive to control loop gain. High gains are required if the robot is to become sufficiently compliant to cope with large position errors. A low gain, on the other hand, makes the robot behave like a very stiff spring when it touches a surface, that is, it only moves a small distance even when subjected to fairly large forces. The normal VAL port through which the telerobotic controller can control the joints of the nuclear engineered robot, NEATER has a low bandwidth and consequently is only able to support low gain control functions. If higher gains are used the system becomes unstable. To overcome these problems the higher bandwidth ESLAVE port, developed under another programme, was installed in the TRC. This increased the available bandwidth by 10 and consequently the loop gain by 3. With the higher gain, the stiffness algorithm was able to use a demanded position of 15mm below the surface rather than the few mm achievable before the adoption of ESLAVE. Thus the system can now tolerate an alignment error of up to 15mm. On contact with the surface the robot becomes compliant and applies an alignment force [F] of

$$F = k\,x$$

where k is preset stiffness [N/m]

x is offset distance [m]

Spring stiffness k was set to give an applied force of 20N at an offset distance of 15mm. This force was sufficient to compress the seal by about 30% of its no load thickness.

The active stiffness algorithm was tested on the nuclear engineered robot, NEATER under teleoperator control. Execution of the algorithm was demonstrated by the operator demanding the robot to approach a surface so that the EHU mounted on its wrist, was at an angle of about 45^{o} to the surface. The edge of the EHU nearest the surface touched first and the rest of the plate automatically aligned with the surface - irrespective of operator demands - as the robot completed its approach.

Following these tests the stiffness algorithm was incorporated into the scanning software to control surface approach and departure. The combined system was successfully tested and work is now underway to optimise the target offset distance and the stiffness coefficient k to maximise system robustness.

4. Clearance Monitoring [B4]

The aim is to extend the reach of the standard robot to facilitate the scanning of a radiation monitoring head over large concrete surfaces in order to confirm or otherwise the removal of active concrete. To increase the normal robot scanning area, a monitoring head will be deployed at the end of a lightweight pole attached to the robot wrist. The 1.3m long pole was made from composite materials to obtain the maximum length within the weight constraint imposed by the robot servo stability requirements. To maintain the head unit parallel to and in contact with the surface to be monitored, the beta gamma dose rate meter was housed in a gimbal arrangement. This provided two degrees of freedom; pitch and yaw. Roll was not required because the dose rate meter was not sensitive to rotations about an axis normal to the surface being monitored.

A scanning strategy was devised and coded in the previous year. Trials to validate the strategy revealed that, although the technique of scanning the upper half of the operating area in the wrist 'up' configuration and the lower half in the wrist 'down' configuration worked, the size of the scanning area was constrained by robot joint limits. To overcome this problem an extension of the above strategy was proposed, implemented and successfully tested. The new strategy uses the system's two redundant degrees-of-freedom to overcome joint limit problems. [Redundancy arises because the robot has five degrees-of-freedom - the sixth degree-of-freedom [wrist roll] is not required - to execute the three degrees-of-freedom scanning task]. In the new strategy, the following procedure is used to avoid joint limits during each line scan.

When one of the robot joints approaches a limit the other joints are adjusted heuristically to move the offending joint away from its limit while maintaining the position of the head unit on the surface. On completion of this task, the line scan continues until the system detects a joint approaching its limit. The procedure continues until the complete surface has been scanned. Using the above technique a flat vertical [5m x 5.5m] surface has been scanned and the program stored in memory on the VAL controller. This program can now be used to monitor other surfaces of a similar size providing the robot is positioned at the same distance from the surface.

5. Glovebox Size-Reduction [B5]

Robotic force control, for the alignment of cutting tools and feedrate optimisation, has been identified as a cost effective method of improving cutting efficiency. As these force control algorithms need to know the forces applied to the cutting tools, a study [2] [3] of the radiation tolerance of strain gauges used by six-axis force/torque sensors has begun. Specific gauges with polyimide backing or free filament sensors were selected and characterised. Special attention was given to the bonding agent. Ceramic bonds with thermal curing were preferred. These gauges have been prepared for gamma irradiation test in the representative conditions of dismantling. For some of the components, on-line loading was applied during the test. In collaboration with manufacturers, special versions of one-axis load cells and six-axis force sensors were designed and fabricated. They have been characterised using adapted calibration procedures and the irradiation test was started at the end of 1993. The main part of the irradiation test, as well as the post irradiation examination and report generation, is scheduled for completion in 1994.

6. Active Trials [B6]

Active trials of the telerobotic glovebox size-reduction system started in October 1992 and, to date, have been relatively trouble free. However, a number of deficiencies in the decommissioning tools have been identified. To rectify this, development work has been carried out to improve the telerobotic efficiency of these tools.

The existing drill has low speed, high torque characteristics to facilitate the cutting of 50mm holes using a hole saw. Its main drawback is that at 18kg it is too heavy. In some robot arm configurations it exceeds the robot payload and trips the robot controller. Stall prevention software, that was designed to eliminate robot tripping, is unable to function with these high payloads. This is because the software limits applied cutting forces by reducing feedrate to ensure that joint motor torques remain within the continuous duty range. The system, however, is unable to compensate for gravitational forces that cause joint motor torques to exceed continuous duty levels. The heavy weight of the tool also increases the difficulty of teaching its tool rack position. The new drill weighs 8kg and as such it overcomes the problems encountered with the existing drill. Its design, while based on its predecessor, is more compact and is of a similar size and weight to that of the reciprocating saw. Functional tests have established that the drill's performance is equivalent to that of the existing drill.

To change the reciprocating saw blade, a spanner was required to remove the two blade retaining nuts on the reciprocating saw. The task was further complicated by the need to stop the saw blade at the correct point in its stroke. On a number of occasions, the retaining nuts were lost during a blade change. To obviate the need for the frequent posting of retaining nuts, spare nuts were stored in the active area near the blade change station. Overall, the blade change task was difficult to execute using gloves and consequently took between 10 and 15 minutes to do. A new blade change system, based on a sprung loaded single screw jack mechanism, has been designed and installed. It has significantly reduced task complexity; blade changing now takes about one minute. No extra tools are required and there are no parts to be removed. The screw jack can easily be rotated by a gloved hand or by a master-slave manipulator and operation is not sensitive to blade position in the reciprocating cycle.

To reduce tool alignment times, a six-axis unilateral input device was installed in the size-reduction facility. Preliminary results of a performance evaluation on the input device indicate a 70% reduction in tool alignment times.

References

[1] COENEN S, Gamma Irradiation Experiment on Electropolishing Head Unit - Test Results, Report G4002-27/93-98/SC, September 14, 1993.

[2] NINA NOPPE, MARC DECRÉTON, Straingauges in a Nuclear Environment, European Seminar on Strain Measurement at High Temperatures, Cologne, October 14, 1993.

[3] NINA NOPPE, MARC DECRÉTON, Straingauges in a Nuclear Environment, Materials and Design, Vol 14, No 6, 1993.

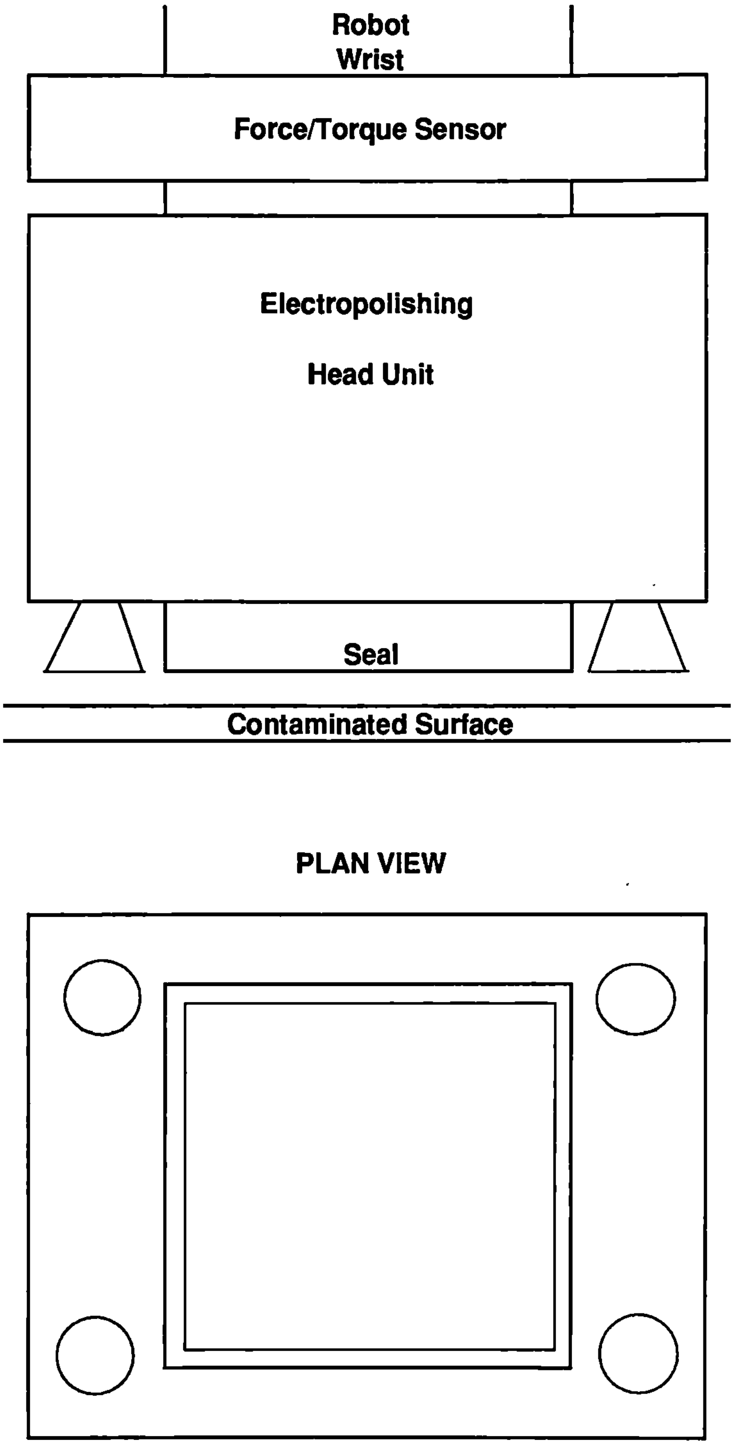


Figure 1: Electropolishing head unit deployment

5.4. ADAPTATION AND TESTING OF A REMOTELY CONTROLLED UNDERWATER VEHICLE

Contractor: AEA Wind.
Contract No.: FI2D-0025
Work Period: July 1990 - February 1995
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 a manipulator, an on-board television system and remote sensors for surveillance purposes. This includes the development of an underwater remotely controlled vehicle for use in nuclear applications, the development of techniques for fuel handling and silt recovery, and the assessment of equipment durability, radiation tolerance and of decontamination problems, e.g. removal of contaminated concrete surfaces, etc.

The potential byenefits of this work programme will be reduced doses to workers, decontamination and inspections, reduced secondary waste and reduced decommissioning costs due to lower labour input.

Information on costs, occupational exposure, work time and secondary waste arisings will be made available.

The work will include non-active trials in a full-size mock-up.

B. WORK PROGRAMME

B.1. Specification of the underwater vehicle, to allow manufacturers to tender.
B.2. Vehicle manufacture, at the tenderer's work.
B.3. Preparation of the full-size test mock-up of the entrance area to the water duct.
B.4. Adaptation and testing in the mock-up of the vehicle/system on its ability to perform the various decommissioning tasks.
B.4.1. Dry testing of the complete system.
B.4.2. Wet testing of the complete system.
B.5. Development in the active environment of Windscale Pile No. 1.
B.6. Final evaluation will include specific data on costs, work time, occupational exposure and the secondary waste arising from the technique.

C. Progress of work and obtained results

Summary of main issues

The Vehicle has been manufactured (B.2.). It was delivered to AEA Technology in April 1992. The Dry Test Facility has been constructed (B.3.). Work is complete in the Dry Test Facility on the Vehicle development programme and operator training (B.4.1.). The Wet Test facility is built and operator training (B.4.2.) has progressed. The vehicle umbilical management system has been manufactured and tested at the manufacturers works. Silt recovery plant has been built and tested at the manufacturers works.

Progress and results

1. Vehicle manufacture, at the tenderer's works.(B.2.)

The vehicle was delivered to Windscale site in April 1992.

The original proposed vehicle would have been a prototype designed specifically for this application. A decision has been made by the AEA to purchase a vehicle built up of proprietary items that will be adapted to work in the water duct.

2. Preparation of the full size test mock-up of the entrance area to the water duct.(B.3.)

It was decided in early 1991 to have two separate test facilities, one dry and one wet because much of the early development programme could be carried out in a dry facility. Design work for the Dry Test Facility was completed and the facility 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 3 metre long section of the Water Duct. It is designed to be light-tight so that a simulation of conditions in the Water Duct can be attained.

The Wet Test Facility is built and represents a 6 metre section of the Water Duct. The vehicle has undergone trials in this facility. Operator training is in progress. Silt recovery trials at the manufacturers works are complete.

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

The dry test programme is complete. Results to date are summarised below.

Training programmes have been completed for the proposed operators of the vehicle. The training was aimed at simulating the operations that will be done during fuel and silt recovery in the Water Ducts. All operators have completed the vehicle manufacturers initial training course.

The training programmes consisted of practice periods followed by regular witnessed tests. Early indications were that the operators quickly became familiar with the vehicle's operation because the time taken to complete the witnessed tests reduces rapidly. Early indications are that the operators reach a plateau at the minimum time they take to complete a witnessed test. Further training is in progress in the Wet Test Facility.

(1) Camera systems: An overview camera mounted on the rear of the vehicle monitors all vehicle operations. An additional camera has been identified and fitted to the manipulator to give a better view of the gripper during operations.

(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 complete. The worst condition is the manipulator hitting the wall giving a peak stress of 0.375 MPa. The yield strength of the concrete wall is assumed to be 2.8 MPa. Therefore the worst condition is only 13% of capacity of the wall. No damage is likely to occur as a result of a manipulator colliding with the duct walls.

(3) Manipulator jaw gripping force: The gripper force will be variable between 0 and 500N. It will have to be limited to avoid damaging the aluminium clad fuel elements during recovery. A system has been designed and fitted to control the gripper force, however technical problems have delayed the use of this control. Gripper pressure can be set manually by adjusting a valve setting on the vehicle.

(4) Vehicle detection system: An equipment survey has identified a suitable sonar device to enable the distance that the vehicle has travelled along the Water Duct to be accurately measured. This will assist in the record keeping of fuel element locations when they are recovered.

The same device is capable of identifying fuel elements buried in the silt that exists in the Water Ducts. This may enable the vehicle to continue working when silt becomes re-suspended in the water.

(5) Simulation of Water Duct silt: A simulation of the Water Duct silt has been prepared. This simulated silt has been used in silt pumping trials. The silt pumping system is an eductor pump. To date the silt pumping system has transported particulate less than 100μm dia up to 30 mm dia over a distance of 150 metres against a 7 metre head.

4. Wet testing of the complete system.(B.4.2.) The design of the Wet Test Facility is complete. Construction of the facility is complete. Work is in progress on vehicle development and operator training in the Wet Test Facility.

Since the publication of the 1992 report, the need for an umbilical cable management system has been identified. This system is to control the reeling in and out of the umbilical cable for the vehicle during operations and recover the vehicle on power failure. This system has been manufactured and tested at the manufacturers works.

5. Development in the active environment of Windscale Pile No.1.(B.5.)

Not yet started

6. Final evaluation will include specific data on costs, work time, occupational exposure and the secondary waste arising from the technique.(B.6.)

Not yet started.

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

This project was completed in December 1992. The final report is available as EUR Report No. 15425.

5.6. UNDERWATER QUALIFICATION OF RD 500 MANIPULATOR

Contractors: CEA FAR, Framatome, TNO Delft
Contract No.: FI2D-0041
Work Period: October 1990 - June 1994
Coordinator: E VILLEDIEU, CEA/DTA/UR, CEN/FAR
Phone: 33/1/46 54 75 58 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. <u>Preliminary testing of the complete system</u>

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. <u>Underwater qualification tests</u>

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. <u>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</u> (All).

C. Progress of work and results obtained

Summary of main issues

At the beginning of 1993, the state of progress of the project was as follows:

- the qualification tasks had all been established together with their test protocol,
- all application specific equipement had been developed, perfected and tested separately,
- global underwater tests were carried out in the CEA's MINERVE pool at Fontenay-aux-Roses.

Progress and results

The qualification tests were carried out in the ATEA pool (FRAMATOME subsidiary) at Nantes between the 22 March 1993 and the 30 April 1993. These tests marked the final phase of the project. They were conducted in several stages:

- The first week was dedicated to installing and setting up the system.
- The second week to the qualification of sectioning work and plasma torch cutting tests.
- The third week to the qualification of plasma torch cutting operations.
- The fourth week to the qualification of electro-erosion operations. (Fig. 1 & 2)
- The fifth week to the qualification of operating using ultra-sonic inspection equipment.
- The sixth week to back-up operations and taking snapshots.

The tests took place according to plan, and the results were entirely satisfactory.

In the case of sectioning, the four tubes were successfully sectioned. This process was problem free. Nonetheless, the close presence of the mock-up induced a force of attraction when the disc was turning, making sectioning even more difficult.

In the case of plasma torch cutting, two windows of three sections each were produced. The pieces of sheet metal from the three plates could then all be removed from the mock-up; the resulting sections were regular. The only problem encountered during the tests were difficulties with lighting the torch under water.

In the case of machining by electro-erosion, six pieces were machined; two of them on the total of the screw head and the rest on the bar. The machining gave good results and was in compliance with the specifications.

The key problem with this process was calibrating the tools. In the case of visual inspection, calibration was performed mechanically, being relatively easily verified in air.

In the case of inspection by ultra-sonic, vision system, calibration could only be performed under water, this stage was therefore more difficult and longer.

However, once the system had been calibrated, the procedure could be performed without any problem.

Two plasma sections were examined using the ultra-sonic camera:

The first only went through the first sheet and was located at one side of the mock-up. This section was examined in its entirety.

The second was a set of three successive sections through the three thicknesses of the sheet metal. A local examination was carried out. The possibility of ascertaining the depth reached at each part of the section meant that the geometry could be understood. Nonetheless, in two cases, visual examination proved insufficient for characterising the section. The first resulted from a spot weld added to secure the place and which had not been detected. The second was due to contraction of the sheets, hiding the rear of the section.

Technical conclusion

These underwater tests consisted in remote handling in a near industrial environment, to a tight schedule which was observed to the day throughout a six-week period.

With regard to the RD500 system and its TAO2 control system, the tests showed that this method had achieved a success unequalled anywhere in the world: remote underwater plasma cutting. This process does have certain limitations:
- the torch-to-surface distance must be constant, in our case 6 mm;
- the travel speed must be constant and slow;
- there is no possibility of visual examination during cutting work;
- the sectioning was performed on a piece of known profile, but following an unknown trajectory.

Only remote robotics technology could rise to this challenge through the combination of learning and automatic control of the arm.

The tests also made it possible to confirm, under real conditions (using the electro-erosion tool), that the placing of a variable load tool was repeatable to within 1 mm. This had only ever been demonstrated before in a laboratory by handling a tool mock-up in air.

The tests proved the reliability of the RD500-TAO2 system. The RD500 arm operated under water for 70 hours, for average periods of 5 hours, performing twenty arm movements to get it in and out of the pool. During underwater operation, there were no unscheduled operations to raise the arm, and all anomalies encountered were resolved from the control console.

The tests nonetheless have certain lessons to teach us for future handling operations:
- The dynamic waterproofing functioned perfectly and no leak was detected when the boxes were opened during the mechanical appraisal carried out after the tests. Nonetheless, the seals used in the experiments were changed as a precaution.
- The mechanical appraisal did not indicate any major problem. Nonetheless, it seems necessary to overhaul the braking system of one of the coupling bearings involved in elevation.
- One of the cables inside the arm suffered mechanical stress; it therefore proved necessary to review cable management in this area when moving the robot.
- The control system exhibited an anomaly during operation and several failures were detected when it was brought into service following transportation. Future plans involving this control system will have to take account of the ruggedness of the components.
- The man-machine interface proved unable to ensure safe and easy control of the system. In many cases, the presence of one operator instead of two increased operator fatigue, resulting in human errors. The installation of a graphics module in the control unit would be a partial solution to this problem.

In the case of the ultra-sonic inspection system, the tests showed the usefulness and wealth of 3D information, and the ease with which it could be used in control or fine tool-adjustement operations. Improved integration of the ultra-sound camera with the RD500 robot would make these much easier to use.

In the case of the electro-erosion tool, the tests showed that the RD500 arm provided, thanks to the use of a support, the degree of stability needed for this process.

The six tests meant that machining could be performed to within 0.7 mm, using the RD500 arm's repeatability qualities and centred position-learning by means of an optical or ultra-sound camera as an external environment sensor. These results were obtained after setting or calibrating the work axes of the two sight and machining tools.

The tests validated the task of machining partitioning screws for pressure vessel internals in 900 MWe PWRs. It was showed that even the most awkward screws could be reached.

In conclusion, these tests have shown that the RD500 arm together with its tools and vision systems was a prototype suitable for use in real operations. It also showed that the system is ready for industrial application.

Improvements to the system will involve increasing the security in manual mode and increasing system productivity, allowing the operator to rapidly "plan-out" the schedule of operations.

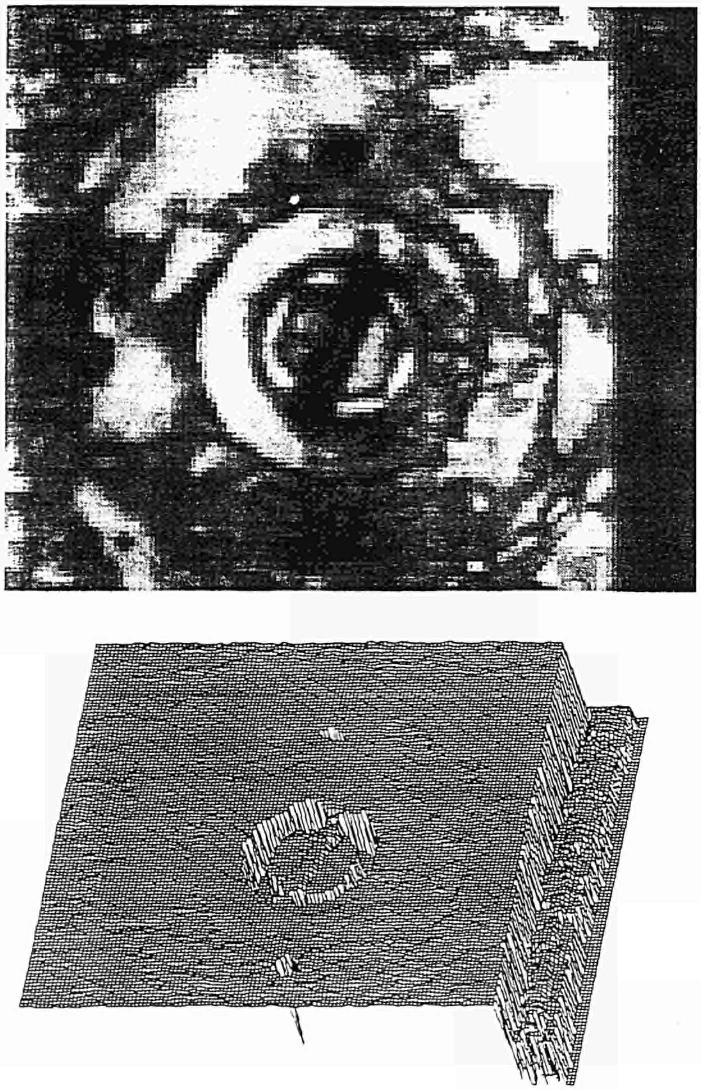


Fig. 1:

Amplitude (top) and travel time (bottom) image used for the electro erosion tests

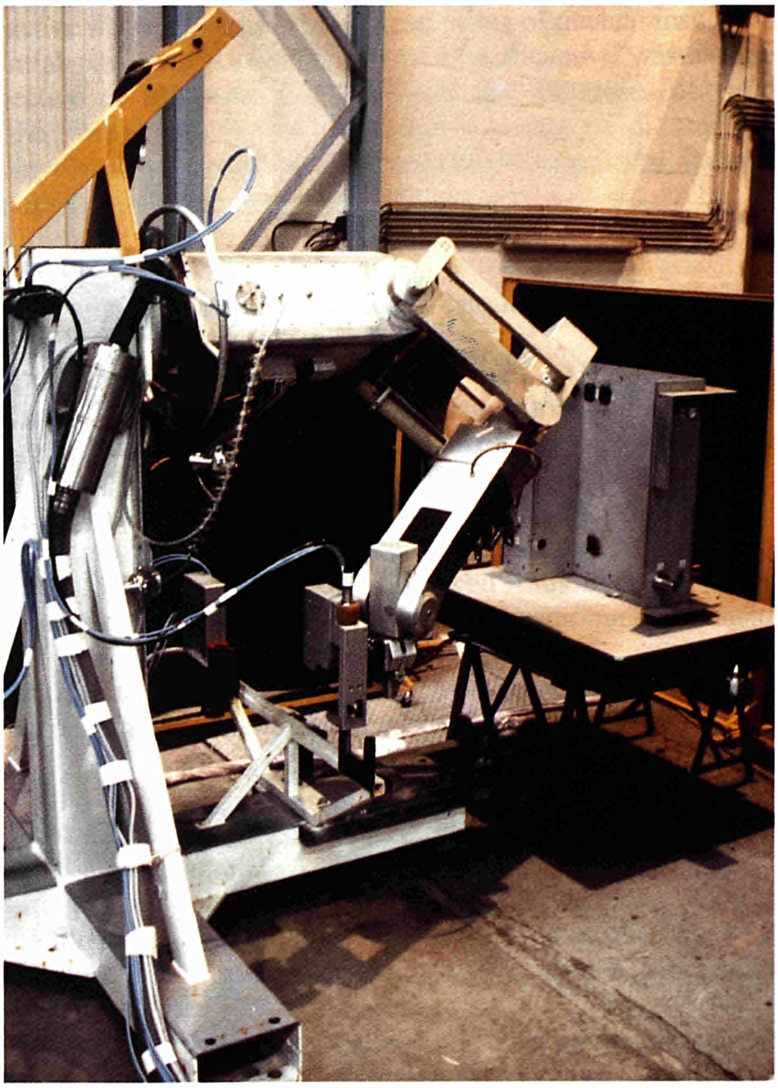


Fig. 2: **RD 500 with its tool rack and the electro-erosion device**

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 to evaluate the risk of decommissioning operations should be developed.
The evaluation of residual activity levels below which materials from decommissioning could be reused should be pursued, including consideration of statistical aspects.

D. Programme implementation

Nine research contracts relating to Area No. 6 were concluded, of which six were completed at the end of 1993.

6.1. METHODOLOGY TO EVALUATE THE RISKS OF DECOMMISSIONING OPERATIONS ON NUCLEAR PLANTS

Contractors: AEA-Culcheth, NRPB, AEA-Windscale
Contract No.: FI2D-0030
Work Period: October 1990 - March 1993
Coordinator: D G JONES, AEA Technology, Culcheth
Phone: 44/925/25 43 76 Fax: 44/925/25 44 37

A. OBJECTIVE AND SCOPE

The theoretical work is composed of two distinct but complementary studies:

1) 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 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;
2) 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. Waste management options (AEA, NRPB)
B.1.1. **Development of a radiological risk evaluation methodology**
B.1.2. **Selection of the waste stream for an example application of the methodology**
B.1.3. **Definition of the radionuclides inventory and their distribution in the waste stream**
B.1.4. **Definition of waste management options**
B.1.5. **Estimation of financial costs for each of the management options**
B.1.6. **Calculation of doses and risks for individuals and the public**
B.1.7. **Assessment of social and environmental impacts of waste management options**
B.1.8. **Demonstration of the methodology by identifying the optimal management options**
B.1.9. **Review of the results and check of their applicability to other decommissioning decisions.**
B.2. Decommissioning strategies (AEA)
B.2.1. **Definition of decommissioning phases of non-reactor nuclear plants**
B.2.2. **Identification of techniques for carrying out decommissioning operations and their risk-bearing elements**
B.2.3. **Identification of risk assessment procedures taking into account accidental risks.**
B.2.4. **Evaluation of procedures for assessing the risks associated with leaving the plant under care and maintenance**
B.2.5. **Examination of methods for the aggregation of risks associated with particular decommissioning strategies**
B.2.6. **Demonstration of the identified methodologies to a non-reactor facility**
B.2.7. **Final evaluation on the suitability and limitations of the identified methodologies.**

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1993. The final report is being prepared for publication.

6.2. DOSES DUE TO THE REUSE OF VERY SLIGHTLY RADIOACTIVE STEEL

Contractors: CEA-FAR, BS, SIEMENS BEW
Contract No.: FI2D-0031
Work Period: September 1990 - February 1994
Coordinator: Mrs H GARBAY, CEA-IPSN, Fontenay-aux-Roses
Phone: 33/1/46 54 73 41 Fax: 33/1/47 35 14 23

A. OBJECTIVE AND SCOPE

The scope of the study is the determination of doses due to the reuse or recycling of very slightly contaminated radioactive steel in case of mechanical and thermo-mechanical treatments applied to scrap when exempted from regulations.

The study will mainly be based on already available data both in the nuclear field and in the conventional scrap industry. Experimental investigations will be performed, as far as possible, on radioactive samples coming from nuclear installations being dismantled. The various treatments applied to scrap before its melting have not yet been studied and are of great interest. In particular, techniques used in scrapyards should be studied in the view of inhalation and external exposure injuries.

This study applies to a large quantity of steel arising from dismantling of nuclear installations (EUR 10052).

Benefits are expected as regards management and cost of radioactive waste arisings, protection still being secured. The results concerning contamination dispersion during cutting of scrap will be useful for the evaluation of future large-scale decommissioning operations.

B. WORK PROGRAMME

B.1. Discussion and documentation of the present regulatory situation. (BS-CEA)
B.2. Performance of steel cutting and aerosol sampling experiments observing industrial conditions. (CEA-Siemens)
B.3. Evaluation of inhalation risk in realistic situations. (all)
B.4. Determination of the radiological impact based either on bibliographic data or on experimental results. (BS-CEA)
B.5. Development of a stochastic programme to obtain the individual dose distribution. (BS)

C. PROGRESS OF WORK AND RESULTS OBTAINED

Experiments on cesium and cobalt (B2-B3)

Three types of steel have been cut by a thermal tool. Carbon steel, rusty and not rusty, have been cut with oxyacetylene torch and stainless steel has been cut with a plasma torch.

Experiments are carried out in a 32 m^3 ventilated cubicle, with an air exhaust rate of 300 $m^3.h^{-1}$. Experimental atmosphere measurement devices are :

- a sampling system equipped on an extraction duct : upstream, a **sampling filter** samples the total quantity of aerosols emitted in the extraction duct ; the sample is sent through an 8-stage impactor, followed by a 6-channel diffusion battery, called **S.D.I. 2001 particle size analyser,** the particle size distribution of the aerosols is determined between 0.0075 micrometers and 15 micrometers.

- Inside the test cubicle, an inertial device, the **SEPIA sampler** collects particles larger and smaller than 10 micrometers on two separate filters. Since the device is located near the operator, the directly inhalable fraction of the aerosols can be determined (< 10 micrometers).

The sampling devices give different information. SDI 2001 indicate the aerosol size distribution collected by the extraction duct. The filter samples the total aerosols emitted and collected in the extraction duct, probably loosing a part of heavier particles which are deposited inside the test cubicle. SEPIA sampler gives the mean value of the ambient aerosol atmosphere in the test cubicle induced by the cutting experiments.

Aerosol thermal behaviour : Relatively good correlation can be made between aerosol emission and elements melting temperatures and also between aerosol emission and boiling temperatures, the lower the temperature, the higher the emission. This correlation can be made not only for cobalt or cesium, deposited on the metal surfacc, but also for the elements constituting the metal itself (Fe, Mn, Ni, Cr) (figures 1 and 2).

Aerosol size distribution : The SDI 2001 gives aerosol size distribution in the extraction duct for carbon steel experiments and stainless steel experiments (figures 3 and 4). Both appear bimodal. Nearly half of the emission is made of particles lower than 0.36 µm, no more than 10% of the total mass are aerosols with diameters higher than 7.5 µm.

Comparison of mechanical and thermal experiments (table 1) : The comparison of mechanical and thermal experiments shows that cutting steel with a thermal tool generates more particle emission than with mechanical tool. The difference between the tools is higher for cesium emission than for cobalt emission.

On carbon steel, a difference rate of 1.8 to 3.4 is shown for cobalt when comparing mechanical tool to oxyacetylene torch ; on stainless steel, mechanical tool generates less particle emission, the difference then increases to 7 to 9.

For cesium on carbon steel, the difference rate between mechanical tool and oxyacetylene torch is reaching 10 to 34 ; on stainless steel, mechanical tool also generates less particle emission, the difference rate then reaches 70 to 78.

Experiments on uranium (B2-B3)

Summary of main issues

Within the scope of the experimental program, three experimental series were performed, in order to determine the release of activity during thermal cutting of contaminated metal scrap.

1. - cutting of austenitic material contaminated with UO_2 powder
2. - cutting of austenitic material contaminated with UN solution
3. - cutting of ferritic material contaminated with UO_2 powder

In addition, for each experimental series, a background measurement on non-contaminated metal sheet was carried out, using the same cutting method as the contaminated metal sheet, and the distribution of activity in the individual phases was also determined. For two experiments of the first experimental series in addition to this the coated filters were examined for uranium content.

The investigation have shown that the activity release determined for austenitic and ferritic material with the selected contamination type UO_2-powder and uranyl nitrate solution varied within the range of a factor of approximately 2.

The activity release code number, C_{RA1} for austenitic material contaminated with UO_2-powder, approx. 2,3 E-02 $Bq.cm^{-1}/Bq.cm^{-2}$ (figure 5), C_{RA2} for austenitic material contaminated with UN solution, approx. 3,3 E-02 $Bq.cm^{-1}/Bq.cm^{-2}$ (figure 6) were determined. For ferritic material, which was contaminated with UO_2-powder, somewhat higher releases resulted, which gave a determined release code number C_{RF} of approx. 4,4 E-02 $Bq.cm^{-1}/Bq.cm^{-2}$ (figure 7).

The particle size distribution could not be determined for all experiments due to the background effect and the obviously low releases in the range of larger particles. Evaluation of the experiments has shown clearly that activity releases caused by the separation process can only be demonstrated for particle sizes < 0,42 µm (back-up filter) over the entire measurement range (figures 8 to 10).

Comparison of cobalt, cesium and uranium release rates(B3)

Release rates for cobalt emission and for cesium emission are compared to uranium results obtained by SIEMENS in table 2.

Cutting ferritic steel gives the higher uranium emission value (0.7 to 1.8 $cm^2.m^{-3}.m^{-1}$) ; cutting austenitic steel gives lower values (0.1 to 0.35 $cm^2.m^{-3}.m^{-1}$). Cesium emission remains at the same emission level whatever the type of steel (1.2 to 1.8 $cm^2.m^{-3}.m^{-1}$), and cobalt emission from any type of steel stays in a narrow range (0.24 to 0.6 $cm^2.m^{-3}.m^{-1}$).

Development of a stochastic programme to obtain the individual dose distribution (B.5)

In the annual report from 1992 the development of a stochastic model to determine the individual dose distribution from manual processing of scrap, which is contaminated with α-emitters, was presented. The model has been modified to calculate the inhalation doses from manual processing of scrap contaminated with nuclides typical of a light water reactor (LWR). The results of the simulations are presented in table 3. The radiological impact of inhaling resuspended activity during manual processing of scrap is several orders of magnitude higher for α-emitters than for the nuclides which dominate in decommissioning of LWRs (e.g ^{60}Co and ^{137}Cs). For these nuclides the radiological most relevant exposure is external irradiation. The results in table 3 show that the inhalation dose from manual processing of scrap can be ignored when compared to the expected external exposure (for the scrap yard [1] estimates 8 µSv/a from external exposure to the scrap).

REFERENCES

1/ COMMISSION OF THE EUROPEAN COMMUNITIES (Ed.), Radiological Protection Criteria for the Recycling of Materials from the Dismantling of Nuclear Installations, Radiation Protection No. 43, Luxembourg 1988.

TABLE 1 : Aerosol measurements with sepia sampler for oxyacetylene torch and plasma torch cutting

	MECHANICAL EXPERIMENTS	THERMAL EXPERIMENTS		
	Element release rate ($\mu g.m^{-1}.m^{-3}/\mu g.cm^{-2}$)	Element release rate ($\mu g.m^{-1}.m^{-3}/\mu g.cm^{-2}$)	size distribution >10µm (%)	size distribution <10µm (%)
Carbon steel				
Cs	0.156	1.6	21	79
Co	0.167	0.57	55	45
Rusty steel				
Cs	0.052	1.8	24	76
Co	0.131	0.6	49	51
Stainless steel				
Cs	0.016	1.11	24	76
Co	0.058	0.4	40	60

TABLE 2 : Cesium, cobalt and uranium release rates

	CESIUM ($\mu g.m^{-1}.m^{-3}/\mu g.cm^{-2}$)	COBALT ($\mu g.m^{-1}.m^{-3}/\mu g.cm^{-2}$)	URANIUM ($Bq.m^{-1}.m^{-3}/Bq.cm^{-2}$)
Carbon steel	1.6	0.3 to 0.57	0.7 to 1.8
Rusty steel	1.8	0.24 to 0.6	-
Stainless steel	1.2	0.4 to 0.5	0.1 to 0.35

TABLE 3 : Individual dose distribution from inhaling resuspended activity during manual processing of scrap metal contaminated with LWR typical nuclides. Clearance level is 1 Bq/cm^2 and 1 Bq/g.

scrap quantity (Mg/a)	number of exposures		largest dose (µSv/a)
	dose category 1 - 10 µSv/a	dose category >10 µSv/a	
1000	0,2	0	4
5000	6,4	0	8
9000	15,5	0	9

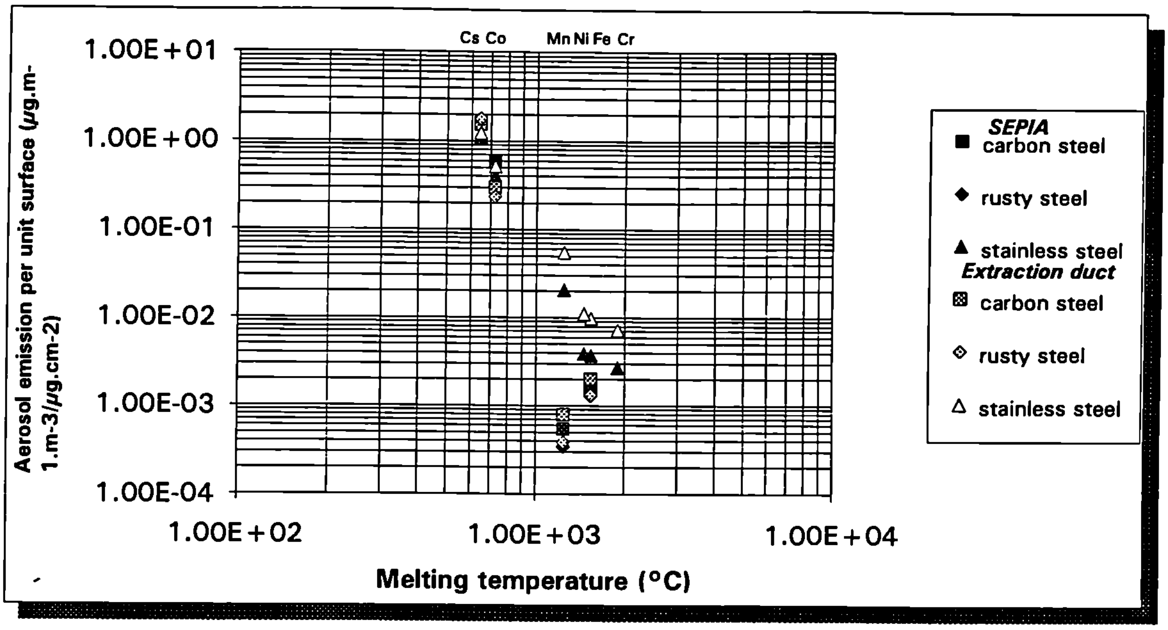


FIGURE 1 : Aerosol emission against melting temperatures (SEPIA sampling inside the test cubicle and extraction duct filter sampling)

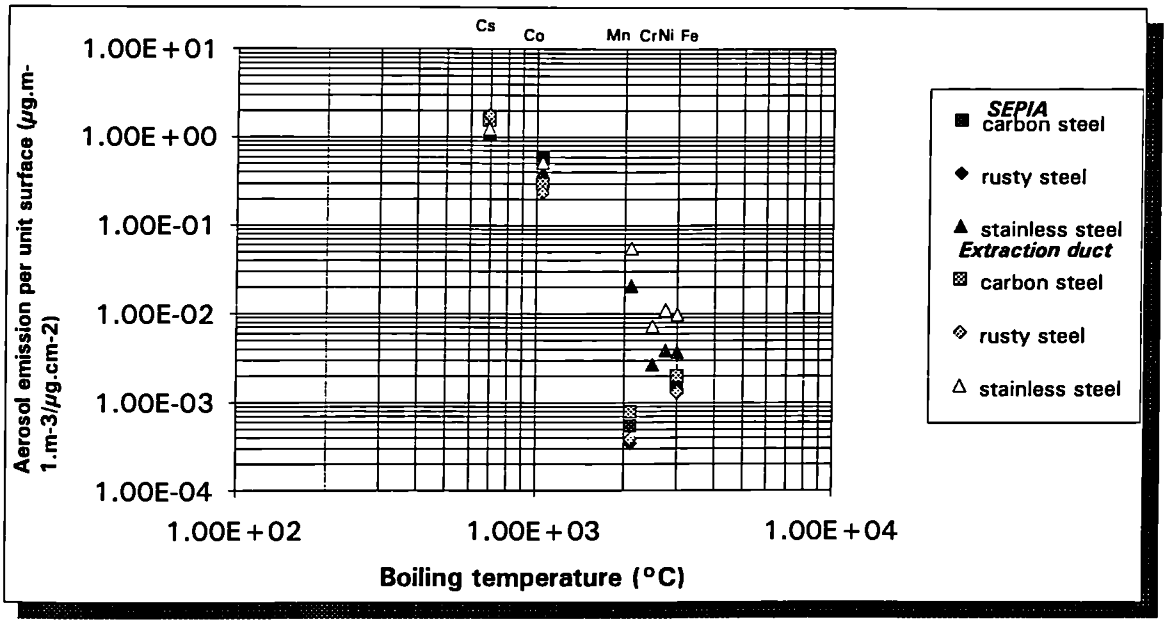


FIGURE 2 : Aerosol emission against boiling temperatures (SEPIA sampling inside the test cubicle and extraction duct filter sampling)

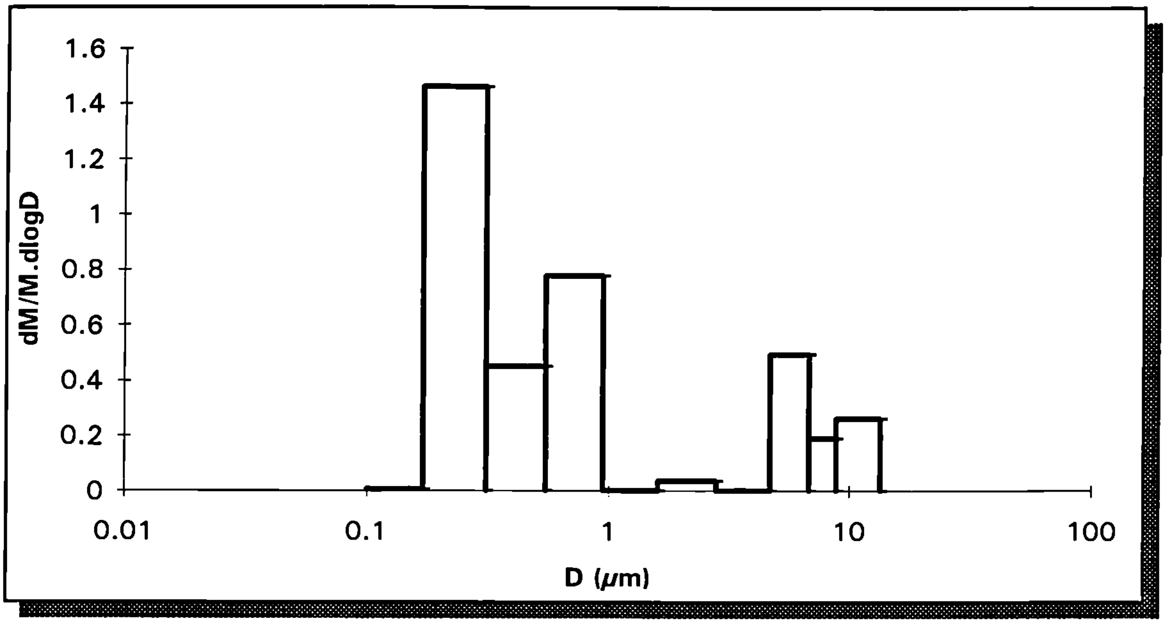


FIGURE 3: S.D.I. sampling - particle size distribution carbon steel experiments

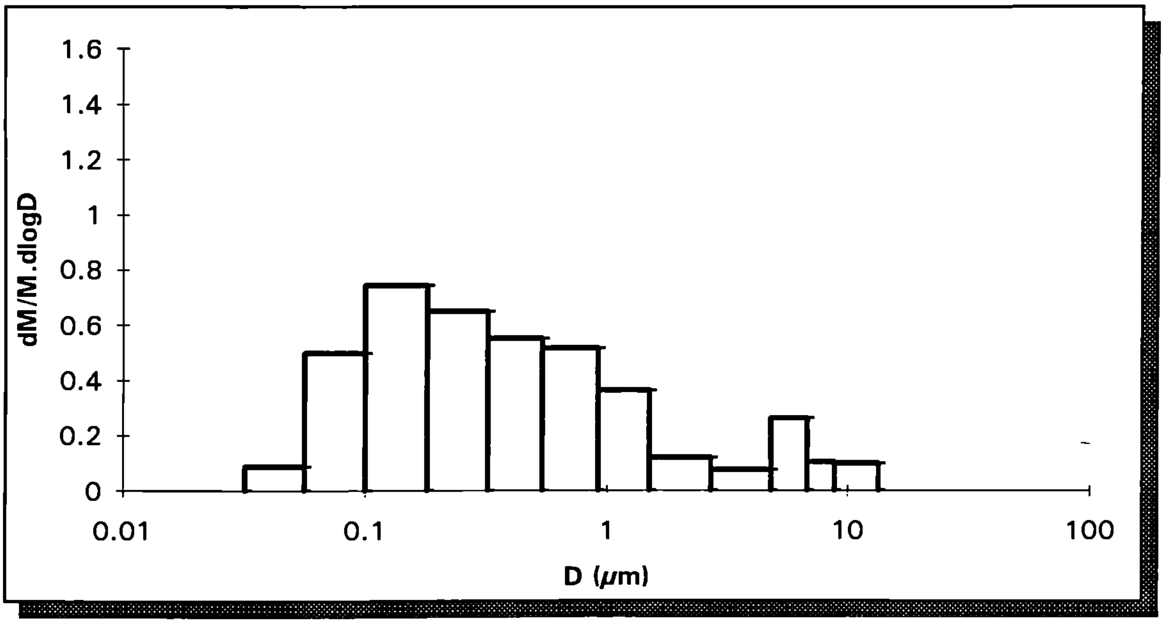


FIGURE 4 : S.D.I. sampling - particle size distribution stainless steel experiments

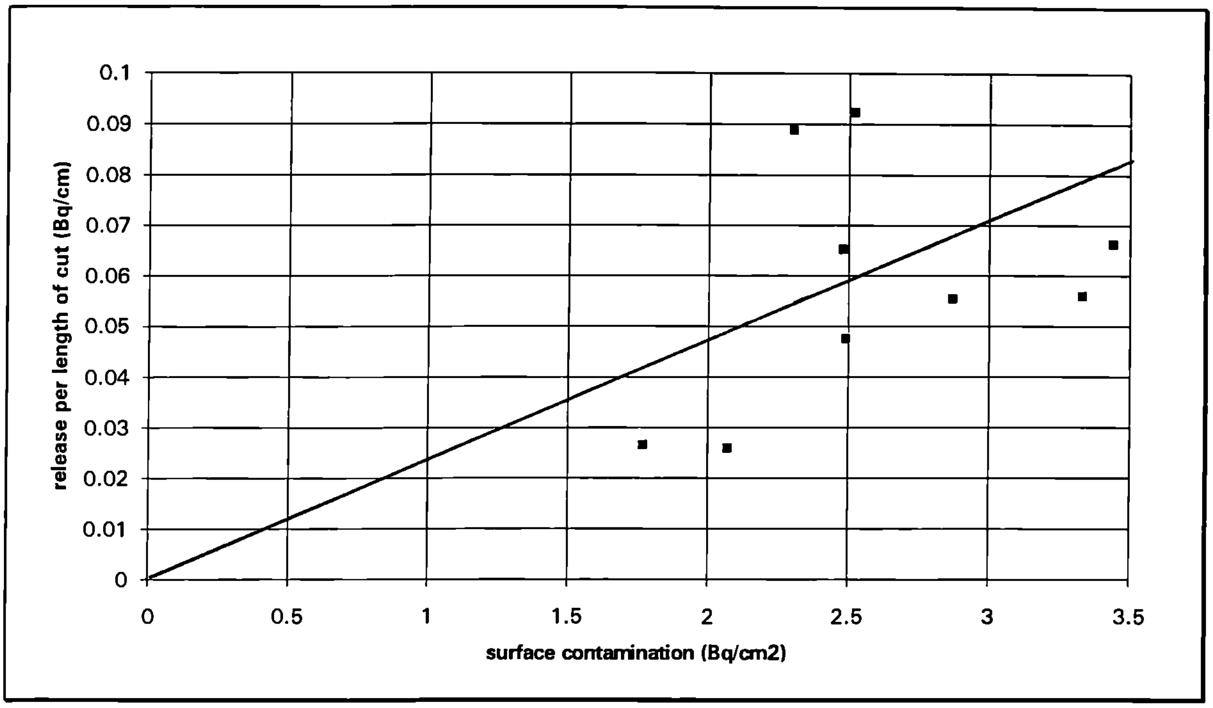


FIGURE 5 : Radioactivity release during thermal cutting of austenitic material contaminated with UO_2 powder

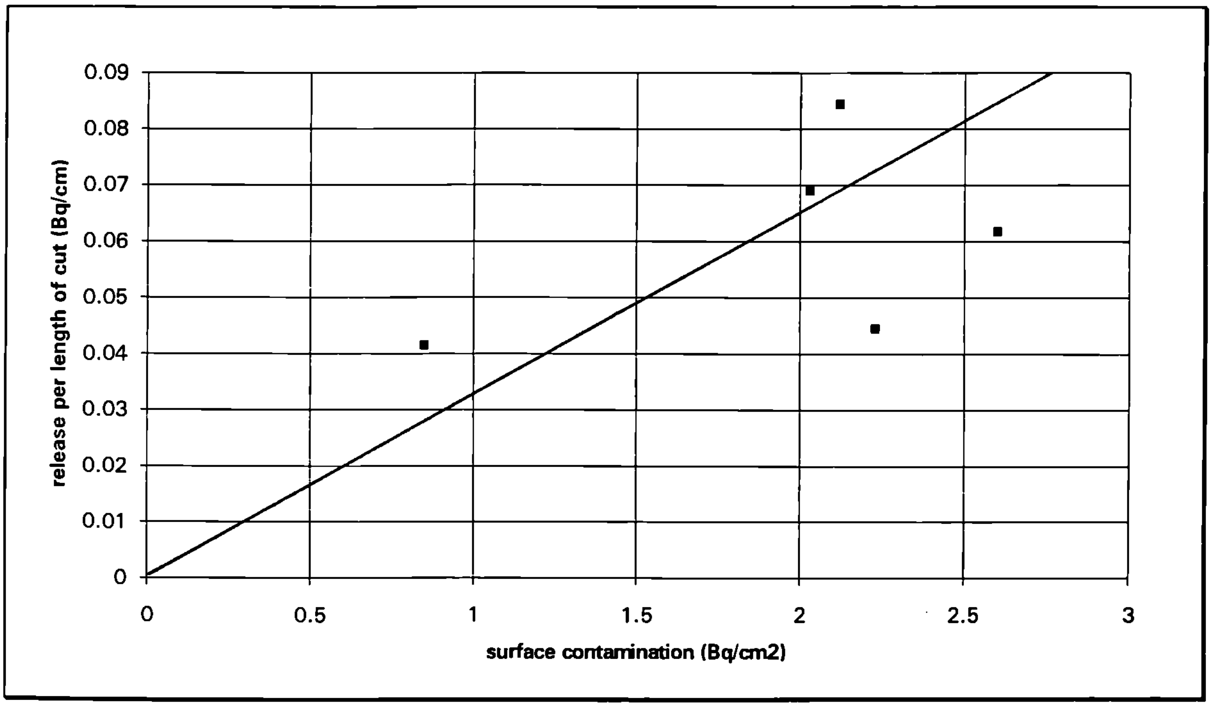


FIGURE 6 : Radioactivity release during thermal cutting of austenitic material contaminated with U-nitrate

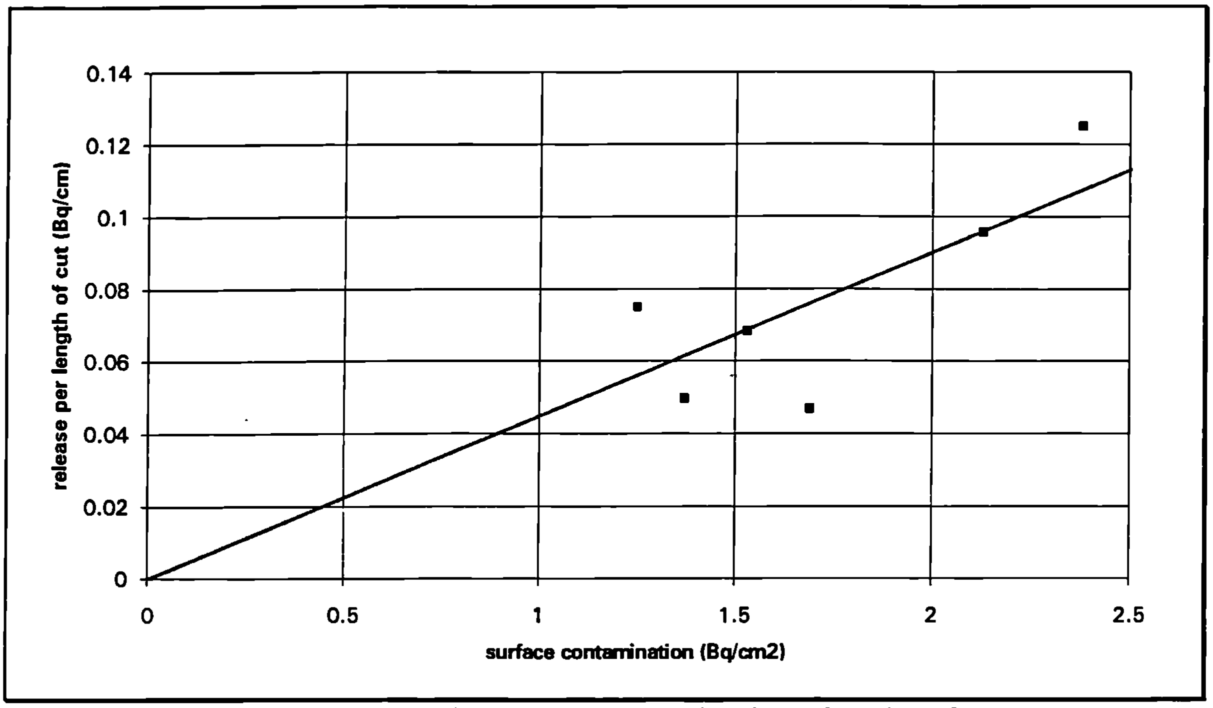


FIGURE 7 : Radioactivity release during thermal cutting of ferritic material contaminated with UO_2 powder

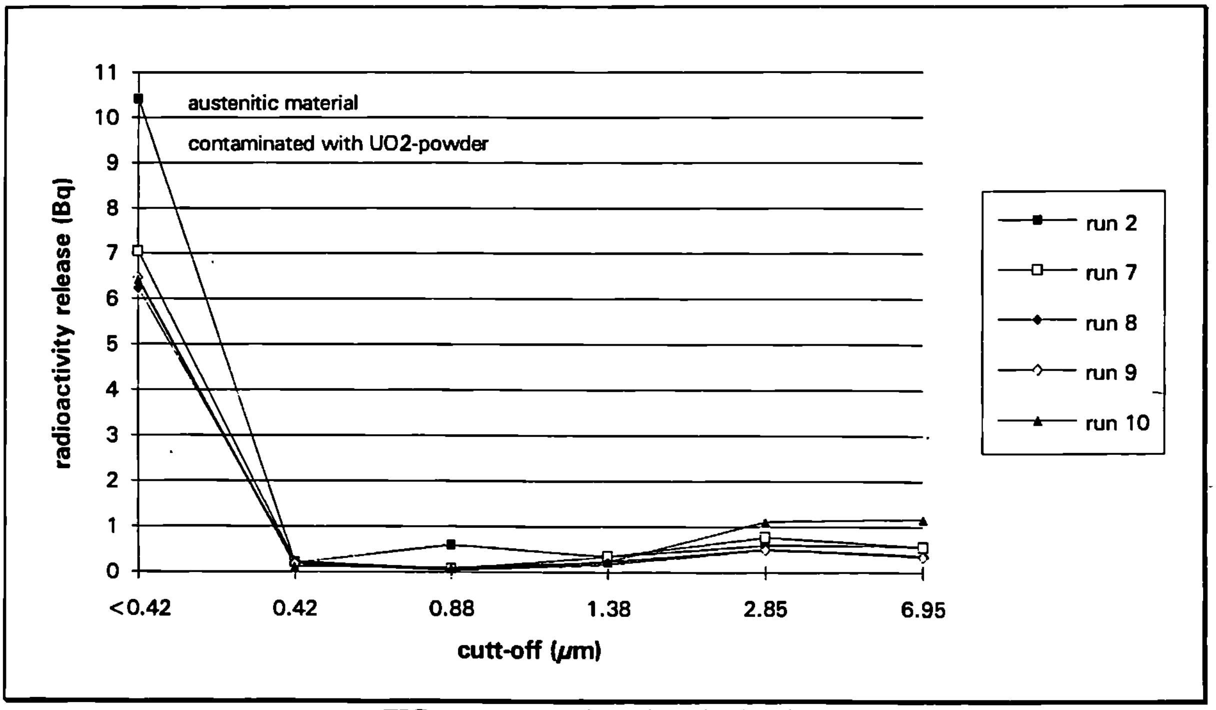


FIGURE 8 : Particle size distribution

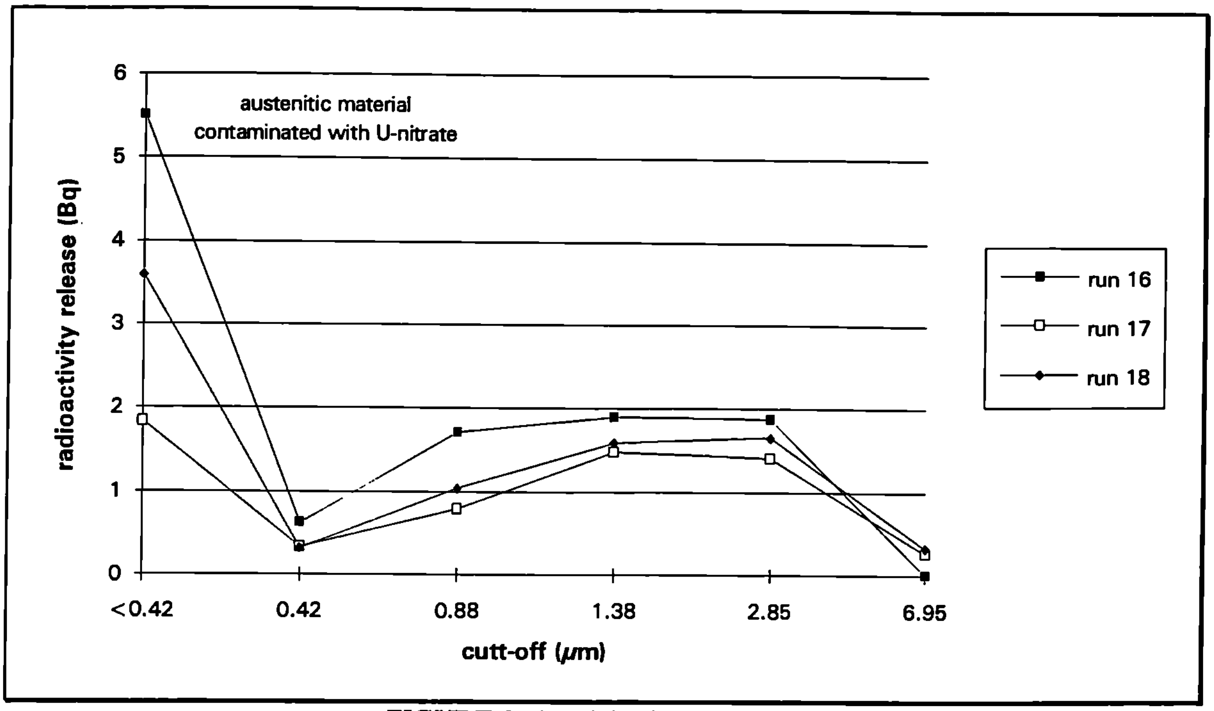


FIGURE 9 : Particle size distribution

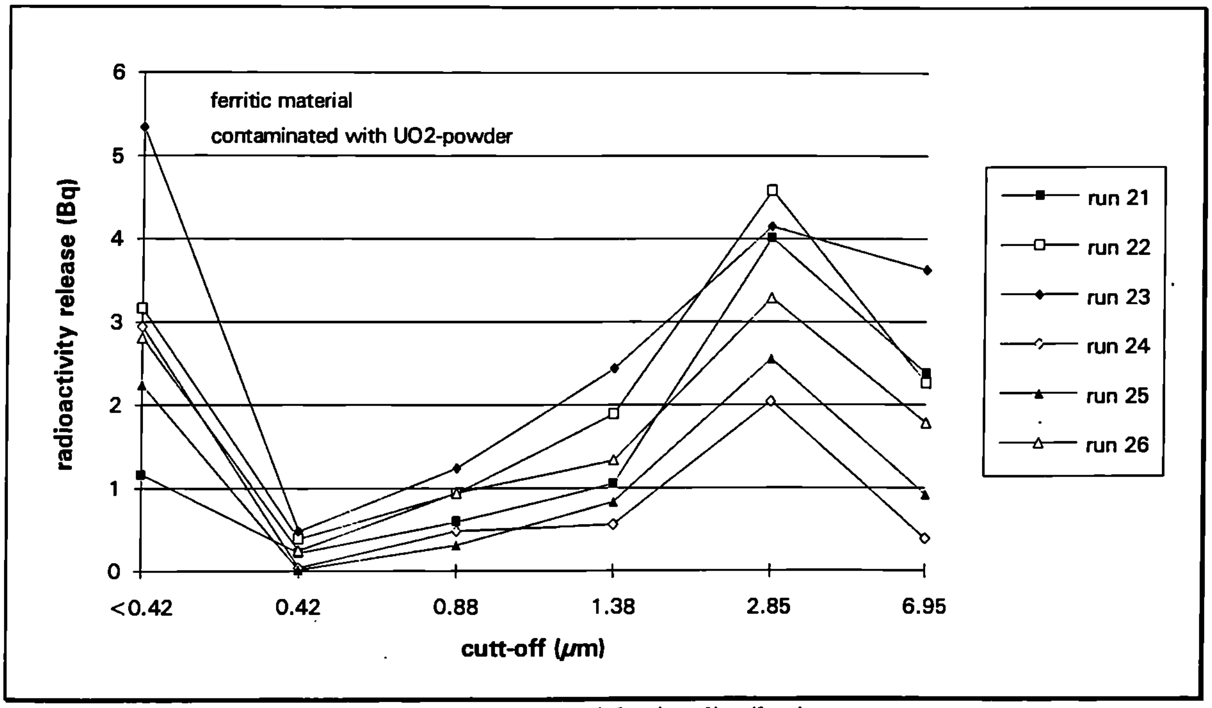


FIGURE 10 : Particle size distribution

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: 49/621/395 475 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

The aim of this research contract is to find methods for determination of beta emitting radionuclides, especially electron capture nuclides and x-ray emitters. These determination methods should be a combination of chemical separation methods and measuring methods.

The determination mainly concerns radionuclides such as Fe-55, Ni-63 and Ca-41 being occured in waste during decommissioning of nuclear facilities. To simplify the handling and the classification for reuse of waste materials it is necessary to determine the amount of these radionuclides.

For this reason amounts of contaminated steel and concrete samples from nuclear power stations were gathered. The samples were analysed by gammaspectroscopy and the alpha and beta activity were measured. The chemical separation and preparation of Ca-41, Ni-63 and Fe-55 samples was carried out at Fachhochschule Gießen. After calibration of the liquid scintillation counter it was possible to determine the amount of Fe-55, Ni-63 and Ca-41 in the above mentioned samples.

Progress and results

1. Acquisition of instrumentation (B.1.)

For the determination of the activity of beta and electron captures it is necessary to use a liquid scintillation counter. To receive reliable results it is necessary to calibrate the liquid scintillation counter because the counting efficiency of the solvent-solution system can be affected by many different factors which may reduce detection efficiency.

The calibration must be carried out with standards of Fe-55, Ni-63 and Ca-41.

2. Choice and procurement of representative samples from the reactors KKP, KKN, WAGR* (B.2.)

The samples for determination of radionuclides by quick measuring methods were obtained from nuclear power plants during decommissioning and refueling. Five samples were taken during refueling in 1989 and 1990. These samples were parts of the reactor cooling system contaminated by activated corrosion and fission products and pieces of concrete from the nuclear power plant of Philippsburg unit 1 (BWR).

KKP - Kernkraftwerk Philippsburg, Germany
KKN - Kernkraftwerk Niederaichbach, Germany
WAGR - Windscale Advanced Gascooled Reactor, UK

Thirty samples were taken during dismantling of the Niederaichbach power plant. The concrete samples were parts of the biological shield and adjoining areas, containing neutron induced activation products.

We also obtained the samples from Harwell Laboratory from the inner (active) shell of the bioshield of WAGR.

3. Laboratory activities, reference measurements and correlation calculations for nuclear determinations on decommissioning wastes (B.3.)

The chemical processing of the concrete and steel samples were carried out by the radiochemical laboratory in the Fachhochschule Gießen. The samples were wetly decomposed, chemically processed and loaded through an anion-exchanger. After separation of Fe-55, Ni-63 and Ca-41 the samples were prepared for liquid scintillation counting with the scintillation cocktail.

Before starting the measurement of Ca-41 with the liquid scintillation counter it is necessary to calibrate it with a Ca-41 standard. A source of Ca-41 was not available so that it was necessary to try to produce a Ca-41 standard by ourselves.

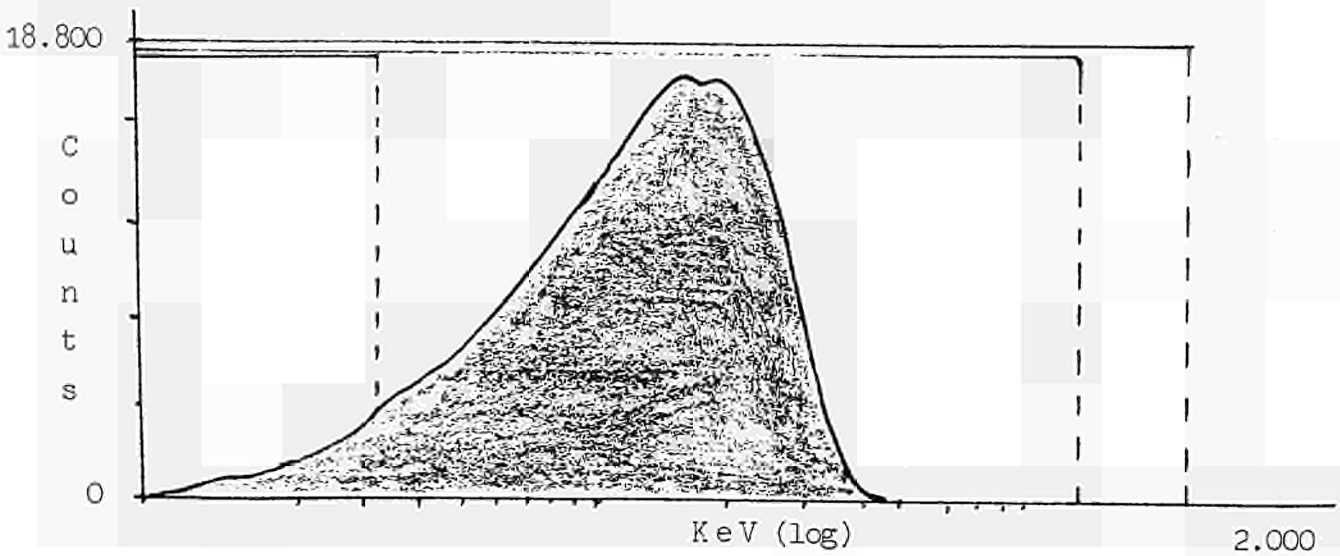
A quartz glass ampoule was filled with 2.5 g waterfree calcium carbonate, evacuated and airtight closed.

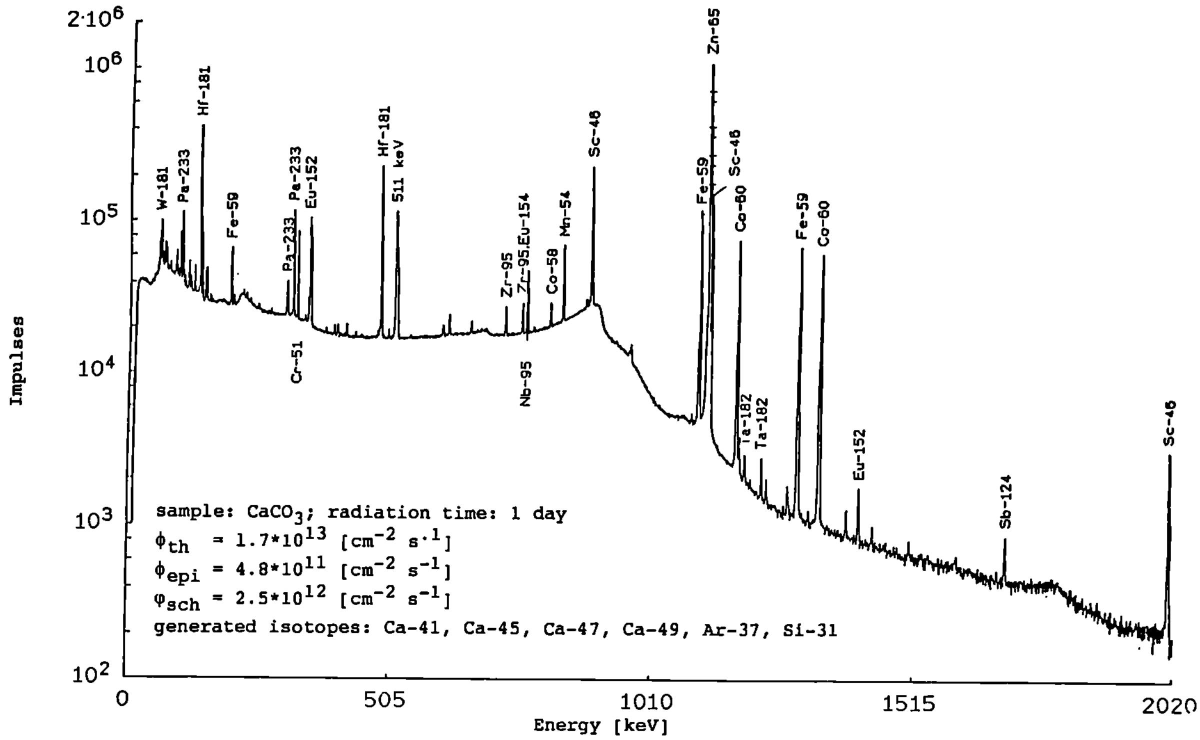
The sample was exposed to neutron radiation at the Technische Universität München/Garching to produce Ca-41.

Besides Ca-41 a more relevant radionuclide was produced, Ca-45. The activity of Ca-45 is about 10^3 times that of the activity of Ca-41. The maximum Ca-45 beta energy of 252 keV corresponds to 5,8 keV of Ca-41.

The spectrum of Ca-41 standard is shown in figure 1.

Figure 1: Spectrum of Ca-41 standard





In the spectrum of the Ca-41 standard only the peak of Ca-45 can be identified. Ca-41 is hidden by Ca-45.

Concluding from this there is no possibility for a calibration of the liquid scintillation counter with the Ca-41 standard yet. To take nevertheless advantage of the produced Ca-41 standard it is necessary to wait up the decay of Ca-45 until its activity is as low as the activity of Ca-41.

The calibration of liquid scintillation counter for Ca-41 measuring must be carried out with radioisotopes with similar decay characteristics.

The most useful nuclide is Fe-55 because its mode of decay is very similar.

The results of Ca-41 measuring are shown in table 1 and 2 and figure 2 and 3.

Table 1: Results of determination of Ca-41 in samples of concrete of KKN

Sample No.	Quantity Concrete [g]	Sample Volume [ml]	Activity		
			[Bq]	[Bq/ sample]	[Bq/g concrete]
A1	4	15	0	0	0
A2	4	15	0	0	0
B	4	30	0	0	0
C	4	30	0.4	1.04	0.26
D	4	30	0	0	0
E	4	30	0.03	0.07	0.01
F	4	30	0	0	0
G	4	30	1.95	5.91	1.5
H	4	30	0	0	0
I	4	30	0.06	0.17	0.04
J	4	30	0	0	0

Figure 2: Results of determination of Ca-41 in the sample of concrete of KKN

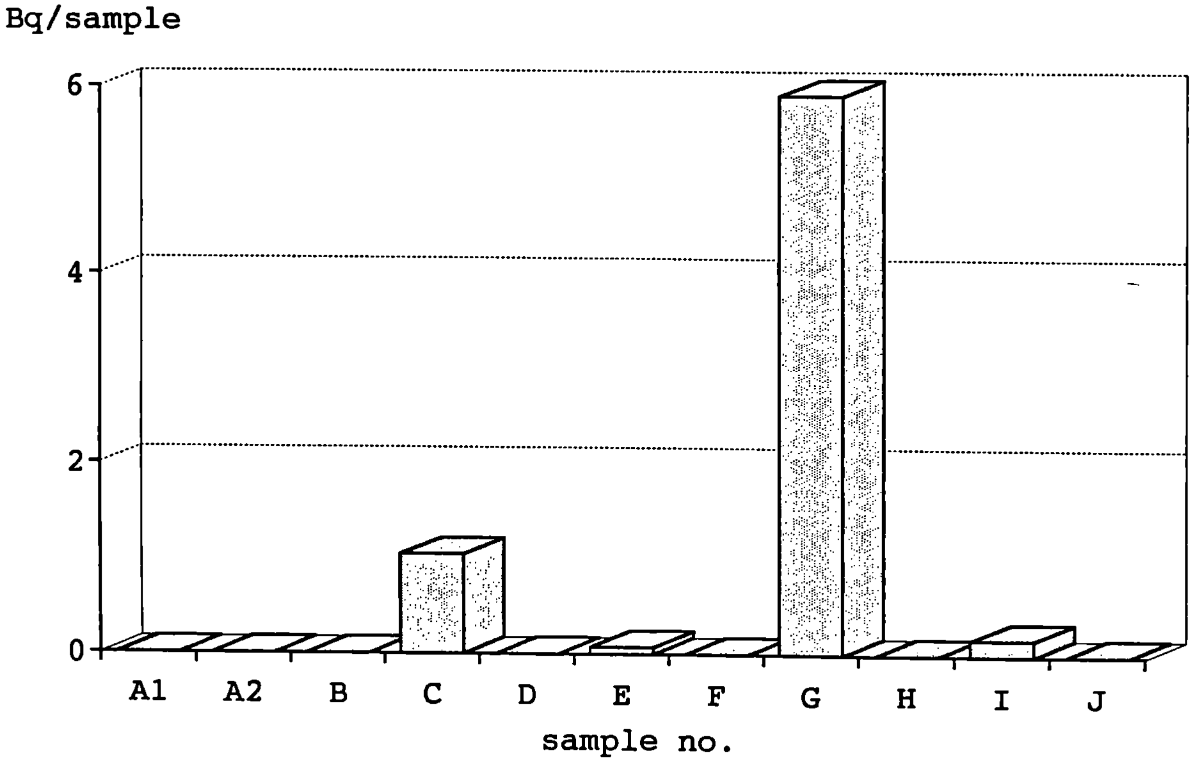
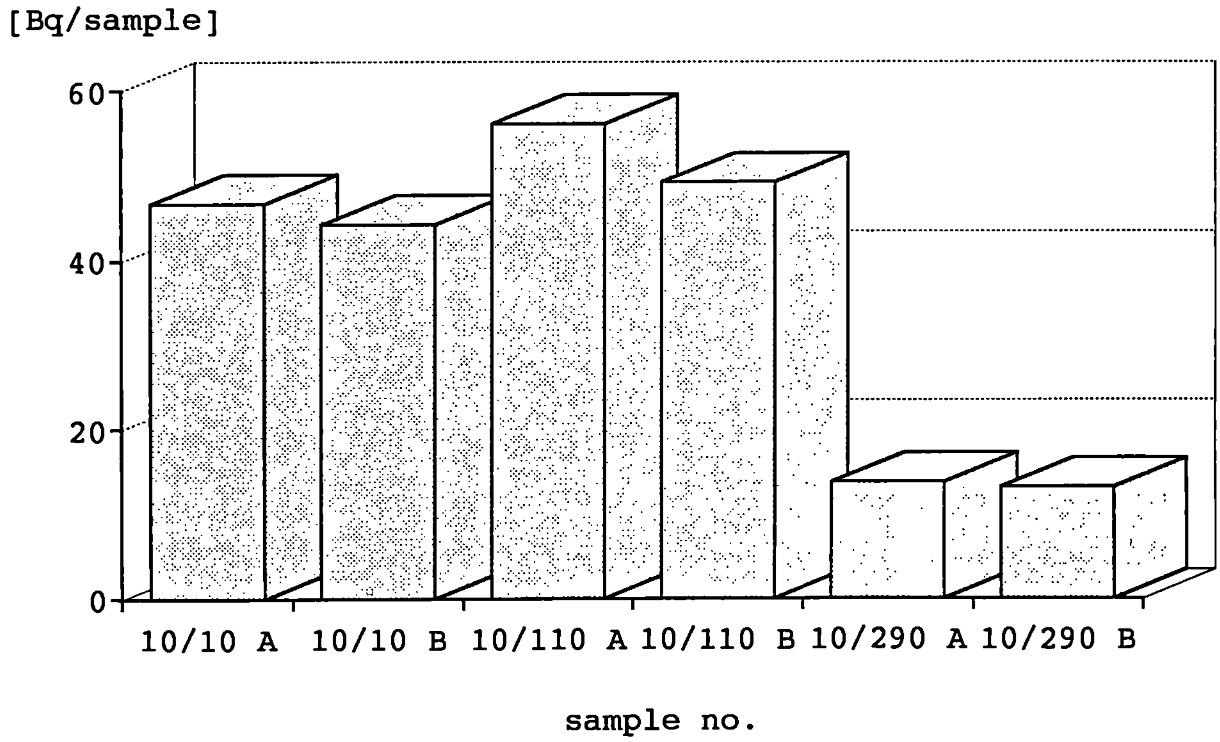


Table 2: Results of determination of Ca-41 in the sample of concrete from WAGR

Sample No.	Ca sample [mg]	Ca content [mg]	Ca content [%]	Activity of Ca-41 [Bq]	Activity of Ca-41 [Bq/sample]	Activity of Ca-41 [Bq/g]
10/10 A	121.3	122.6	101.1	47.4	46.7	126
10/10 B	121.3	120.2	99.0	43.8	44.2	119
10/110 A	87.3	73.9	84.7	47.5	56.1	116
10/110 B	87.3	87.2	99.8	49.1	49.2	120
10/290 A	101.1	97.0	96.1	13.2	13.7	39
10/290 B	101.1	99.8	98.7	12.9	13.1	37

Figure 3: Results of determination of Ca-41 in the sample of concrete from WAGR



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 - June 1994
Coordinator: F J SCHMID, TÜV-Bay.
Phone: 49/89/5791 1470 Fax: 49/89/5791 2606

A. OBJECTIVE AND SCOPE

Limiting values for the release of concrete with low-level residual radioactivity for the selective undangerous utilisation (e.g. for noise barriers, earth fill, earth bank or substitute for foundation material) are presently not defined. The research programme will examine whether it is possible to define limiting values for radioactively contaminated concrete in the range of the limiting values for steel. The effect of radioactively contaminated concrete on the soil (leach out of radionuclides) and on man (radiation exposure) will be determined.

The results of these studies will have an effect on the decommissioning activities as far as buildings of the controlled area and the kind and quantity of the radioactively contaminated concrete are concerned.

The advantage of the studies lies in an economic and safe recycling of large amounts of concrete with a low-level artificial residual radioactivity. Thereby, valuable ground storage space would be saved and natural gravel deposits would be preserved.

The research work will provide data concerning cost saving by recycling concrete from controlled areas, radiation exposure of the decommissioning workers and of the general public.

The research programme is performed in co-operation with CEA-IPSN, which has a research programme with a similar objective (see § 6.4.).

B. WORK PROGRAMME

B.1. Leach tests
B.1.1. Design of the test facility and determination of concrete test specimen. (all)
B.1.2. Construction and operation of the test facility. (TÜV-Bay.)
B.1.3. Literature survey on leaching out problems of radionuclides in concrete. (TÜV-Bay.)
B.1.4. Radiological measurements on concrete rubble before, during and after leach out tests. (TÜV-Bay.)
B.2. Natural radioactivity in concrete
B.2.1. Procurement of samples from recently produced and aged concrete. (RWE)
B.2.2. Measurement of alpha, beta and gamma radiation. (TÜV-Bay.)
B.2.3. Literature survey concerning the natural radioactivity of concrete.
B.3. Development of methods for recycling concrete.
B.3.1. Examination of concrete recycling possibilities by a literature study. (RWE)
B.4. Calculation of radiation exposure and determination of the artificial residual radioactivity
B.4.1. Determination of radiation exposure scenarios. (TÜV-Bay.)
B.4.2. Calculation of radiation exposure for man due to natural and artificial radioactivity. (TÜV-Bay.)
B.4.3. Derivation of criteria for the safe use of concrete with artificial radioactivity. (TÜV-Bay.)

C. Progress of work and results obtained

Summary of main issues

After the general overhauling of the test facility, the wash-out-tests have been continued and finished in September 1993.

The collecting and the measurements on the samples of concrete from conventional buildings have been completed. Their final results are given.

The literature study on the development of methods for recycling concrete has been finished.

Progress and results

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

As already announced in the last report, a general overhauling of the water supply system of the test facility was realized.The supply system has been changed in a way that even if the water flow would be influenced by deposits in the ducts the constant water amount per cycle is guaranteed.

To reach the intended 20000 mm precipitation amount the number of cycles was increased up to 3 cycles per week and also the precipitation amount per cycle was increased to 160 l of water. Besides this it was decided to collect the water of 8 sprinkling cycles and to connect them to one measuring unit. In March the wash-out-tests were started again with these new parameters of precipitation.

The test has been finished in September and the test facility was shut down. The measurements of the samples of wash-out water, sediment and of the test specimen itself have been continued.

2. Natural activity of concrete (B. 2)

In 1993 the collecting of samples of aged and freshly mixed concrete has been completed with 15 additional samples from Northern Germany. Overall 99 samples have been collected, 52 of the samples are freshly mixed concrete, 47 are aged concrete. Their age ranges between 4 and 70 years.

The results of the statistical interpretation show that there are no significant differences from one group of samples to the other, neither from freshly mixed to aged concrete nor between the different regions from wich the samples are taken. This is the reason why in Table I only the statistical values of all samples together are given. Contrary to the tables in former reports the activity of U-235 is no longer shown. Dealing with concrete from conventional buildings, one can assume the uranium isotopes to be in the natural relation. The given values in former reports with their uncertainties seemed to show an enrichment of U-235 that is not real but only an effect of the method of interpretation of the measurements.

In Table II the measurement results (mean value) of TÜV Bayern Sachsen are compared with the values of the natural activity of concrete given by other authors (B.2.3). One can see that most of these values are in the range of one standard deviation of the mean value of the TÜV Bayern Sachsen measurements.

3. Development of methods for recycling concrete (B.3)

In the frame of this literature study 146 references concerning the different parts of the recycling of concrete have been evaluated. The results of the study give a summary of the background of recycling in law and economy. In detail the different methods of crushing and preparation of concrete are given. Further parameters as effectiveness, costs, dimension and emissions of a recycling facility are given.

Furtheron the possibilities of the reuse of the recycled concrete in construction and civil engineering are summarized. This includes information about the quality and physical parameters of new structures built under use of recycled concrete. These informations in combination with the demands of the standard specification show the limits of the possible use.The literature study itself is now finished, in the moment the report of the results is in proof-reading.

4. <u>Calculation of radiation exposure and determination of the artificial residual activity (B.4)</u>

Based on the results of the literature study (B.3) the parameters of the exposure scenarios for workers dealing with the concrete recycling and also for the population are now fixed. With these parameters the calculations of the doses resulting from the reuse of the contaminated concrete will be done.

Table I: Natural activity of concrete

Nuclide/ Decay-series	min	max	Mean value	1 σ	Median
	specific activity in Bq/kg				
U-238	6,3	90	16,1	9,3	14
Th-232	3,8	88	15,4	9,8	14
K-40	28	749	246	113,6	242

Table II: Comparison of measured natural activity of TÜV Bayern Sachsen with results of other authors

	Number of samples	U-238 (Ra-226)	Th-232	K-40
		specific activity in Bq/kg		
TÜV measurements	99	16,1 ± 9,3	15,4 ± 9,8	246 ± 113
/Kri 81/	5	20 - 49	30 - 57	312 - 979
/Kri 87/	8	7 - 15	4 - 18	-
/Mut 84/	?	26	22	333
/UNS 77/	69	67	63	555

<u>References</u>

/Kri 81/ Krieger R., Radioaktivität von Baustoffen
Betonwerk + Fertigteil-Technik 8/1981, p. 468-473

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Beton 2/1987, p. 55-58

/Mut 84/ Muth H.,Keller G., Ergebnisbericht über Untersuchungen und Messungen der Konzentration natürlicher radioaktiver Stoffe und der Radon- und Thoron-Exhalationsraten verschiedener Bimsbaustoffe
Boris-Rajewski-Institut 1984, p. 1-13

/UNS 77/ United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), Sources and Effects of Ionizing Radiation, Annex B
New York 1977

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 RESULTS OBTAINED

This study was completed in 1992. The final report is being prepared for publication.

6.6. THE CHARACTERISATION AND DETERMINATION OF RADIOACTIVE WASTE FROM DECOMMISSIONING

Contractors: AEA-Harwell
Contract No.: FI2D-0042
Work Period: January 1991 - March 1993
Coordinator: J W McMILLAN, AEA Technology, Harwell
Phone: 44/235/43 48 53 Fax: 44/235/43 29 77

A. OBJECTIVE AND SCOPE

The objective of this laboratory-scale experimental investigation is to develop the "fingerprint" method for the characterisation of waste arising from decommissioning projects to the point where it could be used more extensively after its initial limited application.

The "fingerprint" method relies on the ability to carry out comprehensive analysis, for all of a specified range of radionuclides on a statistically justified set of samples. In order to achieve this, development of several aspects, related particularly to the difficulty of measuring some specific electron capture and low energy beta-emitting nuclides, is required.

It is expected that the establishment of accurate fingerprints, when coupled with simple measurement of the total activity, will enable correct sentencing of waste and thus minimise the cost of the disposal of the waste arising from the decommissioning of radiochemical laboratories or reactor facilities.

Collaboration is envisaged with TÜV Südwest, FRG (contract No. FI2D-0033), both on the development of radiochemical methods and on the assessment of measurement techniques.

B. WORK PROGRAMME

B.1. Acquisition of contaminated material and fabrication of simulants
B.2. Development of methods for the removal of radioactive contaminants to solution
B.2.1. Survey on existing methods for removal of radioactivity from contaminated materials.
B.2.2. Leaching experiments.
B.2.3. Investigation on microwave dissolution techniques.
B.2.4. Investigation on electrolysis for the recovery of tritium.
B.3. Development of preconcentration, separation and analysis methods
B.3.1. Selection and commissioning of slow injection analysis.
B.3.2. Development of methods for Fe, Ni and U.
B.3.3. Development of method for I.
B.3.4. Development of method for Ca.
B.3.5. Investigation on combustion techniques for the recovery of carbon and hydrogen
B.4. Development of counting methods for the difficult-to-measure nuclides
B.4.1. Development of liquid scintillation.
B.4.2. Development of gas proportional counting.
B.4.3. Development of x-ray counting
B.5. Statistical assessment to characterise the waste and satisfy the quality assurance standards

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1993. The final report is being prepared for publication.

6.7. QUANTIFICATION OF ACTIVITY LEVELS AND OPTIMISATION OF DOSE RATE MANAGEMENT TO PREPARE STAGE 3 DECOMMISSIONING OF GAS-COOLED REACTORS

Contractors: CEA-VALRHÔ, Radia
Contract No.: FI2D-0044
Work Period: October 1990 - June 1994
Coordinator: J R COSTES, CEA/DCC/UDIN, Bagnols-sur-Cèze
Phone: 33/66 79 63 13 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

As part of the preparatory work for Stage 3 decommissioning of the G2/G3 gas-cooled reactors at Marcoule, the project involves:
- quantifying the activity levels of complex core structures based on theoretical analysis and on a large number of dose rate measurements;
- design of a software package to optimise the dismantling and related operations and best minimising of the dose rates incurred by the personnel.

It is important to determine the dose rates and time necessary on each manual dismantling operation, and to assess the material activity levels for optimum waste conditioning and disposal.

The development of suitable software tools and thorough examination of all the possible scenarios are very time-consuming undertakings requiring aid beyond national boundaries.

B. WORK PROGRAMME

B.1. Dose rate measurements and analyses of core samples after a literature review (CEA).
B.2. Analyses of geometric, physical and radiological data (Radia)
B.3. Development of a computer programme to calculate gamma-activity levels in the entire core (Radia)
B.4. Development of a computer programme to minimise dose rates during human interventions (CEA)
B.5. Comparison of calculated and measured results (CEA).
B.6. Examination of dismantling scenarios (CEA).
B.7. Revision of expert software considering decommissioning time and doses (CEA).
B.8. Evaluation of costs, radioactive job doses, working time and secondary waste arisings (All).

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of Main Issues

Decommissioning a Gas-Cooled Reactor to stage 3 will require intervention inside the reactor pressure vessel. This is not a simple operation. The following scenarios were investigated:

a) Removal of graphite bricks individually; several methods may be considered:
   - using a conventional overhead crane and a derrick, after creating a massive breach in the top of the vessel,
   - using robot devices to allow a smaller breach; this scenario implies coordinating several types of robots for specific tasks, and does not appear to be a feasible solution at this time.
   - using divers after flooding the core.

b) Graphite destruction by *in situ* laser-controlled incineration.

The last two scenarios, currently regarded as the most plausible options, will be examined here in further detail.

Progress and Results

Task B6 Decommissioning Scenarios

1. Graphite Destruction by in situ Laser-Controlled Incineration

If graphite incineration is authorized, it could be economically advantageous to incinerate it directly *in situ* to benefit from the extraordinary shielding provided by the prestressed concrete reactor pressure vessel and its leaktight skin. (see figure point 1).

This procedure would avoid removal and milling of the graphite blocks as well as the construction of a specialized facility. However, it would require a high degree of local control of combustion and very sophisticated monitoring systems.

Principle

An extremely pure grade of graphite was used in the reactor vessel (100 ppm of impurities) – and it was designed to resist incineration. Successful incineration requires temperatures above 1200°C, and preferably in the 1500–2000°C range. The reaction can only be sustained on the surface by supplying enough oxygen and removing the CO_2. With these constraints, a target rate of 150 $kg{\cdot}h^{-1}$ (i.e. 1½ bricks per hour) may be considered feasible.

The incineration process must be extremely safe. This may be ensured by observing the following precautions:

- The bricks (see figure point 2) would be incinerated one at a time by illumination with a 30 kW CO_2 laser beam (see figure point 3). If the beam is shut off, combustion will cease. The high temperature zone is thus strictly limited to the outer surface on one face of a single brick.
- The temperature of the remaining bricks must not exceed 150°C. This condition may be met by exhausting the air and combustion gases via a particle scrubber and HEPA filters, so that only CO_2 and trace amounts of ^{14}C are released.

Penetration of the Reactor Vessel

The simplest solution would be to penetrate through boreholes (see figure point 8) from the top of the vessel. The horizontal cross section of the reactor vessel is practically square (9.05 × 9.53 m), so the number of boreholes would be a function of n^2: 1, 4 or 9. Four evenly spaced boreholes would imply a maximum horizontal working distance of 3.28 m, compared with 2.18 m for nine boreholes.

From the standpoint of radiological safety, the best solution would call for the minimum number of boreholes with the smallest possible diameter to facilitate containment. Conversely, the solution that would favor access to all the bricks using the simplest, most compact working arm would involve a maximum number of boreholes or a full-length hopper.

Laser Beam Transport

The laser beam is guided inside the robot arm (see figure point 6). Each joint requires at least two 45° reflections by fixed mirrors. The beam is aimed along the rotation axis, and external (or hollow) joints are used. A similar solution is used by the *Rold* robot.

This option implies a number of constraints. The mirrors must be water-cooled if high surface densities are used. Preventing particle deposits (dust or smoke) on the mirrors is a major concern, as the resulting energy accumulation would damage the reflecting surface (the reflection coefficient diminishes as the absorption coefficient increases). For this reason (and for general nuclearization purposes) the arm must be maintained at a slight overpressure at all times, even when not in operation.

The mirrors would be replaced at regular intervals for maintenance purposes throughout the beam transport system. Mirrors designed for easy installation and removal and with provisions for accurate positioning are available. Modular design must be a basic criterion for the laser beam transport arm.

Incineration Procedure

An investigation of *in situ* graphite incineration and heat exchange phenomena suggests a number of avenues for optimizing the incineration procedure, by allowing for the following constraints:

- local temperature rises due to the incineration reaction may be of advantage in optimizing graphite preheating (see figure point 5);
- heating must be limited outside the incineration zone to maintain control of the reaction and to meet safety requirements;
- the safety of the mechanical arm must be taken into account: it must be kept away from hot surfaces, the graphite structure must be monitored (to prevent possible collapses) and care must be taken in moving the reactor heat shields (to avoid excessive pressure on the heat shields by the spring units).

Provisional Conclusion Concerning in situ *Laser Incineration*

Laboratory and subsequent pilot-scale tests in conjunction with theoretical studies using a simulator have demonstrated the feasibility of this technique. This incineration method has substantial advantages over other conventional solutions: the bricks are incinerated without prior milling and without removal from the core. Moreover, it entails extremely low occupational dose rates (except for robot arm maintenance and waste ash handling procedures).

Nevertheless, additional development work, notably concerning off-gas treatment and structural cutting methods, will be required before a decision can be made.

2. Dismantling of Reactor Core by Divers

French technicians have acquired considerable diving expertise during numerous maintenance operations in spent fuel storage ponds and fuel dismantling ponds. This experience suggests that it would be possible to dismantle a GCR core under water.

The use of divers has a number of serious advantages in the nuclear industry: collective dose optimization (the ^{60}Co dose rate is halved by 11 cm of water), lower operating costs and shorter down time.

Preliminary Steps

In addition to legal obligations governing the preparation of work sites (400 and 4000-hour limits), the use of divers implies the following preliminary steps:

- preparation of a detailed task description and technical documentation including drawings and procedural diagrams;
- assessment of radiological hazards by determining the activity concentration and nature of the radionuclides in the water, with irradiation mapping of zones accessible under water and of workstations situated around the water volume;
- determination of the physical and chemical properties of the water, notably the temperature and pH;
- assessment of potential risks (radiological, electrical, chemical, mechanical, etc.) in the water and on the surface;
- decontamination of the water to the lowest feasible level using existing or additional means;
- specification of the means required to obtain the highest possible water quality (temperature and visibility);
- determination of the optimum diver access points (by modifying the water level if necessary);
- preparation of a draft operating procedure with allowance for protective measures and restrictions;
- preparatory meeting preceded or followed by an inspection visit;
- distribution of the operating procedure;
- drafting of documents for site preparation, monitoring and shutdown;
- monitoring of respiratory air quality;
- joint acknowledgement of equipment compliance and satisfactory condition.

Cutting of Openings in the Reactor Vessel

Access to the reactor will only be possible after cutting two openings on the top of the reactor casing. After installing the containment, and working facilities, connecting them and the reactor to the ventilation system and flooding the reactor vessel, two openings (each measuring 2.50 × 3.50 m) will be cut at the same positions as the openings used during the construction of the reactor. One will be used exclusively for access by the divers, and the other for removal of equipment and waste.

Irradiation Hazard

The estimated activity of the water at equilibrium is approximately 1500 $Bq{\cdot}cm^{-3}$ for ^{60}Co. This is the only radionuclide that must be taken into account for the diver irradiation hazard analysis. The dose rate from the water is therefore estimated to be about 0.9 $mGy{\cdot}h^{-1}$.

The exposure dose rate from the reactor internals under the planned operating conditions would be between 40 and 400 $mGy{\cdot}h^{-1}$. The consequences of a possible accidental irradiation dose rate of 100 $mGy{\cdot}h^{-1}$ will be limited by the active sensors worn by the divers.

Contamination Hazard

Based on prior experience, the activity due to tritium, ^{14}C and ^{60}Co should not represent a contamination hazard for the divers. The static and dynamic containment provisions will be designed to prevent any dissemination during transfers. Nevertheless, the contamination will be appreciable: dose rates of 0.01 to 0.1 $mGy{\cdot}h^{-1}$ have been measured after diving in water with an activity level of 2000 $Bq{\cdot}cm^{-3}$.

Based on the measured dose rates and activity levels, and on the planned operating procedures, the radiological cost should be as follows:

- diving 700 man-mSv
- surface 120 man-mSv
- waste handling 130 man-mSv

Total 950 man-mSv

Provisional Conclusion Concerning Reactor Core Dismantling by Divers

Diver operations in the past have generally been limited to maintenance tasks, often in emergency situations. Application of these procedures to much longer and less urgent decommissioning operations will require extensive preparation and concern for accident prevention. It would be advisable to implement this technique on a GCR that has already undergone a long cooldown period. Finally, the MINODDIN code (designed to minimize occupational doses during decommissioning operations) will be used to plan the diving task sequence.

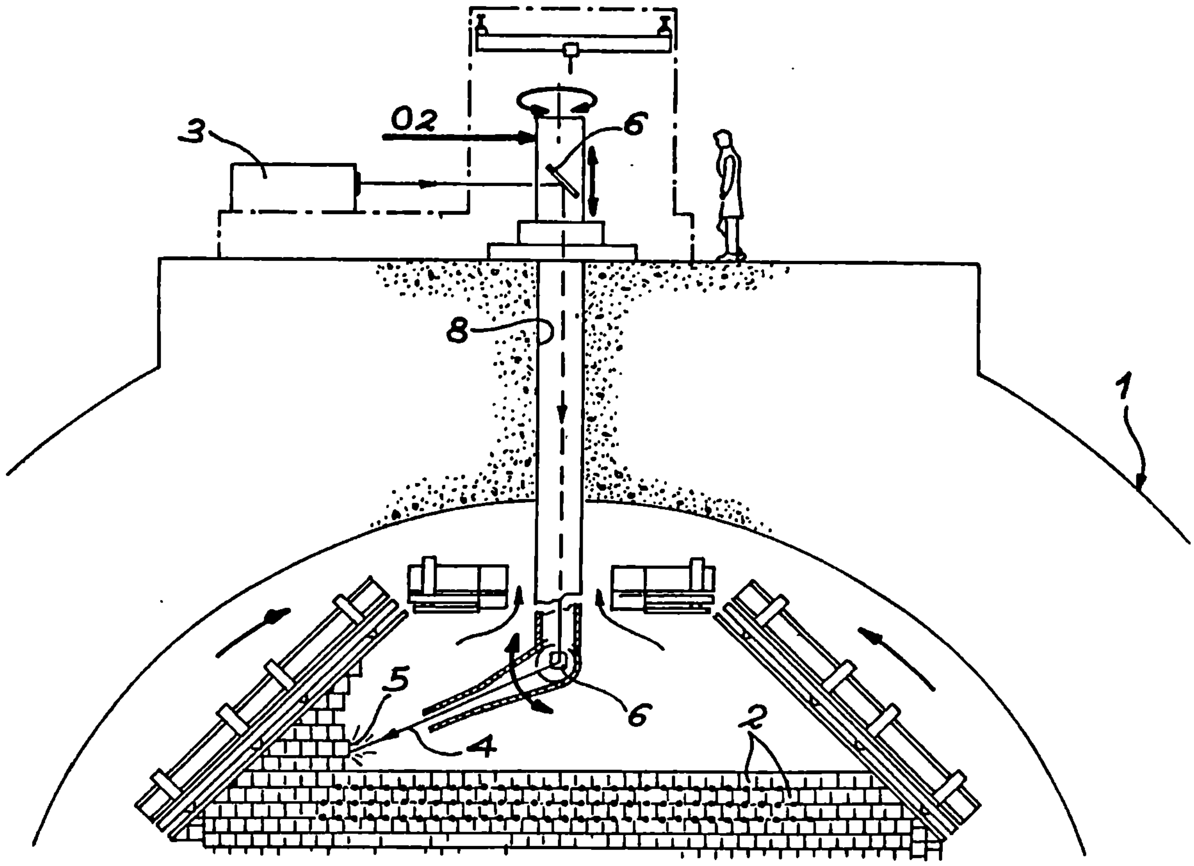


Figure: Graphite Destruction by in situ Laser Controlled Incineration

6.8. DECOMMISSIONING COSTS FOR NUCLEAR INSTALLATIONS

Contractors: NIS Ingenieurgesellschaft mbH
Contract No.: FI2D-0051
Work Period: July 1991 - December 1991
Coordinator: P PETRASCH, NIS
Phone: 49/6181/10 94 58 Fax: 49/6181/12 00 33

A. OBJECTIVE AND SCOPE

In 1977, the Commission of the European Communities initiated a study to calculate the decommissioning cost for nuclear power plants (EUR 5728d).
The main objective of this contract is to update the study on the state-of-the-art, taking into account the technical advances occurred since 1977 in the decommissioning of nuclear power plants as well as in the conditions and means to calculate the decommissioning costs. The study will focus on representative commercial German LWRs. Nevertheless, the calculation method is made in a form allowing comparison/extrapolation to the decommissioning costs of other EC nuclear installations.

B. WORK PROGRAMME

B.1. Description of the boundary conditions for the decommissioning with particular view to nuclear power plants in Germany
B.2. Detailed technical description of decommissioning concepts for a BWR (referenced by nuclear power plant Biblis)
B.3. Calculation of the decommissioning costs for the concepts given in para B.2.
B.4. Comparability with costs of other EC decommissioning projects, either originating from real projects or estimated.

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1991. Final Report is available as EUR report n° 14687 DE.

6.9. DEVELOPMENT OF A PROTOTYPE APPARATUS VISUALISING ON A SCREEN THE GAMMA SOURCES SUPERIMPOSED ON THE IMAGE OF THE VISION FIELD

Contractors: CEA Valrhô, CEA Saclay
Contract No.: FI2D-0055
Work Period: October 1991 - May 1993
Coordinator: G. IMBARD, CEA Valrhô
Phone: 33/66 79 63 10 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

The project consists in further developing a measuring device composed of a video camera, a gamma detector, an image processor and a monitor on which the radioactive radiation intensities will be superimposed on the related visual field.
The instrument (diameter <200 mm, length <400 mm and weight around 50 kg) will be handled by a specific remote-controlled support.

The scope of the programme is to produce a prototype gamma-camera that can be used in hot cells of decommissioning projects.
The development of the R&D programme will entirely be performed by the two CEA research centres, CEN-Valrhô and CEN-Saclay, with CEN-Valrhô as coordinator.

B. WORK PROGRAMME

B.1. Measuring performance optimisation of the demonstration device
B.2. Testing the apparatus under real conditions
B.3. Integration of the measuring chain in a biological protection shield
B.4. Development of the image processing software
B.5. Calibration operations
B.6. Prototype application on a decommissioning site.

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of Main Issues

Despite a setback from the initial schedule, due to a delay in delivery of an image intensifier tube meeting specification requirements, the contractual programme was completed in 1993. The following major tasks were carried out during 1993:

- integration of the detection system in biological shielding;
- development of the software processing and image superimposition functions;
- calibration of the detection circuit for various configurations;
- equipment testing under actual conditions at a decommissioning site (the RM2 facility at Fontenay-aux-Roses).

An operational prototype gamma scanning camera has been available since the end of 1993. Negotiations are currently in progress to license the technology for sale by a French company specialized in nuclear measurements

Progress and Results

1. Integration of the Measurement System in a Biological Protection Shield (B.3.)

The shielding is designed to provide effective collimation of the high-energy gamma photons, while protecting the scintillator and sensitive internal components from radiological damage; the shielding also provides mechanical protection and allows handling of the camera assembly.

Considering the operating conditions found in the hot cells and facilities concerned by decommissioning, the system was implemented in a compact, cylindrical unit 120 mm in diameter and 444 mm long, weighing 39 kg. Figure 1 shows a longitudinal cross section of the shielding, and Figure 2 is a photo of the complete detection system.

The final version includes an innovation over the initial design to enhance the angular resolution: a zoom function allows the target zone to be enlarged by moving the collimator away from the image plane (increasing the focal length between the pinhole opening and the scintillator) using an external control.

2. Development of the Image Processing Software (B.4.)

In addition to the image acquisition module developed in 1992, two specific modules were developed during 1993.

- An image processing module to modify the image obtained from the acquisition module in order to enhance the visible image, to interpret the data in the gamma image, and to allow superimposition of the visible light and gamma images. A full range of gamma and/or visible light processing functions were therefore developed and incorporated into this module (filters, lookup tables, isodensity curves, etc.).
- A superimposition module combining the visible light and gamma images of the same field of view in the same window. As both images are acquired under the same geometric conditions, the topographical superimposition is perfect and the true direction of the radioactive sources is clearly indicated on the screen.

3. Calibration (B.5.)

Calibration operations were carried out for two purposes:

- to assess the sensitivity and resolution performance of all the usable configurations;
- to evaluate the absorbed dose rates of quasi-point sources corresponding to camera input levels.

3.1. Sensitivity and Resolution Tests

Quasi-point ^{60}Co sources were tested in various configurations with the Philips 20xx image intensifier tube. Two collimators with apertures of 2 × 19° and 2 × 26° were used at their long and short focal length limits, with CsI(Tl) scintillators between 2 and 10 mm thick. The experimental results, in terms of image definition for various focal lengths and apertures, are indicated in Figure 3.

The sensitivity of the current configurations corresponds to an absorbed dose rate threshold of about 0.01 $mGy{\cdot}h^{-1}$ for quasi-point sources.

3.2. Linearity Tests

Tests were conducted using quasi-point ^{137}Cs and ^{60}Co sources with the following configurations: 20xx image intensifier tube, both types of collimators, two CsI(Tl) scintillators (2 mm and 4 mm thick) with two gain settings for the second intensifier. Figure 4 shows typical experimental results recorded with the ^{60}Co source.

4. Prototype Application at a Decommissioning Site (B.6.)

Cartographic tests were conducted in the active cells of the decommissioning program in the RM2 radiometallurgy facility at Fontenay-aux-Roses.

The gamma camera was inserted vertically into cells 2 and 3 through the existing overhead ports. Because of the relatively high ambient dose rates (5 to 30 $mGy{\cdot}h^{-1}$) the configuration with the lowest sensitivity and the highest resolution was selected, i.e. the small-aperture (2 × 19°) collimator with the thin (2 mm) CsI(Tl) scintillator.

The gamma camera performed as expected under conditions fully representative of actual operation. The results were fully satisfactory for the localization of radioactive sources and for the detection sensitivity (measurement times of 10 to 150 seconds provided utilizable images).

The photo in Figure 5 shows a shielding defect on a ventilation filter in cell 2; the photo in Figure 6 is an overhead view of the waste containers inside cell 3.

The zones with the highest irradiation levels are displayed in red on the false-color image, corresponding to dose rates of up to 6 $mGy{\cdot}h^{-1}$ on the images included in this report.

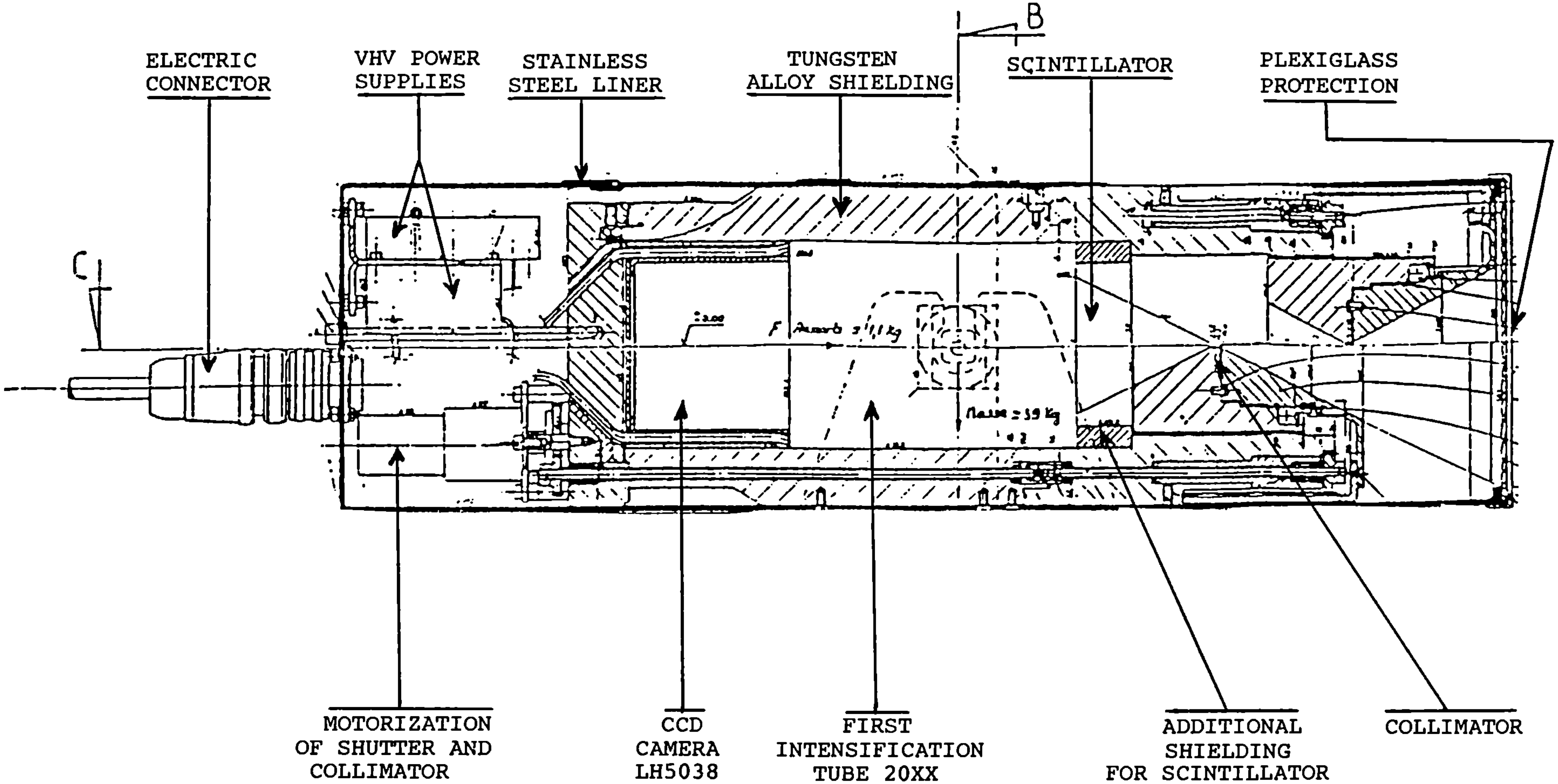


Figure 1 : Longitudinal section of detection set

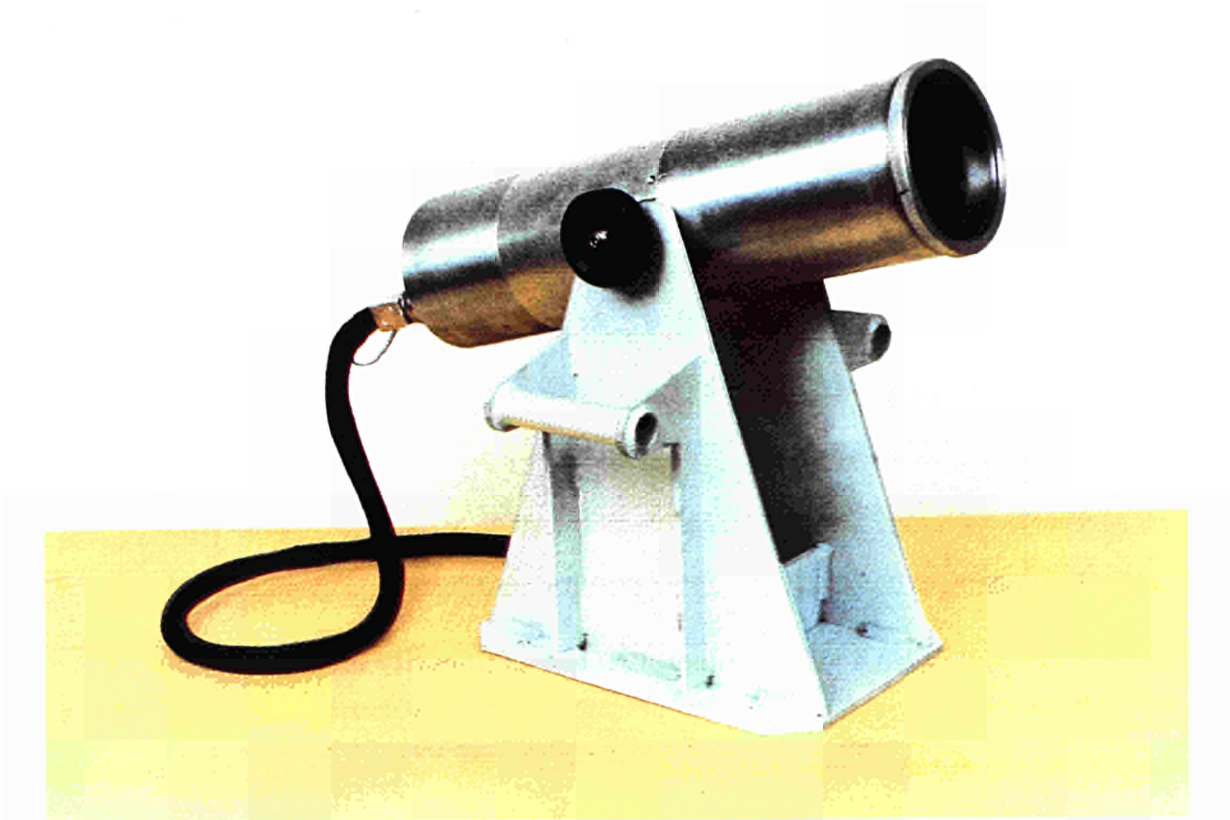


Figure 2 : Photograph of detection set

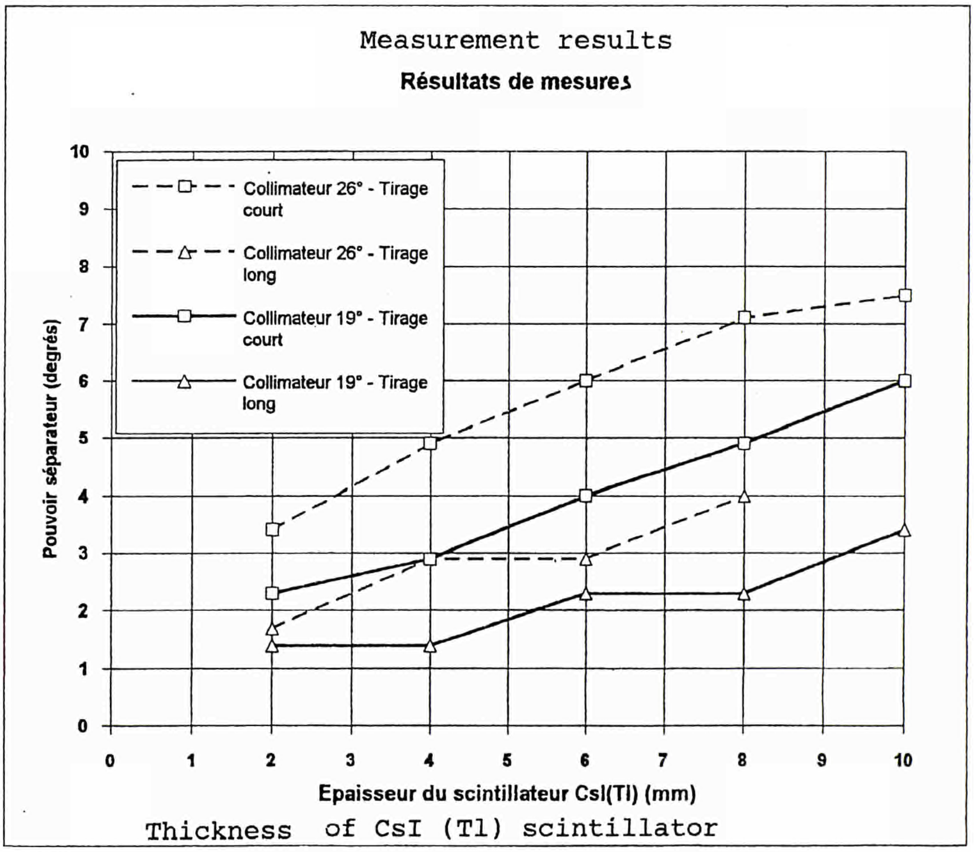


Figure 3 : Measurement results of angular resolution

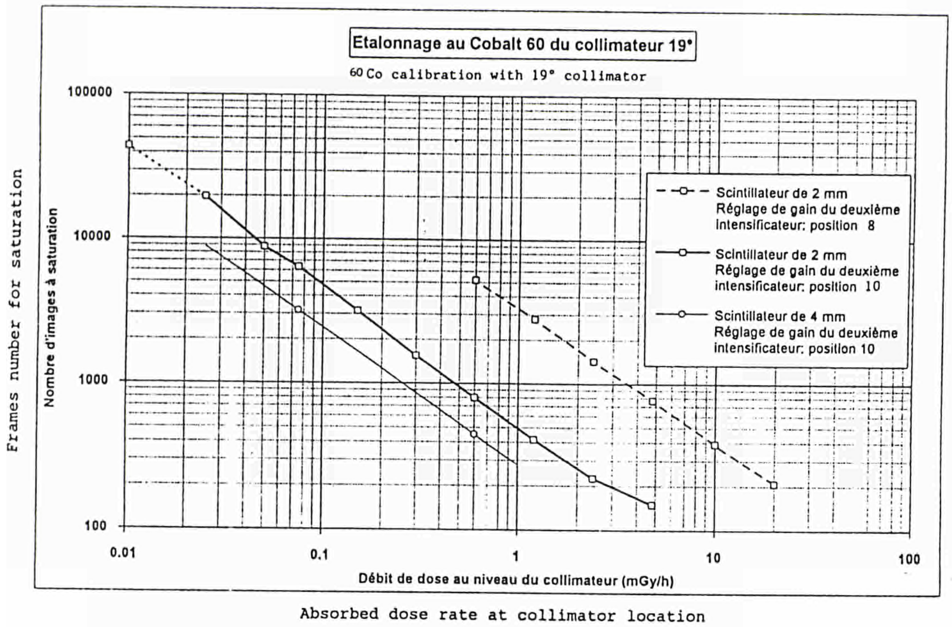


Figure 4 : Experimental results of linearity tests with 60Co

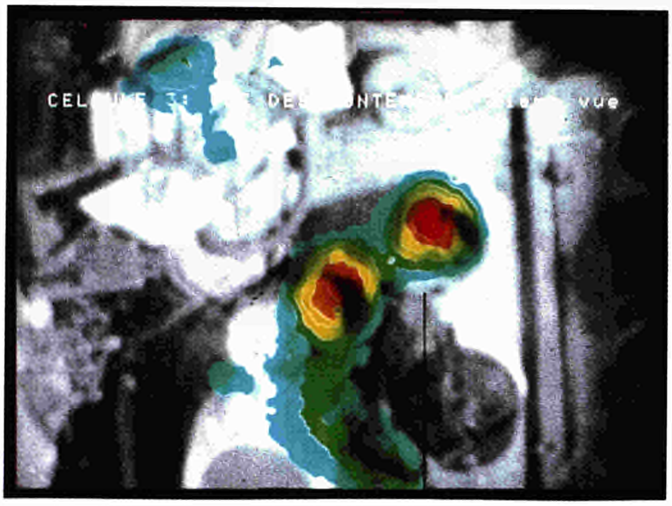


Figure 5 : Example of picture in RM2 hot cell n° 2

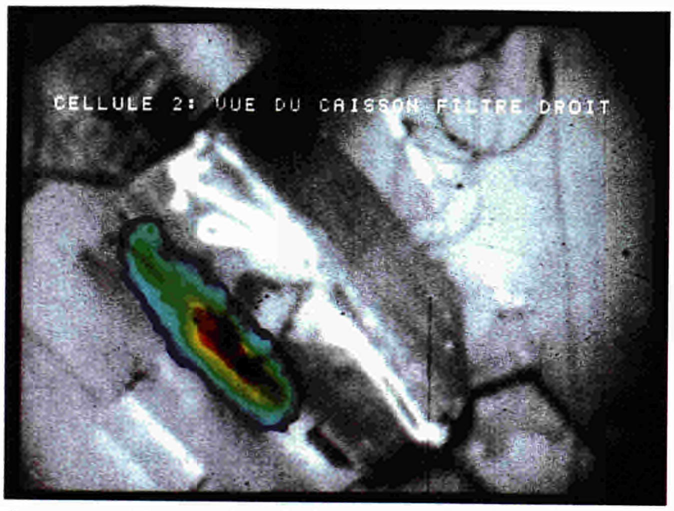


Figure 6 : Example of picture in RM2 hot cell n° 3

SECTION B: IDENTIFICATION OF GUIDING PRINCIPLES

Section B of the programme is concerned with the identification of guiding principles relating to:

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

7.1. POLICIES, REGULATIONS AND RECOMMENDATIONS FOR DECOMMISSIONING

A study has been performed by the Commission together with a group of experts, with the objective of assembling and discussing policies, regulations and recommendations for decommissioning, and of recommending Community actions in this field.

The study is structured as follows:

1. Introduction
2. General principles and international recommendations relevant for decommissioning
3. European Community requirements and recommendations
4. Policies, regulations and recommendations in Member States
5. Present decommissioning practice
6. Conclusions and recommendations of the Working Group

Annex 1 - National policies, regulations and recommendations
Annex 2 - Selected decommissioning cases

The resulting document, "*Policies, regulations and recommendations for the decommissioning of nuclear installations in the European Community*" was submitted to the relevant internal committees for review and will shortly be published by the Office for Official Publications of the EU (OPOCE) - Luxembourg.

7.2. PREPARATION OF A DECOMMISSIONING HANDBOOK

Contractor:	AEA-Wind., CEA/IPSN, ENEL, M. Lasch, F.W. Bach, CEA/UDIN, GNS, ONDRAF/NIRAS, FRAMATOME
Contract No.:	FI2D-0073 to FI2D-0081
Work Period:	April 1992 - December 1993

A. OBJECTIVE AND SCOPE

A handbook of the technology for decommissioning of nuclear installations will be prepared. The main subject of the handbook will be the detailed description of the state-of-the art techniques. For each of these techniques, the following information should be provided (as applicable and available):

- range of application conditions for which the technique is considered first choice,
- performance characteristics,
- by-product characteristics,
- specific radioprotection and safety aspects,
- cost data, including employment of labour,
- existing specific equipment,
- necessary auxiliary equipment,
- an example of past practical application,
- any other relevant information.

Techniques that are not state-of-the art, e.g. obsolete techniques or techniques which are being developed but not yet proven, should only be mentioned and qualified briefly.

The handbook should indiscriminately include available relevant information of any origin.

B. WORK PROGRAMME

B.1. Editorial assistance (AEA-Wind.)
B.2. Characterisation of radioactivity (CEA/IPSN)
B.3. Surface decontamination (ENEL, M. Lasch)
B.4. Dismantling and segmenting (F.W. Bach, CEA/UDIN)
B.5. Management of materials from dismantling (GNS, AEA-Wind.)
B.6. Radiation protection and safety techniques (AEA-Wind., ONDRAF/NIRAS)
B.7. Installation design and operating features (FRAMATOME)

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

One of the major aspects of producing the Handbook is the exchange of information between authors/Member States. To facilitate this, a meeting was held in Brussels on 8 December 1992 for all authors.

Progress and results

1. Editorial assistance (AEA Windscale)

Editorial assistance is divided into three main areas: the collation of a table of contents; the review of draft contributions to assure that the English is clear and correct and the type setting of the Handbook to assure conformity with the editorial requirements.

All sufficient contributions from authors have now been received to finish the editorial procedure: although significant changes to the text are still possible.

2. Characterisation of radioactivity (CEA/IPSN)
The complete chapter has been finalized.

3. Surface decontamination (ENEL, M Lasch)
This chapter of the Handbook is concerned with decontamination. ENEL are providing sections on the general approach to decontamination, decontamination for segmented parts and decontamination for building surfaces. M Lasch is providing sections on basic decontamination methods and decontamination for large volume closed systems.
Draft contributions have been received from both ENEL and M Lasch. The final version of both contributions are expected shortly.

4. Dismantling and segmentation (F W Bach, CEA/UDIN)
Draft contributions of all sections have been received and will be finalized shortly.

5. Management of materials from dismantling (GNS, AEA)
In this chapter, GNS has provided sections on conditioning techniques, conditioning equipment (for liquid and solid materials) and on recycling techniques. AEA have provided sections on materials arising during decommissioning, special conditioning techniques and waste transport and disposal packages. Completion will be achieved soon.

6. Radiation protection and safety techniques (AEA, ONDRAF/NIRAS)
In this chapter, AEA has provided sections on radiation dose control measures, remote operations and filtration techniques. ONDRAF are providing the sections on manual operations, shielding and containment techniques and safe storage. The finalized version will soon be completed.

7. Installation design and operating features (FRAMATOME)
The complete chapter has been finalized. It includes sections on the evaluation of activity and radwaste volumes, design features to facilitate decommissioning, replacement of damaged components, improvements of plant layout design, and on documentation and updating.

SECTION C: TESTING OF NEW TECHNIQUES IN PRACTICE

A. Objective

The projects and studies in this section aim at testing, demonstrating and assessing new decommissioning techniques under real conditions of radioactivity configuration, size, accessibility and the state of the plant. The four large Pilot Dismantling Projects (WAGR/Windscale, BR3/Mol, KRB-A/Gundremmingen and AT1/La Hague) are the focal point of this section. Large-scale active testing of new techniques is also performed in a number of other dismantling projects ("alternative tests").

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 two previous five-year decommissioning programmes. These investigations concerned the dismantling of five reactors, three fuel fabrication facilities and one high-level waste vitrification facility.

C. Programme 1989 to 1993

Section C includes:
- the execution of four pilot dismantling projects
- alternative large-scale tests to be performed in nuclear installations other than the pilot dismantling projects:
- secondment of scientific staff from Member States to the pilot dismantling projects: The operators of pilot projects receive staff from organizations in the other Member States for active cooperation within the framework of the project.
- Development of a data base for costs, operational doses, working times and waste arisings.
- Development of a data base for the performance of cutting/segmenting techniques.
- Studies to identify further R&D requirements.

In order to permit a continuity of important actions until the next Framework Programme becomes operational, a number of contracts could have been extended to cover 1994.

D. Programme implementation

In the pilot dismantling projects, important milestones have been reached: at AT1, after the dismantling and removal of an equipment, the final phase of site cleanup has been reached, whereas in BR3 work is now concentrating on the dismantling of the RPV internals. At KRB-A and WAGR, the commissioning of equipment for remote disassembly of RPV internals is under way. The original intention of decontaminating and dismantling the WAGR heat exchangers in situ was abandoned and their removal and disposal in one piece is being investigated. The collection of specific data for the two data bases was pursued with particular priority for data obtained via real dismantling work. Two major seminars on Practical Decommissioning Experience (SCK/CEN Mol, 6-7 May) and on materials recycling (Krefeld, 26-29 October) were organized.

Thirty-two research contracts were concluded, of which five were completed.

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 - June 1994
Coordinator: C STUBBS, AEA
Phone: 44/9467/72413 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 started in 1994 and is planned to be completed by the end of the century.

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, through shared-cost participation in specific parts of the project, is promoting the use of advanced techniques and the performance of collateral investigations. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is an important objective of this project.

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.

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. This phase was completed in 1993. The work programme for the second phase, eg point B.6., has to be redefined after an AEA internal review.

B. WORK PROGRAMME

B.1. Dismantling of the top biological shield (TBS), a 60 t disc-shaped steel and concrete structure, by thermic lancing.

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 dismantling of active components.

B.6. Dismantling of a WAGR heat exchanger by withdrawing the entire unit vertically through the containment dome. (This point is currently under negotiation).

C.Progress of work and results obtained

Summary of main issues

The two areas of activity during the year have been inactive trials of the remote dismantling machine and the re-evaluation of strategy for removal of the four WAGR heat exchangers. The non active testing of the remote dismantling machine is complete and an analysis of the experiences gained is being prepared. The testing was a combined training period which in practice was dominated by the need to ensure correct operation of the complex remote dismantling machine.

At the beginning of the year a change in conditions for acceptance of Low Level Waste at the U.K. land burial site at Drigg resulted in a strategy review for dismantling heat exchanger A. It is clear that a removal and disposal programme which embraces all four heat exchangers is now more attractive than the decontaminate and piece part dismantling methodology which has been previously developed.

Progress and results

1. Inactive trails of the remote dismantling machine (B.4.)

Plant and Equipment

The WAGR decommissioning programme included a period of inactive trials of the key dismantling equipment. The aims of the inactive trials were to prove the equipment operation and to train operators to use the equipment in a non-critical situation.

Inactive trials were facilitated by setting-up the Dismantling Module in a Machine Test Facility (MTF), constructed within the containment building, adjacent to the reactor. The Dismantling Module comprises the deployment mast, the manipulator platform which runs on the mast, the manipulator and stereo camera boom mounted on the platform and the reeling drums through which the services are fed to the manipulator and its associated tooling. The module was designed to be transferred over into the Remote Dismantling Machine (RDM), in a single operation at the end of the inactive trial period.

Organisation

A Training Manager with a technical background was appointed to oversee the inactive trials. He began by learning the operation of the manipulator in the HERO development facility where it had been set up in isolation before the construction of the MTF.

The Training Manager then structured a programme of trials and training using a progressive system, so that simple tasks could be carried out with low risk to the equipment, before moving on to more difficult tasks where operator error could have led to equipment damage. A team of three industrial staff of process worker grade were selected to learn how to operate the equipment and perform the trials.

Programme and Delays

Training was carried out over a planned 24 week programme which largely kept on schedule, although it was subject to knock on delays. Delays arose from problems installing and functionally testing the Dismantling Module equipment in the MTF for the various inactive trial stages.

Trials

Trials were carried out in two broad progressive phases. Initially all operations were performed from a control station local to the MTF so that when the operators first used the equipment it was in their direct line of sight. For the second phase control was switched to the main control room in an adjacent building so that the operation became fully remote and an accurate simulation of how the active dismantling will be carried out.

In the first stage of trials the operators were taught how to raise and lower the mast platform, to safely start-up and shutdown the manipulator, the importance of payload limitations and simple pick/place and stacking exercises. These trials were all carried out from the local control station with the equipment in direct line of sight. Having achieved competence using the equipment in this regime the equipment was handed back to the main contractor to transfer control to the main control station.

The second stage of trials were fully remote requiring the operators to be introduced to the mono and stereo viewing systems. A dual operator approach was adopted, with one operator driving the boom to position the stereo camera and the other driving the manipulator.

Operators repeated the first stage trials but without the benefit of a direct line of sight to the equipment. They then learnt the techniques which will be required to dismantle the first major reactor component, the Hot Box. Dismantling the Hot Box will involve deploying a gas cutting torch on a non-linear path through an array of vertical refuelling tube stubs. The degree of difficulty was gradually increased by using a polystyrene mock-up on initial trials, moving on to performing practice flame cuts on plain steel, through to cutting up a steel mock-up built to the same specification as the Hot Box. This last exercise was the culmination of the trials incorporating elements of all that had been learnt previously.

Results

The trials highlighted a number of areas where the equipment could be improved. These were in the areas of operator information to improve useability, refinements to the gas cutting system which would improve reliability and modifications to the stereo camera system which would ease deployment of the system.

2. Dismantling of a WAGR Heat Exchanger (B.6.)

The WAGR system includes heat exchangers for the production of high pressure steam using the reactor cooling gas. There are four Heat Exchangers references A, B, C and D each being a steel pressure vessel 3.4M diameter and 20.6M in length and weighing approximately 175te. Heat Exchangers A and C are sited in their original positions within their respective concrete bioshields; each has undergone some early preparation work with respect to removing external pipework. Heat Exchangers B and D have been fully prepared for removal; all external pipework has been removed and the open ends sealed. B and D have been raised in their bioshields and the voids which were created below have been used to build the waste route facilities.

In previous reports, work associated with the dismantling of Heat Exchanger A was described. Three task areas were under consideration: decontamination, operational methodology and the facilities required to carry out the dismantling. A possibility at the beginning of this year, through a change in acceptance criteria to the Drigg disposal site was that it may be possible to dispose of the Heat Exchangers as complete units. The need to dismantle was removed with the possible benefits of reduced operator dose rate and lower capital costs. It is the evaluation of the option to dispose of the Heat Exchangers as complete units which has been undertaken throughout 1993.

A working group established that there were no fundamental problems on technical, safety or operational grounds which would prevent the option proceeding. Subsequently a programme of work was undertaken.

To assist in the evaluation, work was required in two areas for inclusion in the specification of the operational work, these areas were dose assessment and radioactive inventory. Both areas relied upon information provided by a comprehensive film stringer survey of the Heat Exchangers radiation fields. The dose assessment established an estimated total dose for the disposal of all four Heat Exchangers of 384 mSv. The radiation fields were converted using a RANKERN code and the following specific activities $\beta\gamma$ calculated.

Heat Exchanger A: 1.036 GBq/te
B: 1.836 GBq/te
C: 1.517 GBq/te
D: 3.995 GBq/te

The evaluation falls into three main areas: contact strategy, operations and approvals. The WAGR decommissioning project management team have been given revised terms of reference and contract strategy now requires that wherever possible work packages must be placed on a competitively tendered basis. It was decided that all operational work packages would be tendered as would any provisioning and some management support services. The major operations have been arranged into three specifications; an outline of the scope is given below:

Preparation

(a) Prepare the Heat Exchangers. This mainly consists of removing all appendages external to the Heat Exchanger Shell and sealing all openings. It also covers the removal of all fittings between the Heat Exchanger shells and the bioshield walls. The final activity is the fixing of contamination on the Heat Exchanger Shells and the internal bioshield walls.

(b) Preparation of internal plant and fittings within the containment building. This mainly consists of removing cabling and other fittings.

Removal

(a) Lift Attachments Design and Manufacture
(b) Building B50 Penetrations
(c) Containment System Design and Construction
(d) Transport Saddles Design and Manufacture
(e) Lifting
(f) Site preparation
(g) Transport

Disposal

(a) The off-loading and siting of the Heat Exchangers at Drigg site.
(b) The internal grouting of the Heat Exchangers to Drigg site requirements.
(c) The encapsulation of the Heat Exchangers in their final position.

Tenders have been invited for each of the three specifications and detailed commercial and technical assessments are currently being carried out.

8.2. COMPARATIVE ASSESSMENT OF ALTERNATIVE UNDERWATER REMOTE OPERATION AND SEGMENTING TECHNIQUES FOR REACTOR VESSEL INTERNALS OF KRB-A

Contractors: KRB
Contract No.: FI2D-0002
Work Period: October 1989 - September 1990
Coordinator: H STEINER, KRB
Phone: 49/8224/783 730 Fax: 49/8224/782 900

A. OBJECTIVE AND SCOPE

The Boiling Water Reactor plant Gundremmingen A (KRB-A) is one of the four pilot dismantling projects of the EC programme (see also § 8.5.).

The above contract provided a preliminary design and assessment study of alternative remote operation and segmenting techniques for underwater dismantling of the pressure vessel internals of KRB-A. Occupational radiation exposure, costs, and the conditioning and minimisation of the radioactive waste were considered in particular.

In the course of its implementation this study was extended to the case of the VAK BWR of Kahl.

B. WORK PROGRAMME

B.1. Inventory of KRB-A conditions, e.g. materials and geometries of components, local dose rates and radioactivities, accessibility.
B.2. Literature study on the state-of-the-art.
B.3. Analysis of underwater segmenting techniques including thermal, mechanical, electrical and chemical techniques.
B.4. Investigation of remote-operation techniques considering alternative manipulator designs and various degrees of automatisation.
B.5. Comparative evaluation of the alternative techniques investigated, considering all relevant aspects; selection of the optimum technique(s); identification of experimental investigations needed, if any.

C. PROGRESS OF WORK AND RESULTS OBTAINED

This study was completed in 1990. Publication of its final report is in course.

8.3. PILOT DISMANTLING OF THE BR3 PWR. DECONTAMINATION OF A PRIMARY CIRCUIT, REALISATION OF CUTTING EQUIPMENT, SEGMENTATION OF ALL REACTOR INTERNALS

Contractors: SCK/CEN, Siemens-KWU, Framatome
Contract No.: FI2D-0003
Work Period: October 1989 - July 1994
Coordinator: V MASSAUT, SCK/CEN
Phone: 32/14/33 21 11 Fax: 32/14/31 19 93

A. OBJECTIVE AND SCOPE

The BR3 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 BR3 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) (see 8.5 and 8.10).

As a Pressurised Water Reactor, the BR3 is representative of the reactor type most frequently used in the Community. 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.

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. Disassembling the main reactors internals
B.4.1. Lower core support assembly
B.4.2. Upper core support assembly
B.4.3. Reactor vessel collar and instrumentation basket

B.5. Segmenting of stainless steel reactor internals subassemblies
B.5.1. Annular cylindrical geometries (3D geometries)
B.5.2. Plates of grids (2D geometries)

B.6. Segmenting of a thick carbon steel ring (reactor vessel collar).

Progress of work and results obtained

Summary of main issues

1. Phase 1 work (B.1., B.2.)

The decontamination and segmentation work have been completely achieved in 1991 and were previously reported. Only the points B.1.3. (treatment and removal of decontamination waste) and B.2.5. (waste treatment and packaging) were completed during 1993.

2. Phase 2 work (B.4. to B.6.)

The main objectives of the 1993 activities were to complete the design and fabrication of the main dismantling equipments needed for dismantling all the reactor internals (except for the thermal shield already dismantled in phase 1) presenting very complex geometries and shapes. The cold testing on actual scale mock-ups of all these equipments was also one of the main objectives of this year : it included the cold testing of the supporting and positioning equipment, the so-called turn-table, of all the cutting equipments (circular saw machine for horizontal cutting, alternative saw, hydraulic jaw cutter, Metal Disintegration Machining, Electro Discharge Machining), and the handling devices.
The band sawing technique had to be tested on a prototype machine to assess the technical feasibility of such a technique to be used under water.

Progress and results

1. Phase 1 work (B.1., B.2.)

The remaining waste from the decontamination operation was finally evacuated. This long delay for evacuation was mainly due to administrative and regulatory problems.
The removal of the plasma arc torch prefiltration strainer led to unforeseen problems. First, following the cold tests carried out, the use of grinding or even saw-like grinding (as reported previously) was not accepted due to the spreading of the sedimented dross (collected in the strainer) by the high rotating speed of the disc. Finally, tests with a reciprocating saw were carried out successfully, to cut the strainer (O.D. : 200 mm, wall thickness : 2 mm, height : 1200 mm, containing highly radioactive dross) into two smaller parts able to be easily transported in a shielded container. The actual cutting was performed with success but then the second problem arose : the strainer did not contain more than about 10% of the foreseen amount of dross. The dross were indeed not collected in the strainer but, due to the vortex water stream in the plasma cutting chamber, they found a way to fall between the strainer and the chamber wall (see Figure 1).
Therefore, a special tool was designed, fabricated and cold tested for removing the mud (mixture dross-water) accumulated at the bottom of the chamber. This tool was developed to minimize, as far as possible, the secondary waste volume to be evacuated. The removal of the dross was performed successfully, and the plasma chamber was cleaned. The dross collection tank and an additional filter set (used for pumping the last remaining dross) were afterwards transported to the deactivation pool for intermediate storage.

2. Phase 2 work (B.4. to B.6.)

Following the scenario and the strategy set up in 1992, the different equipments were designed, fabricated and cold tested.
First a prototype of bandsawing machine, delivered late in 1992, has been tested under water. These tests confirmed the feasibility and the reliability of such technique for cutting metallic pieces (of different shapes) remotely under water. Moreover, the main cutting parameters for cutting stainless steel of different thicknesses (ranging from 1.5 mm to 150 mm) as well as thick carbon steel and heterogeneous material (carbon steel clad with stainless steel, representative of pressure vessel material, see Figure 2) were established.
These parameters included the band speed, the cutting feed, the band tensioning force, the secondary waste removal and characteristics, the cutting time, the saw blade type and lifetime, and so on. The summary of the obtained results is given in Table II.

The results of these tests led to set up the specifications for the definitive band saw machine to be used for cutting vertically all the reactor internals and the reactor vessel collar (thick carbon steel ring clad with stainless steel, inserted between the RPV upper flange and the reactor head). This machine will also be able to cut vertically the reactor pressure vessel. This band saw machine has been fabricated during the last trimester of 1993 (see Figure 3).
The turn-table, central equipment for supporting, clamping and positioning the workpieces during dismantling, has been delivered, assembled and then tested under water with the different dismantling equipments and reduced full scale mock-ups.
The circular saw machine as well as the MDM (Metal Disintegration Machining) has been cold tested on full scale mock-ups in order to optimize the cutting parameters. The main cutting parameters for the circular sawing of the different pieces are summarized in Table II.
The Metal Disintegration Machining tests were carried out using the following parameters :
- oscillating electrode
- feed = 0.5 mm/min for cutting pipes Ø 44/40 mm
  0.5 - 2.5 mm/min for cutting the core baffle
- electrode current : ± 300 A D.C.
- electrode-workpiece voltage : ± 4 V D.C.

Nevertheless some problems, mostly related to the electrode design, have still to be solved. The electrode design has been reviewed and optimized. Additional cold tests are foreseen early in 1994.
Finally, after the last cold testing of the EDM cutting equipment for removing not accessible bolts, the desolidarization of the "Rod Shroud Support Plate" using EDM (with hollow electrode) and an impact unbolter has been performed without major problems.
The total time needed for removing 8 bolts by EDM was 27 working days. The total dose intake for performing the operation was 5.347 man-mSv. One can thus summarize the operation as follows :

Operation	Working days		Dose Intake (man-mSv)
	Calendar days	Man-days	
Preparation	3.5	16	0.979
Preliminary operations	1	5	0.449
EDM cutting of 38 bolts	19	80	2.931
Impact unbolting of 16 bolts	1	3	0.265
Plate removal	0.2	3	0.348
Miscellaneous	2	5	0.385
TOTAL	27	112	5.347

The present situation of the different equipment is summarized in Table III.

References

[1] MASSAUT, V, Fourth CEC Seminar on practical decommissioning experience, Mol (Belgium), May 6-7, 1993. Proceedings pp.3-28
[2] MASSAUT, V, III. Stillegungskolloquium Hannover/Greifswald, September 23-24, 1993
[3] MASSAUT, V, Seventh Progress Report 01.01.93-30.06.93 Contract CEC FID-0003-B(TT)av.1

Table I : Results of the cold tests of the vertical band saw

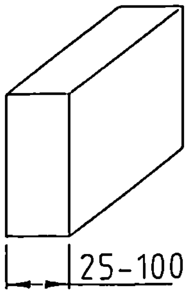
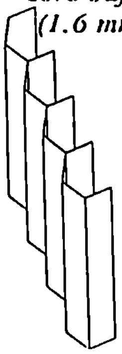
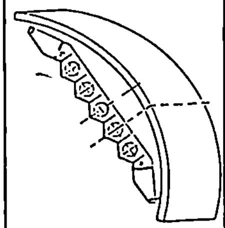
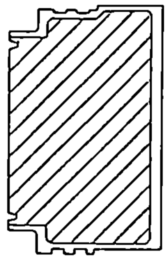
RESULTS OF THE COLD TESTS OF THE VERTICAL BANDSAW				
OPTIMAL PARAMETERS				
	LCSA S.S. (AISI 304) *width 25 - 100*  	LCSA S.S. (AISI304) *core baffle* *(1.6 mm)* 	LCSA S.S.(AISI 304) *core barrel +* *core baffle* 	COLLAR Carbon steel + S.S. cladded *Dim. 310x210* 
Cut type	1	2	3	4
Band saw type **tpi**	Invader 3 - 4	Silencer 10 - 14	Silencer 7 - 11	Invader 2.7 - 4
Saw speed **(m/min)**	32 - 34	32 - 34	32 - 34	32 -34
Feed **(cm²/min)**	8.9	10.13	4.8	5.4
Band tension **(N)**	12000	12000	12000	12000
SIMPLIFIED RESULTS				
Band saw tpi	2.7 - 4	7 - 11		2.7 - 4

Table II : Results of the cold tests of the circular saw

Piece to be cut	Thickness (mm)	Material	Saw blade		Rotational speed (RPM)	Peripheral speed (m/min)	Cutting depth/pass (mm)	Feed (mm/min)
			Diameter (mm)	Pitch (mm)				
Cylindr. vessel or plate	25-38-155	Stainless steel 304	600	19	6-7	11.5 to 13.4	25 to 30	15 to 30
			600	9.6	6-7	11.5 to 13.4	20 to 25	10 to 25
	25		400	19	9-10	11.3 to 12.6	25 to 30	15 to 20
Core baffle	1.6	Stainless steel 18/8/1	600	5	7-8	13.4 to 15.3	80	5 to 6

Table III : Summary of the present situation of the different equipments

Dismantling or handling equipment	Situation	Remarks
Turn-table	Fabrication completed. Step 1 and step 2 cold tested. Step 3 delivered.	The clamping of the plates will be carried out using adapted clamping system from the step 2 (system for horizontal cutting).
Telemanipulator	Long delay before delivery. A lot of software and hardware problems to be solved.	Due to the delay for the manipulator availability, some studies are carried out to do most of the job using long handling tools.
Horizontal cutting by : . circular saw . MDM	Fabrication completed. Equipment cold tested on full scale mock-up. Ready for performance (circular saw).	Some small problems remain in the MDM electrode fabrication and design.
Remote removal of bolts by: . EDM (hollow electrode) . impact unbolting	Performance carried out successfully.	
Pipe cutting using an hydraulic jaw cutter	Cold tests performed using a long handling tool for positioning.	
Pipe penetration cutting using a pneumatic reciprocating saw	Fabrication of special saw holder in progress. Fabrication of reactor collar mock-up in progress.	The same collar mock-up will be used for testing the alternative saw (cutting the pipes penetration) and the band saw machine (for segmenting the collar).
Band saw machine	Prototype tests completed. Full size machine in fabrication.	The full size machine fabrication has been completed at the end of 1993. Cold testing on full scale mock-up will be carried out early 94. The machine is able to cut pieces up to 500 mm wide and 600 mm high.

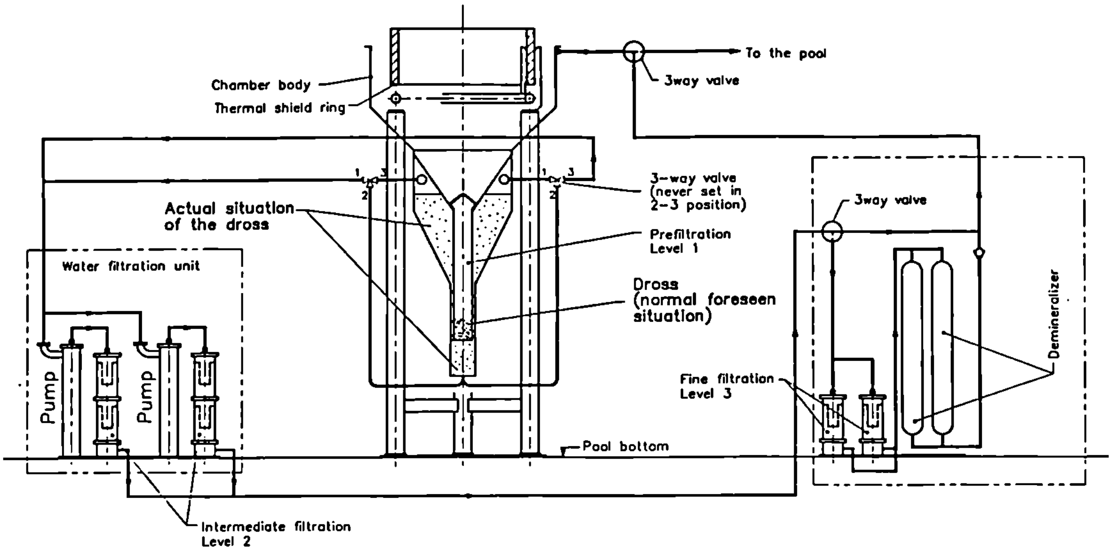


Figure 1 - Flooded chamber for plasma cutting (lower part only)
Actual situation of the dross

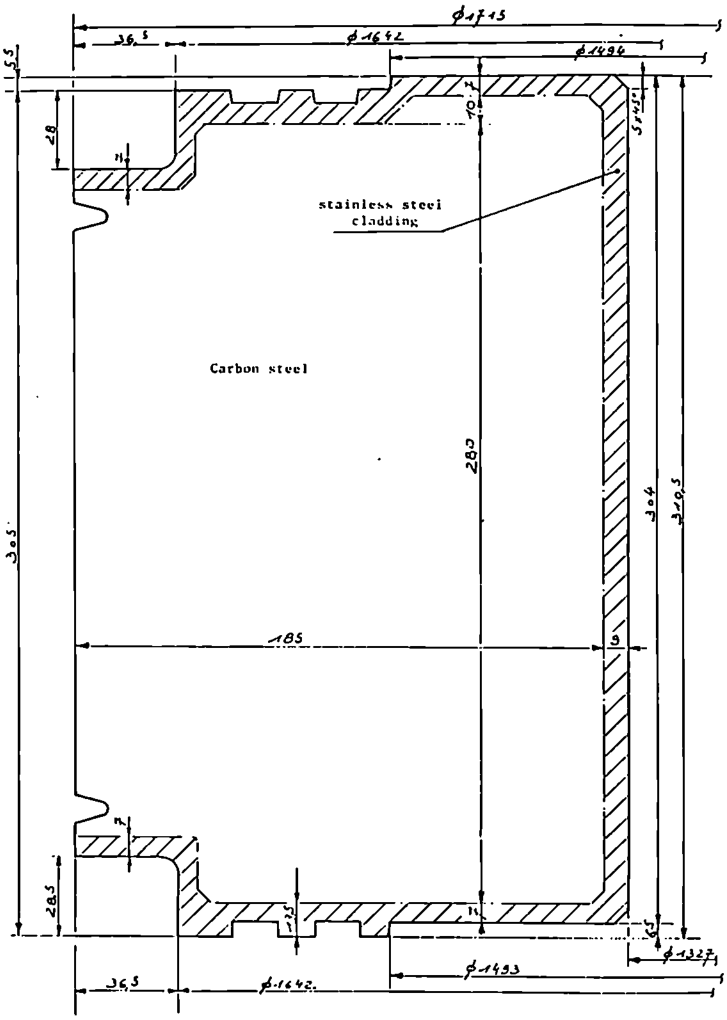


Figure 2 - Cross section of the reactor vessel collar

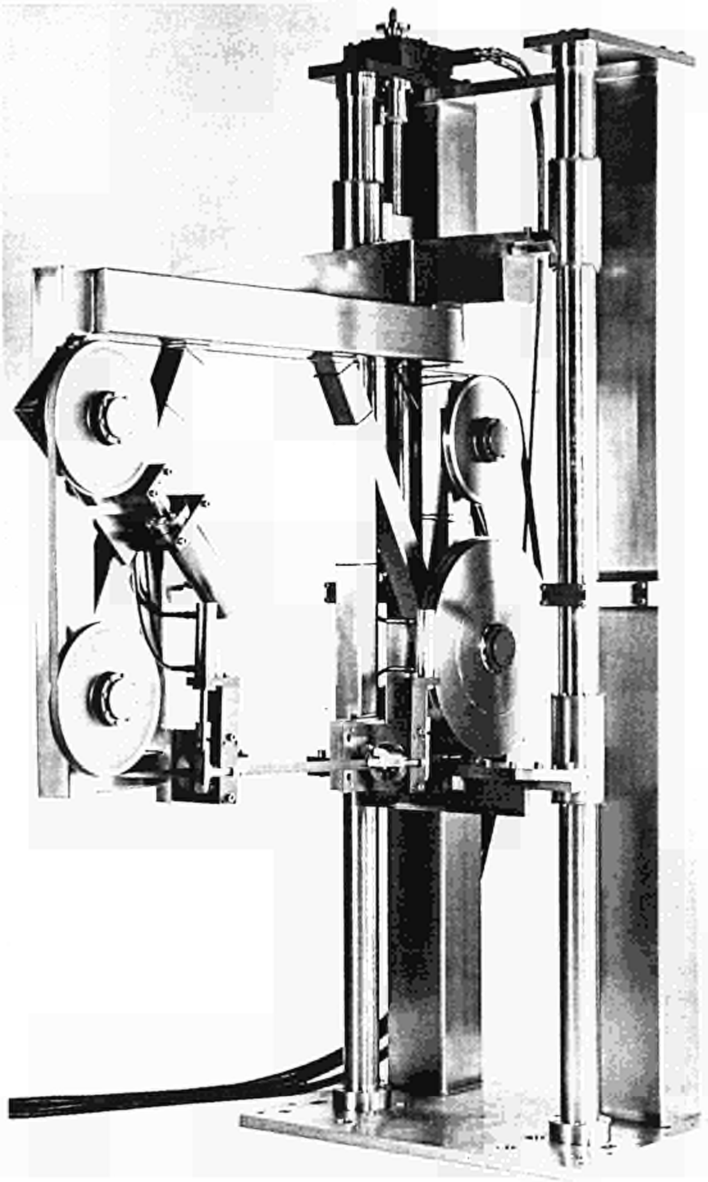


Figure 3 - Underwater band saw machine

8.4. PILOT DISMANTLING OF THE FBR-FUEL REPROCESSING FACILITY AT1. 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 - June 1994
Coordinator: X PETITET, CEA/Ets. COGEMA, La Hague
Phone: 33/33 03 68 49 Fax: 33/33 03 60 14

A. OBJECTIVE AND SCOPE

The pilot facility AT1 for the reprocessing of FBR-fuel had a capacity of 2 kg/day and had been operated from 1969 to 1979.

Considering that the experience to be gained from the dismantling of the first representative nuclear installations in the Community should be made available to all Member States, the Commission selected AT1 as a pilot dismantling project for the 1989-93 R&D programme on the decommissioning of nuclear installations. The Commission, through shared-cost participation in specific parts of the project, intends to promote the use of advanced techniques and the performance of collateral investigations. In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

The dismantling of the AT1 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 operations in high activity cells (B.1.), the removal and conditioning of waste (B.2.) were completed at the end of 1992; the workshop cell which was used for conditioning the waste was emptied in the second part of 1993.

After three years of operation, the running order of the remote equipments (ATENA machine and MA 23) was investigated. An investigation has been realised to try correlating the costs of contact dismantling and decontamination operations with the amount of produced waste (B.4.).

Two semi-automatic carriers have been ordered for the decontamination of walls of high activity cells (903/904/905), which will begin in February 1994.

Progress and results

1. Remote-operated dismantling of equipment out of the strongly contaminated cells (B.1.)
   Dismantling operations in high activity cells were completed in 1992. The clean-up of these high-activity cells was completed in January and February 1993: some pots of dust were removed from these cells.
   The ATENA machine has been placed above its maintenance cell before transport to its storage area.

2. Measurement of radioactivity and conditioning of waste (B.2.)
   The "*workshop cell*" (cell used for conditioning high activity waste) was used in February 1993 to remove pots of dust produced by clean-up operations in cells 903 and 904.
   The amount of waste which was removed from the high activity cells is presented in Table 1.
   In July 1993, the "*workshop cell*" was decontaminated after the dismantling of tools and equipments of this cell. The iron panels which constituted the main parts of walls of workshop cell, were decontaminated and removed for reusing another dismantling site.
   This programme item is completely finished.

3. Dismantling of tanks for the storage of fission products
   - Cell 920 (extension building storage):
   The cell was totally dismantled at the end of 1990, including the floor liner of the cell.
   - Cell 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.
   Task B.3. is therefore completed.

4. Generation of specific data
   The efficiency balance of the teleoperation equipments (ATENA machine and MA23) remote manipulator for the years 1990, 1991 and the first half of 1992 shows: one can record the high reliability of the ATENA carrier around an average of 0.8/0.9; the reliability of the MA23 telemanipulators was around 0.4/0.5 for each arm, during dismantling operations in 1991; however, during 1992, the reliability reached 0.8/0.9 for each arm: working conditions for tape-driven manipulators are better for waste removal than for dismantling operations.
   During 1993, a study was achieved about the costs of contact dismantling and decontamination operations: the main results are summarized in Table 2.

5. Remote dismantling of a reinforced wall (B.5.)
   A concrete wall separated cell 903 and 904.
   To introduce the ATENA machine for dismantling operations in cell 903, the ATENA machine made an opening of 4.5 m high, 1.2 m long and 0.2 m thick in the separating wall.
   For this operation, ATENA machine used a diamond disc-saw cooled with liquid nitrogen.
   This programme item is also terminated.

6. Semi-automatic decontamination and contamination measurements of concrete walls and floors (B.6.)
   - **"Light" carrier or "COMEX" carrier**
   A carrier has been ordered and will be delivered at La Hague in January 1994: this carrier can support either a decontamination tool, or a contamination measurement instrument and must work on vertical surfaces. The movement of tools on the carrier is automatic but the assembly and the move of the carrier are manual and necessitate the work of operators in cells. The carrier consists of several parts and has been designed to be assembled in a very short time.
   Note: at the beginning of the study, another carrier was ordered for horizontal surfaces, mainly for decontamination of recovery iron pan. It was decided to dismantle the floor liner in cells 904 and 905, so the carrier for horizontal surfaces was less interesting and its order has been cancelled.
   - **"Heavy " carrier**
   After clean-up operations in cells 903 and 904, contamination and radiation levels were measured in these cells: the radiation level in cell 903 was too high for operator access to allow assembly of the light carrier, it was therefore decided to order another semi-automatic carrier or "heavy" carrier (BROCK) which can break blocks of concrete and lower dose levels in cell 903. This equipment is driven from outside the high activity cells.
   In spite of important financial difficulties, the decontamination of the three high activity cells has been ordered and the operation will begin as soon as the required equipments is delivered (February 1994).
   The contamination and radiation measurement will be carried out by multi-panel gas-filled "Mosaique" detectors. The decontamination target is to obtain activity levels below 3.4 Bq/cm^2 beta/gamma and below 0.7 Bq/cm^2 on alpha counts.

CELL	904	903
WEIGHT (kilo)	16642	6893
ACTIVITY OF WASTE IN GBq	818 (alpha) 1403 (bêta, gamma)	14 (alpha) 24 (bêta, gamma)

* without recovery iron pans

TABLE 2 - RELATION BETWEEN COSTS OF CONTACT DISMANTLING OPERATIONS AND WEIGHTS OF WASTE

OPERATION	NET WEIGHT OF WASTE kilos	COST: Francs/kilo of produced waste (1989 Values)	MAIN COMPOSITION OF WASTE	OBSERVATIONS
907bis and 911 cells	125350	62	Process equipments (pipes and tanks)	Completed operation (dismantling and cleansing)
901/902/902bis cells	29200	47	Mechanical equipments (fuels pins cutting)	Cost without cleansing or dismantling of iron wall
907 cell	7200	46	Pipes and tanks (Liquid waste)	Partial operation
908 and 909 cells	30000	32	Pipes and tanks (2x15m3 for Fission products storage)	Completed operation
Extension PF	48760	61	Pipes and tanks (2x30m3 light contamination)	Completed operation
952 cell	33090	51	Pipes and tanks under gloves boxes	Completed operation
906 cell and gloves boxes	113850	62	Process under shielded cell and gloves boxes	Completed operation
905 cell	10618	63	Process/ Extraction cycles	Cost without cleansing of concrete walls and floors

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 - September 1995
Coordinator: H STEINER, KRB
Phone: 49/8224/783 730 Fax: 49/8224/782 900

A. OBJECTIVE AND SCOPE

The prototype Boiling Water Reactor Gundremmingen A (KRB-A BWR) of the Kernkraftwerk RWE-Bayernwerk GmbH (KRB) had a capacity of 250 MWe and was operated from 1966 to 1977. Dismantling work has been started for some time (especially the turbine hall has been dismantled), and complete removal of the power station is foreseen to be completed by 2000. The two foregoing EC programmes have been involved by four R&D contracts in the past dismantling work on KRB-A. KRB-A is a European enterprise 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 to promote 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 BR3 and the VAK BWR, respectively (see chapters 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 for the dismantling of all remaining core internals 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.
B.10. Dismantling of feedwater sparger
B.11. Segmenting of the core shroud
B.12. Conceptual planning for the dismantling of the RPV.

Progress of work and results obtained

Summary of main issues
In 1993 the following main activities have been carried out:

- ice-sawing and post-segmenting of a secondary steam generator,
- comparison of cutting techniques at the cladded RPV-cover,
- dismantling of the steam dryer shell,
- implementation of inactive cutting tests for the removal of the biological shield.

Progress and results

1. Dismantling of a secondary steam generator (SSG) (B.1.1.)

After completion the freezing of the secondary steam generator to -15°C end of 1992, the dismantling of this high contaminated component was carried out by ice-sawing. A pre-segmentation of this 10 m high and 64 tons heavy component had to be executed at its installation place because a transportation out of the narrow loop room in one piece was not possible.

The first cut, below the top dome, was executed with an attached band saw without ice in order to test the installed equipment and to check the sawing arrangement. Afterwards several cuts have been performed by sawing through the ice-fixed internals, such as the water cyclones and austenitic U-tube bundle. Due to moving the sawing equipment and the working platforms to the new cutting position, it was possible to realize an ice-sawing cut after two days of preparation. The segmenting of the bottom dome from the tube plate was carried out again without ice. Figure 1 gives the segmenting time for each cut and contact dose rates of the cut-off segments. Figure 2 shows the lifting of a segment during the dismantling of the secondary steam generator and demonstrates impressively the smooth cutting surface. No defrosting effect could be determined at the cutting line.

After defrosting the segments in the turbine hall, the shell parts and the cladded bottom dome, in total 35 tons, have been post-segmented and decontaminated by electropolishing for unrestricted release. The low contaminated water cyclones were manually packed into drums for subsequent high pressure compaction and final storage. The packing of 15 tons single heat exchanger tubes into drums is presently carried out with a mobile and remote controlled hydraulic vehicle, which is equipped with a special hydraulic gripping device.

Until now, for the dismantling of a secondary steam generator an effort of about 10,000 manhours and a collective dose of 110 mSv was recorded.

2. Dismantling of the recirculation pump (B.1.2.)

The segmenting of the casing of a recirculation pump by a band saw is completed. The austenitic component with a mass of 23 tons was cut into 20 segments for pool decontamination (see Figure 3). The first cut to halve the casing lasted about 60 hours without any detectable release of radioactive aerosols to the atmosphere. It could be proved, that the applied decontamination procedures are efficient enough to reduce the initial contamination from 50,000 Bq/cm^2 to less than 0.5 Bg/cm^2 for unrestricted release. Electropolishing and subsequent manual removal of hot spots at this component is still under way.

3. Dismantling of the RPV-cover (B.1.5.)

The post-segmenting and the comparison of different cutting techniques at the RPV-cover is finished. The tests at the activated RPV-cover have been carried out in a cutting cabin. The wall thickness of the ferritic base material is between 73 mm and 112 mm, while the austenitic cladding at the cover inside is 7 mm thick.

Sawing, plasma arc cutting and flame cutting with oxy-acetylene or oxy-propane with and without iron powder additives, have been tested under the aspect of its application to thick-walled and cladded material. The emission of radioactive aerosols and the costs depending on cutting time and secondary waste generation have been investigated.

The comparison of different cutting techniques resulted into the following recommendations: Generally, sawing of thick-walled and cladded material is cost-efficient and is restricted only by geometry and dimensions of the component. Plasma arc cutting with a hand-hold torch applied at the atmosphere came to its technical limit and generated a high aerosol concentration in the cutting cabin. The best cutting technique for the RPV-cover was the flame cutting with oxy-acetylene and without powder additives. This technique was working well and cost-efficient if the cut was started from the ferritic outside; but continuous operation of a suck-off filtering device and wearing of protection masks is necessary.

4. Segmenting of the steam dryer (B.2.1.)

The segmenting of the steam dryer shell with a underwater plasma torch was successfully completed. About 160 single segments have been cut off the steam dryer shell. They have been decontaminated and prepared for the external nuclear recycling by melting. The overall cutting length was about 200 m. Figure 4 shows the situation at the steam dryer after partial removal of the shell. During the dismantling of the 5 mm thick shell by plasma torch cutting, the conditions of visibility in the water did not get worse, also the specific activity of the water did not increase substantially. Also no increase of the dose rate and the aerosol activity at the working place on the reactor floor could be recorded.

About 1000 manhours and a dose of 7 mSv were necessary to remove the steam dryer shell by remote-controlled underwater plasma cutting. Additional 700 manhours and 5 mSv have been spent for pre-decontamination prior to external melting.

The planning of the further dismantling of the steam dryer is under way.

5. Qualification and large-scale testing of a wire saw device for the dismantling of reinforced concrete walls (B.7.)

In order to select a procedure for segmenting the biological shield at KRB A, a literature study on principally suited concrete-segmenting techniques was executed. Different techniques were investigated and valued for an active implementation. As result of the study, the diamond wire sawing and the chain sawing were the favoured concrete cutting techniques which should be tested in detail at an inactive mock-up of the biological shield.

For the large-scale testing, an inactive mock-up of the cylindric biological shield was constructed outside the controlled area (see Figure 5). In this model all main construction details which are important for the dismantling procedure, e.g. dimensions, reinforcement, vertical cooling tubes, a steel liner at the inside of the structure and the surrounding reactor building structure have been realized. The wall thickness of the model is 1.3 m, whereas the height is 4 m.

Preparatory work was necessary to install all auxiliary equipment like: water suppling and water collecting system, working platforms, etc.

The first job was to remove the inner steel liner in 0.6 m wide sheets. The vertical cuts at this 3 mm structural steel was executed remote-controlled by a hydraulic driven circular saw running on a vertical guide rail. The diamond blade of the saw was cooled by water. Only the short horizontal cuts had to be performed manually using a conventional grinding machine.

It was decided to test two different types of diamond sawing machines and one chain sawing machine.

Before installing the diamond sawing machine at the top of the model, it was necessary to perform a horizontal and vertical core drilling to insert and close the diamond cable to a whole loop for operation.

The removal of two concrete blocks by cable sawing could be preformed without substantial problem.

The last concrete block was segmented by a chain saw which was vertically guided at a special rail. This cut was performed in two steps by using a sword of 0.6 m and 1.6 m finally. During this test some problems occurred after a break of one chain segment.

Altogether three concrete blocks, 20 tons each, have been cut off and lifted up from the remaining model of the biological shield.

In addition to the time recording, the consumption of wear material and the arise of secondary waste for both techniques have been regarded and evaluated. The main results are given in Table I. The tests indicated, that the diamond wire sawing technique has a much higher reliability and should therefore be preferred for the implementation at the real biological shield.

Table I: Results obtained by testing concrete-segmenting techniques at the model of the biological shield

Removal of the steel liner [1]			
time for assembling and disassembling	[h]	0.8	
time for segmenting	[h]	1.8	
Core drilling [2]		**vertical**	**horizontal**
depth of boring	[m]	4.0	0.5
time for assembling and disassembling	[h]	2.3	1.2
time for drilling	[h]	3.0	0.4
Concrete segmenting [3]		**diamond wire saw**	**chain saw**
time for assembling and disassembling	[h]	5.8	5.2
time for segmenting	[h]	5.1	6.8
kerf width	[mm]	11	16
water consumption [4]	[m³]	~4	~4
wire/chain consumption [4]	[m]	~10	~1
sludge	[m³]	~0.15	~0.2

1) per sheet (4 x 0.6 m)
2) per drilling (d = 125 mm)
3) per cut (5.1 m²)
4) data from the manufacturers

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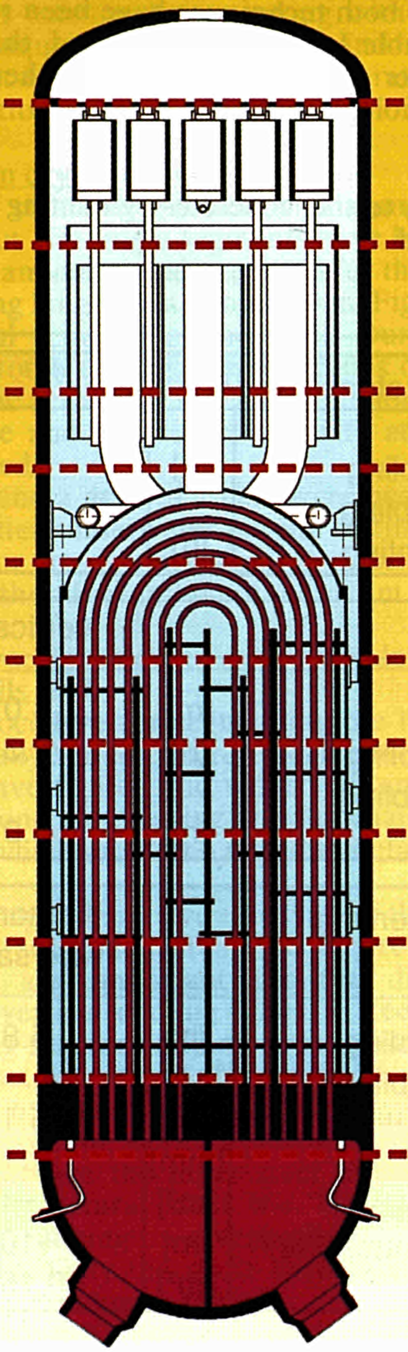
Max. Dose-rate [mSv/h]	Seg-ment
0.1	1
0.1	2
0.1	3
0.1	4
0.1	5
0.2	6
0.4	7
14	8
14	9
20	10
7	11
1	12



Cut	Segmenting time [h]
1*	11
2	6
3	6
4	6
5	8
6	8
7	8
8	8
9	8
10	11
11*	6

* without ice

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Figure 1: Ice-sawing of the secondary steam generator

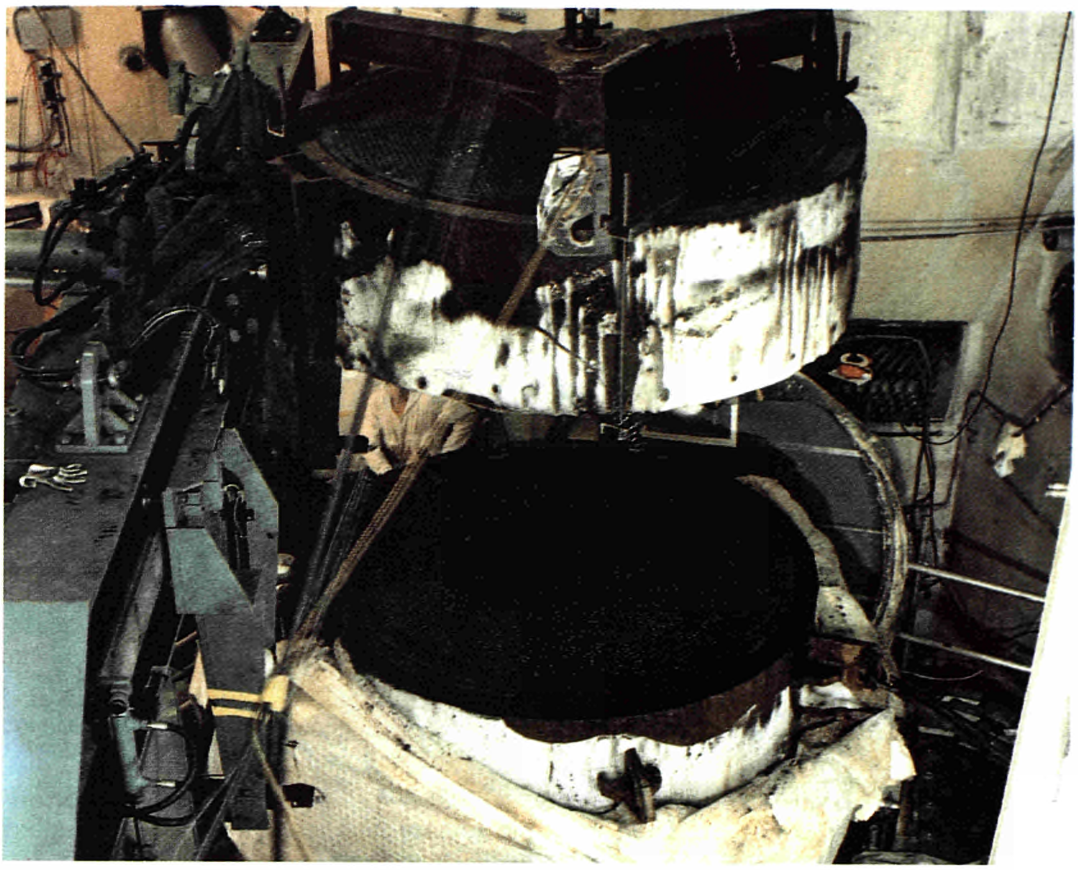

Figure 2: Lifting of an ice-sawed segment of the secondary steam generator

Figure 3: Post-segmenting of the recirculation pump by sawing

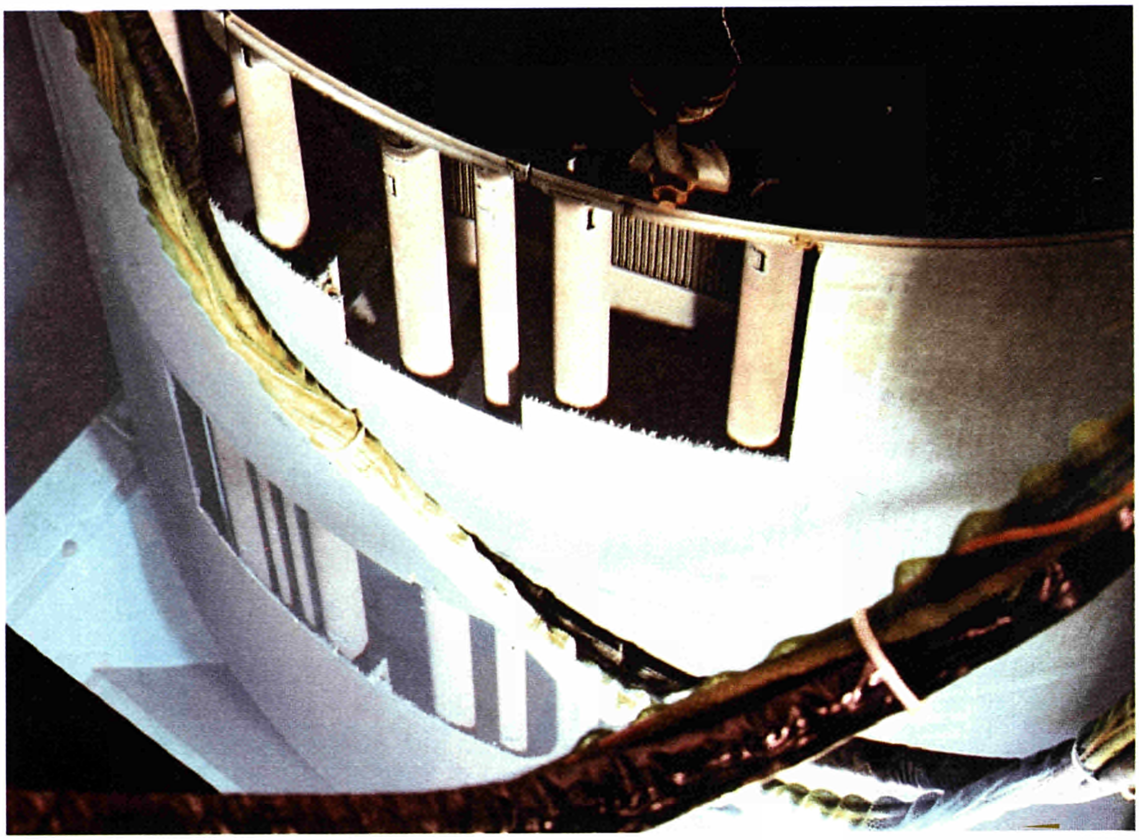

Figure 4: Dismantling of the steam dryer shell by underwater plasma arc cutting

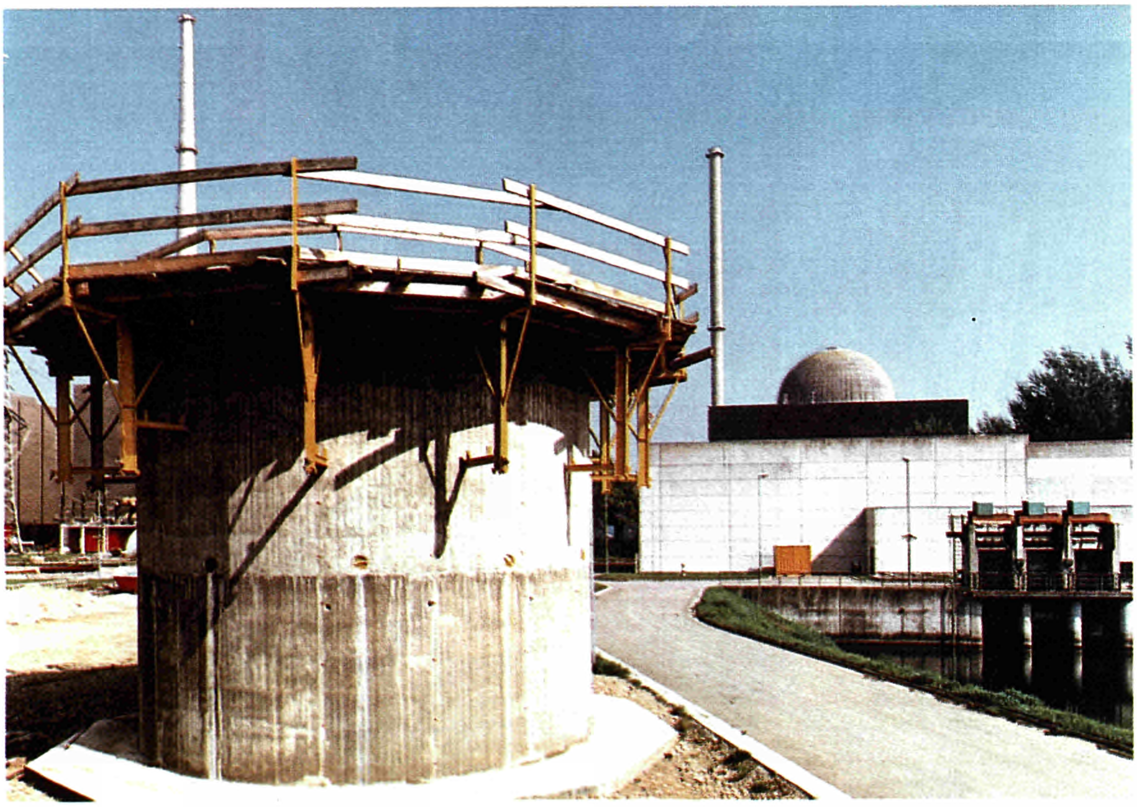

Figure 5: Mock-up of the biological shield

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: 45/42 37 12 12 Fax: 45/42 35 11 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 BNF 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. <u>Progress of work and results obtained</u>

Summary of main issues

The main results for 1993 were as follows:

- obtaining the final state for all the concrete cells, including frogman work in two cells;
- removal of part of the cell ventilation equipment;
- indications that the chimney is clean;
- detailed planning of the remaining work;
- the project will be delayed by some seven months.

Progress and results

1. Removal of fissile material (B.1.)

This item is finished.

2. Removal of large contaminated equipment (B.2.)

This item is finished.

3. Removal of large contaminated facilities (B.3.)

Smear tests taken in the lower part of the chimney showed that this is clean.

4. Decontamination of concrete cells (B.4.)

Remote vacuuming of the last concrete cells was finished. The radiation levels were then measured by exposure of thermo-luminescence (TL) dose meters. Eight to ten TL dose meters were placed in each cell in some representative matrix and were exposed for one hour. The readings are assumed to arise from homogeneously distributed contamination on all surfaces. Smear tests were taken from the table, the floor and the walls of each cell.

Cells No. 5 and 6 were used as prototypes for in-cell high pressure water jetting. The pressure supplied by the pump was up to 200 bar. During the water jetting some new waste particles were found and more clean surfaces were generally observed; only few spots of paint were torn off. The water jetting lasted one hour per cell. After the in-cell cleaning the radiation and contamination levels were remeasured similarly to the procedure applied before cleaning. The above mentioned operations (measurements before/after and the water jetting) required approximately 10 man-hours and gave a total dose of 11.8 man-mSv to four persons. About 40% of the activity in cell 5 and 6, respectively, were removed by the water jetting.

The results from the TL dose meters for all the cells in their final condition are given in Table I. The derived total contamination per cell is given in Table II. In Tables I and II are also shown the dose rate and total contamination in the shutters and shutter housings. The smear tests showed β/γ-activity from ^{60}Co, ^{134}Cs, ^{137}Cs, ^{152}Eu and ^{154}Eu. The α-emitters were - with some uncertainty - ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{244}Cm and ^{246}Cm.

Based on the results on cells No. 5 and 6 and on the measured radiation levels in the remaining four cells it was concluded that performance of a similar water cleaning of cells No. 1 to 4 would not be justified. The arguments were as follows:

- the "hands-on" work will only transfer the activity to another temporary place, involving doses of up to 20 mSv/man for the available personnel;

- an estimated removal of some 40% of the activity will be helpful later, but it cannot be regarded as an essential benefit now;
- the radiation levels are now so low that final "hands-on" cleaning can be done, if sufficient personnel are available;
- delay of the final cleaning will result in lower doses.

5. Decontamination of cell ventilators and ducts (B.5.)

It was decided that some of the ventilation equipment for the cells could best be left in-situ, connected to the cell volume or isolated. It appeared useless to transfer some of this equipment to another temporary storage. The complete equipment for two of the cells was removed. The most contaminated parts of the remaining equipment will also be removed; what remains will be thoroughly sealed, whether or not being connected to the cell volume.

6. Decontamination of room surfaces (B.6.)

This item was not reached in the project time.

7. Removal of active drains (B.7.)

This item was not reached in the project time.

8. Remaining work

According to the plan the project should be finished now. The delay is mainly caused by:

a) An additional, unforeseen job under item B.2, "Removal of large contaminated equipment", consisting in reception, cutting, packing and shipping of a large amount of scrap material from another site at Risø.

b) A delay of job B.5, "Decontamination of cell ventilators and ducts". This job was planned performed in parallel with job B.4, "Decontamination of concrete cells". This was not possible because the acceptance of the final state for all the concrete cells could first be taken when all the cells were in their present condition.

The remaining work consists in the following tasks, which are considered relatively conventional:

- checking/cleaning/removal of all remaining cell ventilation equipment and filters;
- leak testing of cell volume including remaining connected equipment and ducts;
- checking/cleaning/removal of the room ventilation system;
- checking/cleaning of surfaces in all classified rooms;
- removal of the active drains in the building.

The project is planned to be finished by the end of July 1994. The full decommissioning project will require about 120 man-mSv, given to some 20 persons. The highest yearly man dose will be 12 mSv.

Dose meter No.	Dose rate [mSv/h]										
	Cell 1 a)	Shutter	Cell 2 a)	Shutter	Cell 3 a)	Shutter	Cell 4 a)	Shutter	Cell 5 b)	Shutter	Cell 6 b)
1	6.0	-	7.0	-	12.0	-	0.40	-	2.0	-	0.50
2	4.5	-	5.5	-	15.0	-	0.45	-	1.8	-	0.55
3	5.4	-	6.0	-	9.0	-	0.50	-	1.6	-	0.50
4	3.0	-	6.0	-	9.0	-	0.50	-	2.0	-	0.70
5	3.3	-	4.5	-	14.0	-	0.85	-	2.7	-	0.40
6	9.5	-	12.0	-	11.0	-	0.50	-	1.9	-	0.40
7	11.0	-	6.0	-	10.0	-	0.50	-	2.0	-	0.90
8	8.6	-	5.0	-	13.0	-	0.70	-	2.3	-	0.40
9	6.8	-	-	-	-	-	-	-	-	-	-
10	6.6	-	-	-	-	-	-	-	-	-	-
Average	6.5	0.026	6.5	0.050	11.6	0.021	0.55	0.015	2.0	0.008	0.54

a) Dose rate at 130 cm above table/floor after remote vacuuming, no in-cell high pressure water jetting.
b) Dose rate at 115 cm above table/floor after remote vacuuming and in-cell high pressure water jetting.

Table II: Final state (contamination) for the concrete cells.

Dose meter No.	Total contamination [GBq]										
	Cell 1	Shutter	Cell 2	Shutter	Cell 3	Shutter	Cell 4	Shutter	Cell 5	Shutter	Cell 6
1	621	-	389	-	668	-	24	-	89	-	21
2	418	-	397	-	1080	-	27	-	93	-	24
3	590	-	375	-	562	-	35	-	71	-	24
4	279	-	383	-	575	-	35	-	103	-	34
5	342	-	226	-	702	-	49	-	130	-	18
6	829	-	811	-	743	-	31	-	103	-	21
7	992	-	305	-	508	-	34	-	107	-	37
8	778	-	260	-	675	-	40	-	118	-	16
9	613	-	-	-	-	-	-	-	-	-	-
10	576	-	-	-	-	-	-	-	-	-	-
Average	604	0.228	393	0.438	689	0.186	34	0.133	102	0.071	24

8.7. FINAL CLEAN-UP OF THE PIVER PROTOTYPE VITRIFICATION FACILITY: DECONTAMINATION OF THE HOT CELL

Contractors: CEA-Valrhô
Contract No.: FI2D-0018
Work Period: July 1990 - June 1991
Coordinator: A JOUAN, CEA-Valrhô.
Phone: 33/66 79 63 76 Fax: 33/66 79 66 03

A. OBJECTIVE AND SCOPE

The PIVER pilot vitrification facility at Marcoule was operated between 1969 and 1980, first using a batch process to vitrify Gas-Cooled Reactor fuel element reprocessing waste, and then to develop a continuous process to vitrify Fast Breeder Reactor (FBR) fuel reprocessing waste. A total of 12 t of glass was treated. It was then decided to remove the equipment and clean up the cell in order to install new equipment (PIVER II).

PIVER is the first HLLW vitrification cell to be decommissioned. Under a previous contract (FI1D-0057), all process equipment items of the main cell were removed, followed by preliminary decontamination carried out in remote operation. So, the internal radiation level was reduced from several Gy/h to less than 10 mGy/h. The remaining radioactivity inventory is estimated at about 1.1×10^{13} Bq (300 Ci). At this level, access to the cell is now possible for durations not exceeding about one minute; the cell remains highly contaminated and requires the use of ventilated protective clothing under severe working conditions.

This project is aimed at continuing decontamination and dismantling work enabling further dismantling of in-cell equipment with hands-on techniques and finally to reach a radiation level allowing the installation of new equipment with standard working conditions for controlled zones. The generation of specific data on costs, working hours, job doses and the amount of created secondary waste is an important objective of this project.

The contractual work will result in assessed decontamination procedures for highly contaminated cells.

B. WORK PROGRAMME

B.1. **Dismantling of the telemanipulators in the PIVER cell including two MT 200 master-slave manipulators, a robot manipulator (CAROLINE) and a pantograph manipulator (ANTOINE).**

B.2. **Further decontamination of the PIVER cell with various decontamination techniques (chemicals using liquids, foams and gels, electropolishing, and cryogenics), accompanied by radiological measurements.**

B.2.1. First stage decontamination by short time in-cell work, aimed at strongly reducing the dose rates.

B.2.2. Second stage decontamination by long time in-cell operators'work.

B.2.3. Final decontamination aimed at obtaining standard working conditions for controlled areas in the cell (dose rate $\leq$ 0.1 mGy/h).

B.3. **Dismantling of the remaining pipes not needed for the future use of the cells.**

B.4. **Identification of the remaining cell internals by photogrammetry for facilitating design work for the reuse of the cell.**

B.5. **Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.1. to B.3.**

C. PROGRESS OF WORK AND RESULTS OBTAINED

This project was completed in 1991. The final report is available as EUR report No. 14764.

8.8. DESTRUCTION OF CONTAMINATED SODIUM OF THE PRIMARY CIRCUIT OF EXPERIMENTAL RAPSODIE REACTOR

Contractors: CEA-Cadarache
Contract No.: FI2D-0022
Work Period: July 1990 - June 1993
Coordinator: J ROGER, CEA-Cadarache
Phone: 33/42 25 76 45 Fax: 33/42 25 72 56

A. OBJECTIVE AND SCOPE

French regulations prohibit, for safety reasons, the disposal of sodium with other low-level radwaste in shallow land burial. The development of an industrial-scale procedure for the transformation of sodium into an acceptable product is thus a useful target generally for all LMFBRs.

The CEA has developed, at laboratory-scale, the so-called NOAH procedure transforming sodium by controlled addition of water into aqueous sodium hydroxide.

The objective of the present contract is to conceive and manufacture an industrial-scale facility (600 Kg/d), based on the NOAH process and its application to 13 t (out of a total of 37 t) of contaminated sodium (specific activity 4.1 KBq/g, mainly Cs-137) from the RAPSODIE pilot FBR. The facility will be conceived thus (mobile system, limited dimensions, easy adaptation), that it can be used on other FBR-sites.

The facility will be installed at the containment building of RAPSODIE (DESORA programme). Contractual work will be implemented in cooperation between two departments of the CEA-UDIN (Unité de Démantèlement des Installations Nucléaires) and LEPE (Laboratoire d'Etudes, de Procédés et d'Expertises).

In particular, the generation of specific data on costs, working hours and job doses as well as on the amount of created secondary waste is considered as an important objective of this project.

B. WORK PROGRAMME

B.1. Conceptional studies
B.1.1. Studies for the industrial application of NOAH (LEPE)
B.1.2. Studies for the installation of NOAH into the RAPSODIE containment building, including the needed auxiliary equipment (UDIN)
B.2. Manufacturing, installation and testing of equipment
B.2.1. Manufacturing and installation of equipment (UDIN)
B.2.2. Commissioning, testing of equipment and operator training with non-radioactive sodium (UDIN)
B.3. Main operation for the transformation of sodium (UDIN)
B.4. Conditioning and disposal of generated liquid waste
B.4.1. Investigations into possible ways for utilisation or treatment of waste including associated costs (LEPE)
B.4.2. Temporary storage of liquid waste (UDIN)
B.5. Technical and economical balance on the feasibility for an industrial application of NOAH (UDIN)
B.5.1. Preliminary balance before main operation
B.5.2. Final balance after main operation, including generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.3. and B.4.

C. PROGRESS OF WORK AND RESULTS OBTAINED

The DESORA (Destruction du sodium de RAPSODIE) tests with non-active sodium (started on 28 October 1992) continued until 23 February 1993, resulting in the destruction of 834 kg of sodium in approx. 25 hours of operation at different flow rates.

The experience gained from the tests allowed drawing a list of technical modifications and improvements as well as of the operational parameters. Modification work on the technical equipment started on 28 June 1993 and the documents necessary for the complementary tests were written.

The NOAH prototype N-loop functioned from March 3 to April 10, 1993. It allowed to verify the validity of the performed modifications, to test reliability of the LEWA equipment and of the de-humidification unit.

Furthermore, a series of gamma spectrometry calibration measurements were carried out. The measurement points on piping and the DESORA tanks were defined.

Conclusion

It is reminded that the objective of the contract was the destruction of 13 out of 37 t of very low-contaminated sodium (mainly Cs-137) by an industrial-scale facility based on the NOAH process. It can be stated that this objective was not reached within the contract's period as none of the contaminated sodium was tackled. This is planned to be carried out during the second semester of 1993.

8.9. DECOMMISSIONING OF THE JEN-1 EXPERIMENTAL REACTOR

Contractors: CIEMAT, ENRESA, ENSA, LAINSA, UH-IW
Contract No.: FI2D-0023 / 0062
Work Period: July 1990 - June 1994
Coordinator: L MAÑAS, CIEMAT
Phone: 34/1/346 60 00 Fax: 34/1/346 60 05

A. OBJECTIVE AND SCOPE

JEN-1 is an experimental reactor of the swimming-pool type, moderated and cooled by light water, with a power of 3 MWt. It was operated from 1958 till 1984 with a total generated energy of 2,700 MWd. The radioactive inventory is estimated in the order of 3.5 x 10^{11} Bq (9.5 Ci), the dose rates are estimated in the range of 20 to 150 mGy/h.

The main aim of this project is the study and development of decontamination, cutting and melting techniques on contaminated or neutron-activated aluminium components of JEN-1.

Underwater segmenting of aluminium components still represents some problems to be solved, which consist especially in the limited visibility of the cutting environment, due to an important amount of very small suspended articles (10%, compared to steel 1%) and in the difficult filtration of these particles. An important aspect relevant to safety is the high H_2 generation rate due to a rather long lasting reaction of molten aluminium particles with the surrounding water.

Industrial-scale melting of aluminium components still needs development work for appropriate foundry techniques, especially concerning crucible material and slag formation.

Results obtained in this contract will be useful in the future for the dismantling of numerous research reactors with aluminium components.

The project is expected to produce specific data on costs, working time, waste arisings and radiation exposures to operators for the dismantling of the JEN-1 reactor.

After the second call for proposals in Section C a follow-up contract was concluded for the dismantling of further components: the primary circuit coolant collector and the support structures of the ionization chamber. Work in contract FI2D-0023 was limited to following components: core grid, support grid and control blade housing. Work in the second contract will need an adaptation of techniques and procedures already developed in the first work-programme for components having now larger wall thicknesses and different geometric configurations.

The work programme will be implemented in co-operation between following Spanish organisations: CIEMAT, ENRESA, ENSA and LAINSA, and with Institut für Werkstoffkunde of Universität Hannover (UH/IW), CIEMAT being the coordinator. A co-operation on aluminium melting will be installed with Siemens AG KWU Group and Siempelkamp Giesserei Krefeld (SG).

B. WORK PROGRAMME

B.1. Radiological characterisation of components to be dismantled, and of melting products (CIEMAT)

B.1.1. Radiological characterisation of the grid and support grid.
B.1.2. Radiological characterisation of the control blade housings.
B.1.3. Radiological characterisation of the primary coolant circuit collector.
B.1.4. Radiological characterisation of the support structure for ionization chambers.
B.1.5. Radiological characterisation of the melting products.

B.2. <u>Development, manufacturing, testing and subsequent installation in the JEN-1 reactor of an underwater cutting facility</u> by plasma arc and by consumable electrode techniques (UH-IW, CIEMAT).

B.2.1. Development and manufacturing of prototypes of plasma arc torch and consumable electrode torch (UH-IW)

B.2.2. Cutting tests with both tools on representative aluminium sheets, aiming at defining optimal working parameters, cutting effluents and appropriate air and water filters (UH-IW, CIEMAT)

B.2.3. Comparison of both tests with respect to cutting performance, generation and type of cutting effluents and tool handling abilities with subsequent selection of the most appropriate tool (CIEMAT + UH-IW)

B.2.4. Design and manufacturing of a cutting facility, including the selected cutting tool, handling and sensor systems and the cutting cell (UH-IW)

B.2.5. Testing at UH-IW and optimisation of the whole system in water depths of 5 m (UH-IW, CIEMAT)

B.2.6. Training of the CIEMAT staff at UH-IW (UH-IW + CIEMAT)

B.2.7. Transport and assistance for the installation of the cutting facility in the JEN-1 reactor (UH-IW)

B.3. <u>Underwater dismantling of reactor internals</u> after preceding dismounting work (CIEMAT + UH-IW)

B.3.1. Dismantling of the grid and grid support

B.3.2. Dismantling of the control blade housings

B.3.3. Dismantling of the primary circuit cooling collector

B.3.4. Dismantling of the support structure for ionization chambers

B.4. <u>Decontamination of reactor internals</u> (ENSA, LAINSA).

B.4.1. Selection of suitable procedures with respect to decontamination efficiency, amount and type of arising secondary wastes, reprocessing abilities and radiological impact

B.4.2. Decontamination of the grid and grid support

B.4.3. Decontamination of the control blade housings.

B.4.4. Decontamination of the primary circuit coolant collector

B.5. <u>Melting of aluminium waste</u> (CIEMAT, ENRESA)

B.5.1. Selection, manufacturing and adaptation of a melting furnace and implementation of cold melting tests.

B.5.2. Main melting programme, including generation of data on volume reduction and decontamination effects.

B.6. <u>Assessment of radiation protection</u> including both the personal and the ambient radiological impact (CIEMAT, UH-IW)

B.6.1. Assessment of radiological impact during dismantling operations (CIEMAT, UH-IW)

B.6.2. Assessment of radiological impact during decontamination operations (CIEMAT)

B.6.3. Assessment of radiological impact during melting operations (CIEMAT)

B.7. <u>Generation of specific data</u> on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.3., B.4., B.5. and B.6.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

During 1993, the activities related to the following tasks:

Final revision of characterization relating to JEN-1 internals.

Improving of designs about modular plasma torch and consumable electrode water jet device.

Completion of manufacturing and introduction in the JEN-1 pool of the cutting basin where internals will be cut, in order to keep contamination from spreading all over the pool.

Design completion of handling device for cutting tools and manufacturing by ENSA (Equipos Nucleares).

Development and manufacturing of the control system for the fine positioning of cutting tools.

Progress and results

1. Radiological characterization of components to be dismantled, decontaminated and melted (B.1)(CIEMAT).

Though such a characterization was practically completed during 1992, α and β emitters failed to be reported as two JEN-1 internals were concerned: grid support and collector of primary cooling circuit. Table I shows the whole radioactive content for all internals of JEN-1 reactor, after a late screening of results obtained.

2. Development, manufacturing, testing and subsequent installation of an underwater cutting facility (B.2)(CIEMAT, IW).

2.1 Development and manufacturing of prototypes of a plasma torch and a consumable electrode torch (B.2.1)

The three modules of plasma arc torch concerning the first version were improved by an-only-two modules plasma arc torch, in order to avoid loss of power of the igniting current and damages especially in the conecting areas, just like to avoid a secondary arc occurred between modules.

Likewise, the consumable electrode water jet cutting, to be clamped to the swivel and pitch device, was redesigned so as to obtain a better cutting performance.

2.2 Design and manufacturing of a cutting facility (B.2.4)

2.2.1 The cutting basin

The manufacturing of the basin where the components of JEN-1 will be cut was introduced, with its internals (grid, funnel and dross collecting basket) inside JEN-1 pool, (Fig. 1).

2.2.2 The handling device for cutting tools (Fig. 2-3)

The device was designed by IW (Hannover University) and manufactured, on charge to CIEMAT, by ENSA (Equipos Nucleares) in its installations of Santander (North Spain).

It is provided by three translatory axes and two rotational ones, these late by means of a swivel and pitch fastened to the bottom of the vertical mast. The three translatory axes are achieved by a gantry, which moves along the reactor pool of JEN-1, the cradle which moves horizontally along de gantry across the pool and the vertical mast which is carried by the cradle and can be lowered five meters into the pool.

The swivel and pitch, carrying the cutting tools, is provided with two rotational axes allowing to turn the tools around the horizontal and vertical axes.

Gantry, cradle and vertical mast are driven by means of servomotors with cyclo-gearing installed and open toothed belts, enabling the acceleration of each component of the handling device to a velocity of 3000 mm/min within one second.

2.2.3 The control system

Such a control system is related to the one concerning the fine positioning of cutting tools. Its development and manufacturing were achieved during the present year.

The system is a sensor one called INDUS (inductive-ultrasonic) to be used for the orientation and distance to metallic structures, and for the measurement of thickness for underwater components.

It includes the following elements: sensor head, transmitter-receiver unit and control units. The sensor head is composed of a sealed housing covering an ultrasonic sensor and, concentrically around it, an eddy current sensor. The latter is made of two pairs of spools arranged crosswise, which enables the obtention of information on the distance to the work piece, as well as the inclination of the sensor over two planes placed perpendicularly to each other and over the workpiece.

The transmitter-receiver unit feeds the transmitter spools and converts the signals of the receiver spools into an analogous voltage. The inductive control unit maintains the power supply for the complete eddy current sensor system, and is the indicating device for the eddy current sensor signals, one for distance and two other ones for orientation.

The system is completed by an ultrasound analyser, enabling water distance measurement, as well as workpiece wall thickness, by means of the ultrasonic sensor placed in the sensor head. Fig. 4 shows a photograph of the INDUS.

Table I. JEN-1 Reactor Internals : Main Nuclides Specific Activity (Bq/g)

Nuclide	Grid	Housing No. 1	Housing No. 2	Ion Chamber Support	Collector	Grid Support
Fe 55	2.90E+02	3.00E+04	2.50E+04	3.10E+02	2.50E+03	4.30E+03
Co 60	1.10E+04	1.70E+05	1.40E+05	1.00E+04	6.80E+02	8.70E+02
Ni 63	5.50E+01	3.20E+02	3.10E+02	5.00E+00	8.70E+01	7.20E+02
Nb 93	-	-	-	-	1.40E+03	5.80E+03
Nb 94	-	-	-	-	5.00E+00	-
Eu 152	-	7.90E+02	2.20E+02	8.00E+01	-	-
Eu 154	-	2.10E+03	2.40E+03	-	-	-
Eu 155	-	-	6.00E+02	-	-	-
Cs 137	5.70E+01	1.80E+03	1.70E+03	6.10E+01	1.00E+02	1.40E+01
Sr 89	-	4.20E+02	2.10E+02	3.00E+00	-	6.80E+00
Sr 90	5.30E+01	1.60E+03	1.60E+03	7.30E+00	1.10E+01	1.00E+01
Pu 238	8.00E-01	3.90E+01	3.10E+01	-	-	-
Pu 239	2.60E+00	6.60E+01	5.80E+01	-	2.00E-01	4.00E-01
Pu 241	-	2.60E+03	2.30E+03	-	-	-
Am 241	4.00E-01	9.60E+01	4.30E+01	-	-	-
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Cm 244	-	1.00E+00	3.00E-01	-	-	-

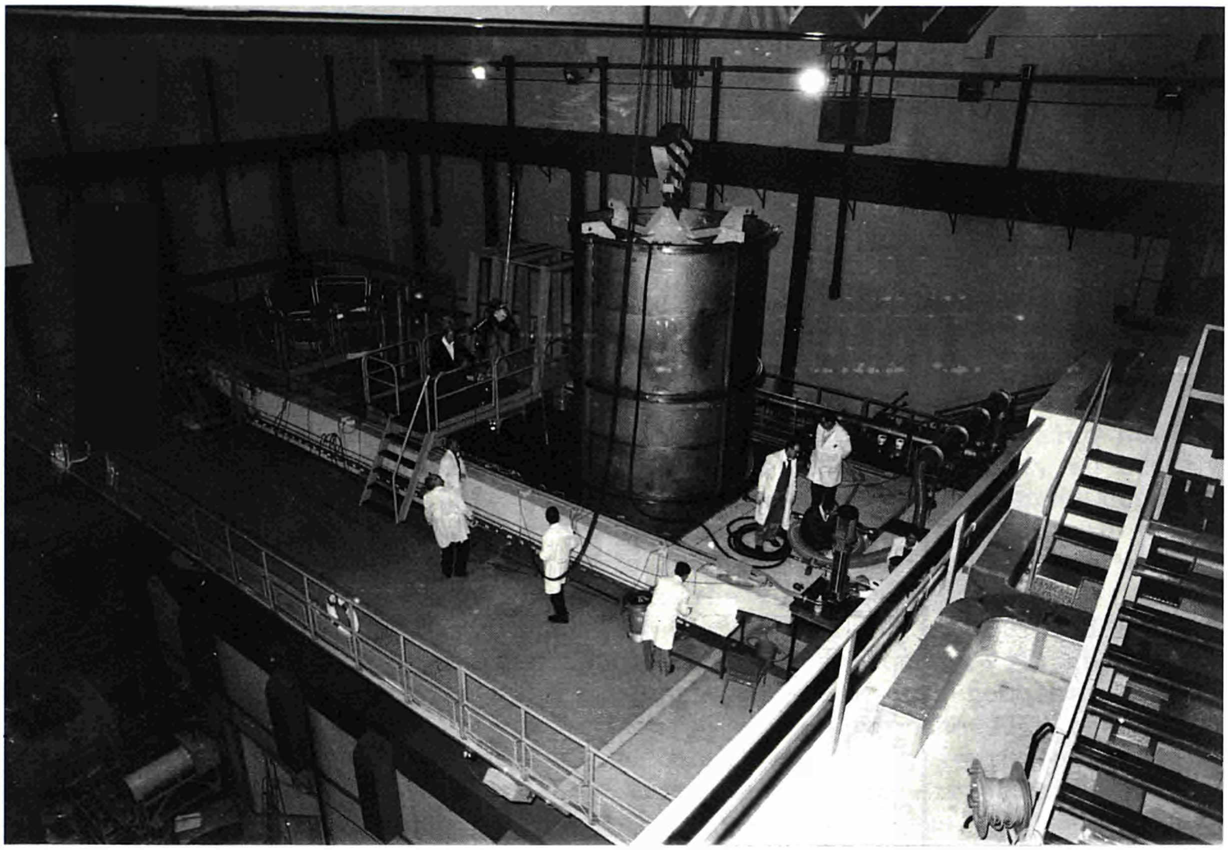


Fig. 1 - Cutting Basin

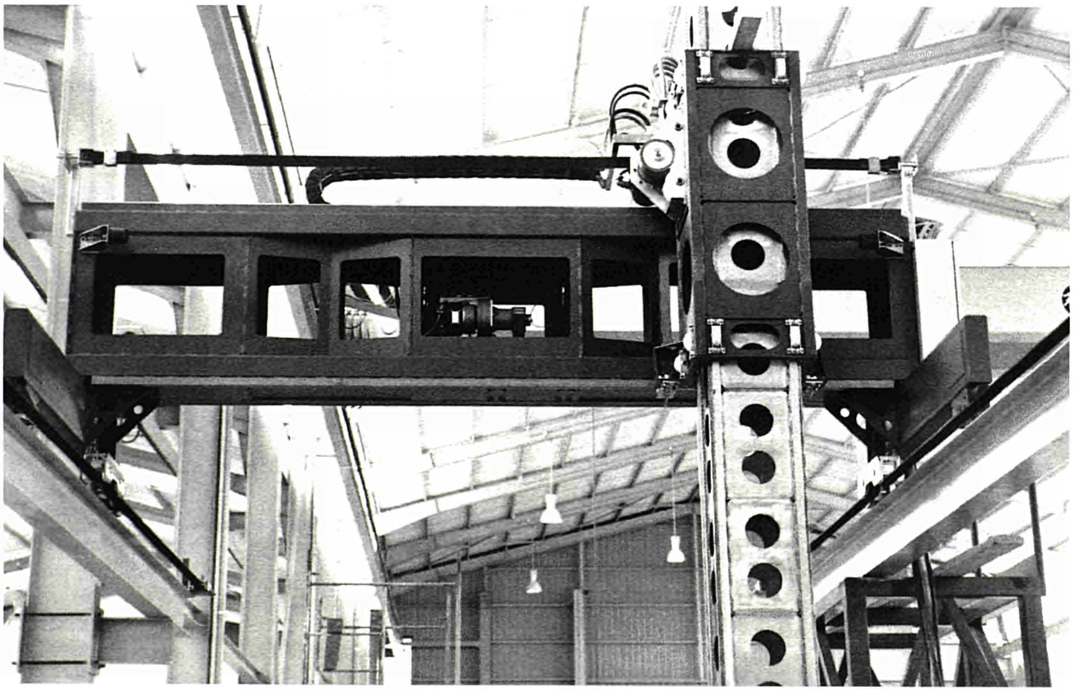


Fig. 2 - Handling Device

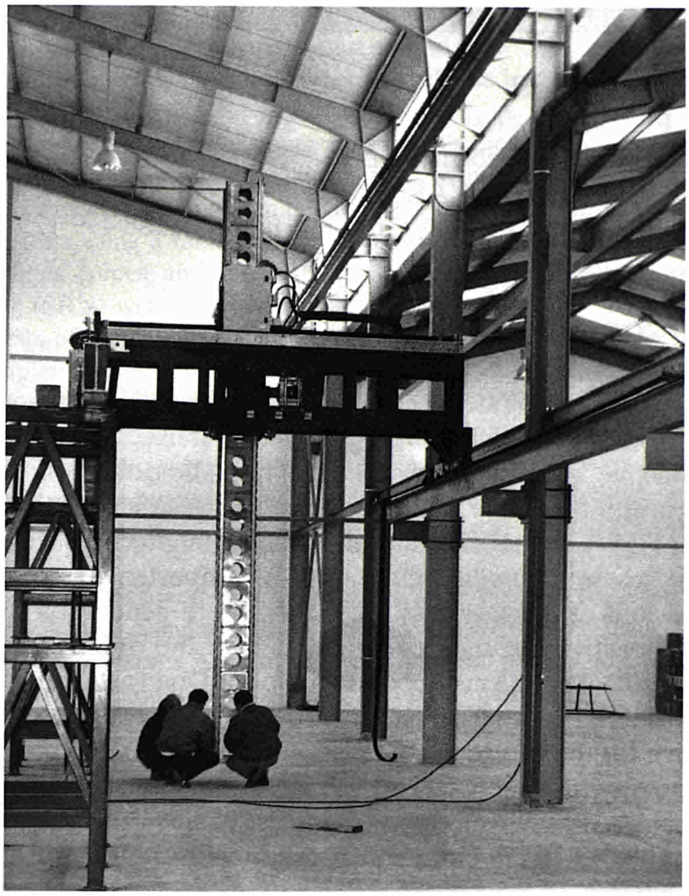


Fig. 3 - Handling Device

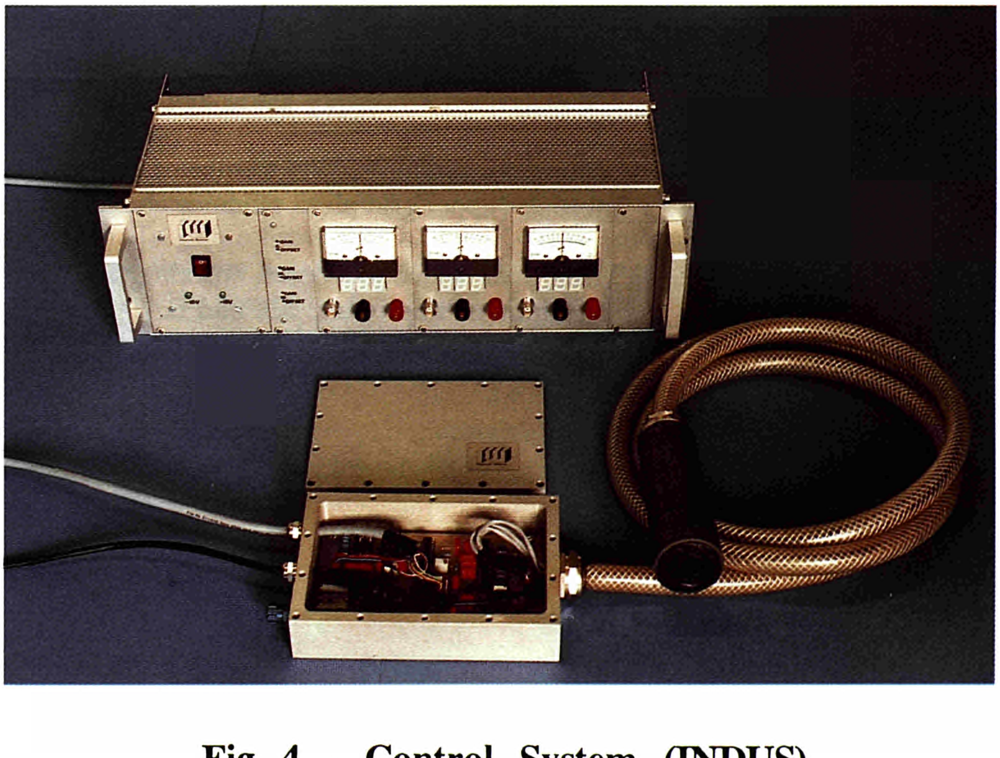


Fig. 4 - Control System (INDUS)

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 1994
Coordinator: H H KALWA, VAK
Phone: 49/6188/499 136 Fax: 49/6188/499 125

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 BR3/Mol (§ 8.3.) and KRB-A (§ 8.5.).

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 reactor components
- B.4. Dismantling of a series of RPV internals (upper grid plate, chimney above the core, control systems)
- B.5. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2., B.3. and B.4.

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

To avoid a significant secondary and tertiary waste arising important targets of the work during the reference year are a **further simplifying of the tool support**, designing reduction of the tool support force to a more practical level, minimization of aerosol and secondary waste generation and designing universal usable cutting and handling principles to reach synergetic effects in other NPP and even for multiple use, if possible.

For the dismantling of the complex geometry internals the following **Special Tool Supports (SWT)** were used. The **SWT1** for sawing and nippler cutting tests with pneumatically driven tools, the **SWT2** for guiding the corresponding electro-discharge machining (EDM-) head, supplied by different electrodes, the **SWT3** for laser cutting tests using a rack-and-pinion supported laser tool.

The cross-sections (pipes and the distance plates) of the mock-up of the sprinkler ring for emergency core cooling (SRN) were successfully cut by **hacksawing.**

Dismantling tests of a chimney above the core (KMK-) mock-up in air were performed with the newest commercial **nippler** tool. The handling of the nippler tool used was practically **reaction-force-free because of the closed force current.**

Application of the **EDM** technology to **thin-walled** internals (KMK, primary water distribution ring [KVR]) was proved as a possible method for their dismantling. The KMK mock-up was cut by various **electrodes (pure copper, tungsten/copper** and even **CFC = carbonfibre reinforced graphite)** with different results. Its application to relatively **thick-walled** internals (thermal shield) is the subject of other decommissioning projects. /1/

Laser cutting experiments were repeated with a modified Nd-YAG laser of the last commercial power generation in air. As the original material (X10CrNiNb18.9) was not available, a 8mm sample of X8CrNiTi1810 material was chosen. A cutting **kerf of approx. 0.5mm** was achieved which is of a vital advantage in view of a **low kerf material generation and of a small specific secondary waste arising.**

Progress and Results

1. Preliminary tests on non-radioactive components including devices for segmentation and remote operation techniques, definition of generated secondary waste, and studies of dismantling scenarios (B.2.)

1.1 Devices for segmentation

For the above-mentioned necessity of reduction of the tool reaction force, the contractor tried to use cutting technologies with so-called **indefinite cutting edges** (grinder), tools with an **alternating reaction force** (reciprocating or hacksaw), **EDM** methods, **laser** methods and **nippler** or shear methods.

In the VAK test installation were successfully cut the cross-sections of pipes and distance plates of the KMK mock-up by **hacksawing** (Figure 1). Cutting was performed under water with a modified pneumatically driven **hacksaw** (Figure 2).

EDM tests were performed with a Siemens/KWU EDM equipment in Erlangen (Figure 3). The cross-section of a 1m long plate of a 6mm X8CrNiTi18.10 material (KMK) were successfully cut in one step under water (1.20m).

The application of **laser** cutting (percussion laser drilling) is of more interest, even considering the aerosol generation because of the expected thin cutting kerf (0.5mm) and the corresponding small kerf material generation. To gain a better understanding of this cutting technology under NPP dismantling conditions, **further preliminary tests** were performed with a modified Nd-YAG laser of the last commercial development in air. The possibility of placement of the source outside the controlled area is of decisive advantage. The source power was of 30 kW. Because of the evaporated metal waste and the water whirling caused by process gases, the waste collection will be performed by a strong suction unit and by 1μm filters with a high loading capacity.

Nippler cutting tests in air were performed with the above-mentioned KMK mock-up. The cutting kerf was 12mm wide. The handling of the nippler tool used was practically reaction-force-free because of the closed force current inside the tool.

1.2 Remote operation techniques

The importance of a master slave manipulator (MEM) in the dismantling concept is considerable. It will not only be used for important **auxiliary tasks** such as adjusting video system, placement of suction nozzles, supporting the tool changing, assisting crane rigging, support of the container handling devices etc.

As a step before specifying the tasks for a complicated master-slave-manipulator, a pneumatically driven, so-called **trunk-like manipulator were chosen** to gain experience in handling. This was done because of the costs of an EMSM. Trunk-like manipulators are of advantage in handling because of a simple management. Two pneumatical cylinders are able to manage the manipulator's hand inside a semisphere. Each cylinder is assigned to one direction of motion (backwards-forwards and the other one then left-right). Thus, it works without being prone to disturb computer-aided translation.

1.3 Definition of generated waste

A question of increasing importance will **not be the total mass of secondary waste in itself, but the disposal volume** which is necessary for the packagings of the **primary, secondary and tertiary waste.** The amount of secondary and tertiary waste depends on the decommissioning technology, especially on the cutting method.

Thus, an important target of the work is to minimize the disposal volume by minimizing the generation of specific waste (g/cm²), maximizing the cutting swarf size, without using heavy and large tools. To **minimize the specific cutting swarf** cutting technologies generating less swarf were planned, preferably shearing (rollshearing, nibblers). To **maximize the cutting swarf size** technologies generating a particle size >0.5mm (appr. 90% of all particles) were chosen. One reason for this is the high separating speed and the resulting constant water visibility without simultaneous sucking of the swarf. The other one is the possibility to use simple strainers for swarf collection not during the cutting phase, but **after** performance of several cuts. This will lead to the aimed separation of different handling tasks. The strainer meaningfully should be of the same material as the kerf material. Strainer units are of the lowest cost. Its effective loading density is the highest in comparison with all other types of filters because of the absence of filter cartridges. Filter cartridges are mostly of plastic material and therefore the packaged waste will be of a mixed type, which is a disadvantage for disposal. For specific data see chapter 4.

1.4 Studies of dismantling scenarios

Extended preliminary studies about different dismantling scenarios were already performed and a decommissioning technology was chosen, allowing a low level of personnel doses and a low disposal volume by use of a clean cutting, handling, decontamination and works for waste treatment. /2/

Thus, the parameter "cutting speed" was not optimized because of the emphasis on the above-mentioned **reliability of the whole decommissioning technology.** Nevertheless, the parameter "cutting speed," was always observed but primarily to define the **tendency** of the more important parameter of cutting tool steadiness.

The present cutting layout for the KMK, sprinkler ring for emergency core cooling (SRN), condensate distribution ring (KVR), the upper grid plate (OGP), the core case (KML) and the lower grid plate (UGP) will be formulated in more detail after a sampling and gamma monitoring of the OGP, which is probably one of the most activated components and the measurement results will give the lowest error level.

2. Qualification of dismantling procedures for application to reactor components (B.3.)

After opening the pressure vessel, mock-ups for the reactor components were constructed .

2.1 Dismantling of the sprinkler ring for emergency core cooling (SRN) mock-up

Cutting and handling tests were performed with a sample of the sprinkler ring for emergency core cooling, which would be the first reactor internal to be dismantled because of its location in the RPV. The cutting sample was placed in the VAK test installation at the original distance from the tool operator (Figure 1).

A modified **pneumatically driven hacksaw** (PSS1) was used, manufactured by the German "Fein" company (Figure 2).

For cutting the pipes of the sprinkler ring for emergency core cooling, a tool support was constructed, which met the following requirements: easy to handle, does not need same manipulation of cutting force on the operator's level, the tool support force depends on the preliminary chosen strength of the spring and can be altered by the pneumatical cylinder.

For fulfilment the above requirements, the hacksaw was modified with view to redirected exhaust air location, increased alternating gear inner-pressure and improved tool support.

A so-called "self-supporting" tool support system was chosen. The tool supports itself after accurate placement to the component to be cut, and this is done independent of the operator's ability. For the operator it is not necessary to make a defined supporting force, but only to switch on the saw and make the pneumatical cylinder pressureless.

The handling and performance of this tool now corresponds to the planned objective, thus, the tool together with its support is ready for the authorities qualification.

2.2. Dismantling of the Chimney above the core (KMK) mock-up

The KMK dismantling is presently performed by two methods: EDM and nippler cutting. The application of **EDM** allows the treatment of the whole KMK, but the dismantling of the 5 mm plates of a length of 900 mm will take significant time and generates a suspended secondary waste. Thus, the use of a nippler tool is meaningful and is not only a back-up technology.

The KMK mock-up was similar to the geometry of the original. The thickness of the plates was 6mm and conservatively chosen (4.75mm original). The chimney plates were welded on the flanges, which were of the original thickness of 25mm and also of the same material. The currently existing cutting layout foresees a simple cut through the chimneys' half. For EDM cutting the sample was placed in the KWU test installation in Erlangen. The equipment is shown on Figure 3. The KMK mock-up was cut by various electrodes (pure copper, tungsten/copper and CFC= carbonfibre reinforced graphite electrode). The electrodes used were of a length of 900mm and of a width of 60mm. It was therefore possible to cut a chimney plate along the whole length simultaneously. The electrode was very slim (900x60x2mm) for an easy insert into the chimneys' vertical channels. Using the **CFC-electrode** 50% more erosion was obtained, but it generated a significant amount of hydrogen bubbles and of **Ni-, Fe- and Cr-ions,** which turned the water blue/green (typical for a high power density). Activated ions are of a great disadvantage even in comparison to small particles because of the **additional necessity of an ion exchanger unit,** which would not simplify the periphere handling tasks (personnel dose, waste). First cutting tests with a CFC electrode were performed at about 60A. After the above-mentioned disturbances, **Cu electrodes** which work by 40...50A were used. Better results were achieved with **W/Cu-electrodes,** which have the lowest self-erosion in comparison to the cheaper Cu-electrodes. The thicknesses were 2mm (CFC) and 0.5, 1.0 and 2.0mm for Cu electrodes. The best performance was achieved with the last two electrode thicknesses. It was possible to adjust the electrode in a coarse position supporting the EDM head by a simple spindle gear and to use the automatic fine tuning controlled by the generator, so the handling of the EDM equipment is relatively reliable. The cut of a cross-section of 900x6mm continued 150 minutes (W/Cu) and 230 minutes (Cu). The currently existing cutting layout foresees 8 cuts of 230 minutes, 8 cuts of 100 minutes and 2 cuts of 200 minutes through the chimneys' half, accordingly the whole cut will take 20h.

Nippler cutting tests were performed with a pneumatically driven tool in air for KMK dismantling in the VAK test installation at Kahl. It was possible to cut the KMK, except the flange. It would be possible to guide the tool by a rod of the SWT1 type (similar to the tool support for the PSS1 hacksaw).

2.3 Dismantling of the condensate distribution ring (KVR) Mock-up

Cutting and handling tests on a sample from the KVR are being performed in the KWU test installation. The present dismantling concept foresees a cutting of the KVR into four pieces. The W/Cu electrode was guided by the same EDM support as above, but in vertical direction. The reason for that lies in the possibility to perform one cut without change. The electrodes' dimensions were 2x74x150. The burn-up of the electrode was about 60mm along the plates. The cut continued 8 hours and would have been 400 minutes by consequent use of W/Cu electrodes.

2.4. Dismantling of the control rod mock-up

The stellite balls of the Control rod which are of guiding tasks consist of an alloy which contains >63% cobalt and has led to a specific activation of 1.3E11Bq/g up to now! The removal of the stellite balls is meaningful, especially in view of the much better packaging conditions for the whole control rod prepared in this way. Although the application of EDM for stellite ball removal seems to be a more surgical application, it works safely and in a well reproduci ble way.

3. Dismantling of a series of RPV internals (B.4.)
No results obtained to date.

4. Generation of specific data (B.5.)

For comparable data, the specific secondary waste generated is calculated in **gramms per 1 cm² of cut cross-section.**

The secondary waste generated by use of the PSS1 **hacksaw** is of a coarse, easy-to-collect swarf (0.5...1.0mm). Using a hacksaw, the average specific waste generation is in the range of 1.0g/cm².

The specific secondary waste amount generated by use of **EDM** is in the order of 1.0g/cm² (copper electrode, thickness 1mm, burn-up=2.5:10) and of 2.0g/cm² (CFC= carbonfibre reinforced graphite electrode, thickness 2mm, burn-up=3:10. The secondary waste consists of the cutting kerf material (swarf) and of the burn-up of the electrode, but is nevertheless easy-to-collect because of their slow method of generation and the fact that there is no disturbance by whirling caused by process gases or thermal sources.

By use of **laser**, the secondary waste generated is appr. 0.5g/cm² cut cross-section. Because of the evaporated metal waste, its high generation speed and the water whirling caused by process gases, the waste collection should be performed by a strong suction unit and by 1µm filters with a high loading capacity. The achieved cutting **kerf of between 0.4...0.6mm,** which is of a decisive advantage with respect to a **low kerf material generation allowed an expected small specific secondary waste arising.**

The secondary waste generated by use of the **nippler** tool is of a kerf material, which consists of **coarse chips** of 12mm x 6mm x the material thickness, which can presently be up to 6mm stainless steel or 10mm carbon steel. Although the specific "secondary" waste generated by nippler is approx. 9.6g/cm², the big size of chips, which can be identified as an **evoluted primary waste**, allowed an economical storage.

For further **comparison**, using **diamond cutting** 2.0g/cm² were generated, and for **corundum grinding cutting** 0.5g/cm² of an easy-to-collect coarse swarf. The **consumable electrode water jet cutting** generates 14g/cm² of partly suspended particles. /3/

5. Conclusions

Hacksawing is interesting with respect to the large swarf particles generated, practical tool steadiness and constant water visibility without cleaning. There is no practical power dissipation of the tool drive when working under water. In comparison to the hacksawing results, the **EDM** application to thin-walled complex geometry internals is a method with the tendency to surgery cuts. The kerf is relatively small (0.5 ... 1.5mm) so the EDM generates less secondary waste while cutting. As the cutting swarf consists of relatively fine particles, the swarf must be sucked up simultaneously. The sucking efficiency is easy-to-reproduce due to the very **slow method of swarf generation** and the sucktion is **not disturbed by bubbles of process gases** or **by thermic effects**. The generation of electrolysis gases is not a practical problem. Application of EDM to relatively **thick-walled** internals (thermal shield of BR 3, 76mm) is the subject of other decommissioning projects. **Laser** cutting is interesting because of the very thin kerf obtained while cutting in air, which is also expected under water. Because the cutting swarf consists of relatively fine particles and the necessity **of process gases**, the swarf must be sucked up simultaneously. Due to the very **fast method of swarf generation,** the suction will be inefficient or very expensive. Application of **nippler** cutting to **thin-walled** internals proved to be a possible method for their dismantling up to a material thickness of 6mm. Nippler tools are more interesting in respect to the large chips generated, reaction force-free handling, practical tool steadiness and constant water visibility without cleaning.

References

/1/ FRAMATOME, S.C.K./C.E.N., SIEMENS; "The BR 3 Pressurized Water Reactor pilot dismantling project", Contract FI2D-0003, 4th Progress Report

/2/ NIS/NUKEM, "Weiterführende Bewertung von Stillegungsstudien", 02/91

/3/ CIEMAT ENRESA ENSA LAINSA IW(UH) "Decommissioning of the JEN-1 experimental reactor", Contract FI2D-0023, 3rd Progress Report

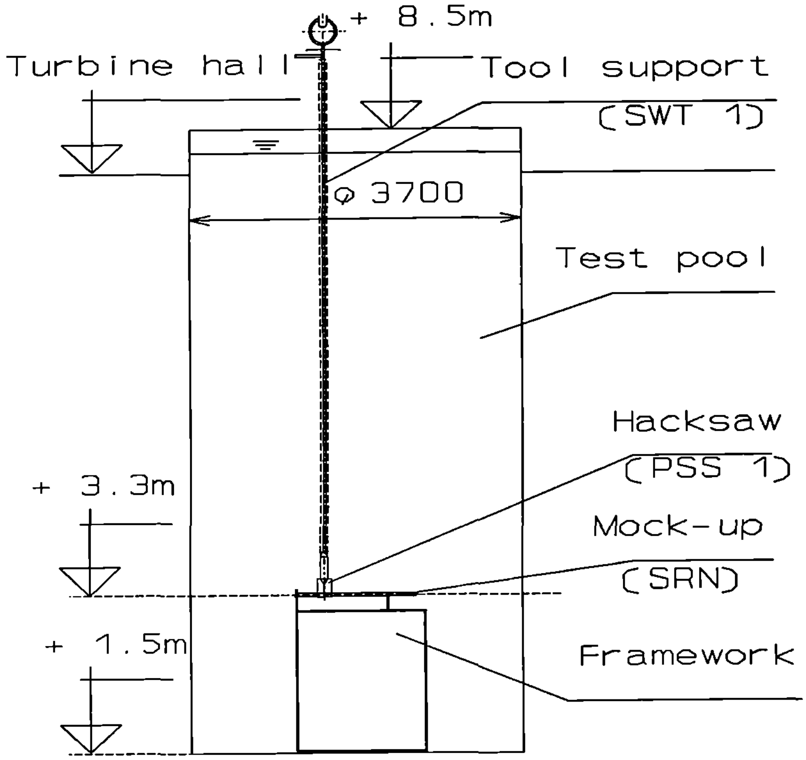


Figure 1: VAK test installation

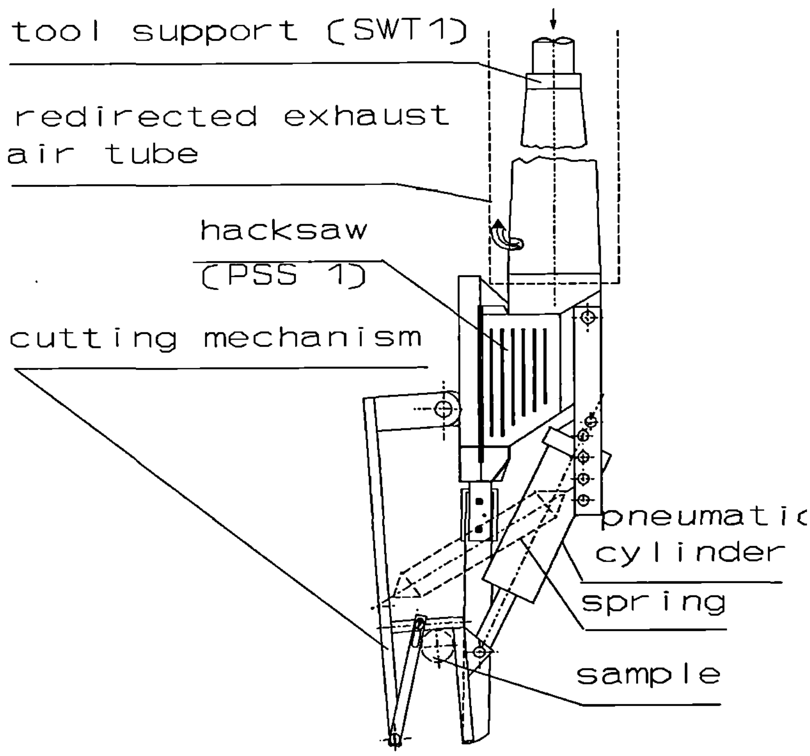


Figure 2: Pneumatical hacksaw (PSS1), sketch

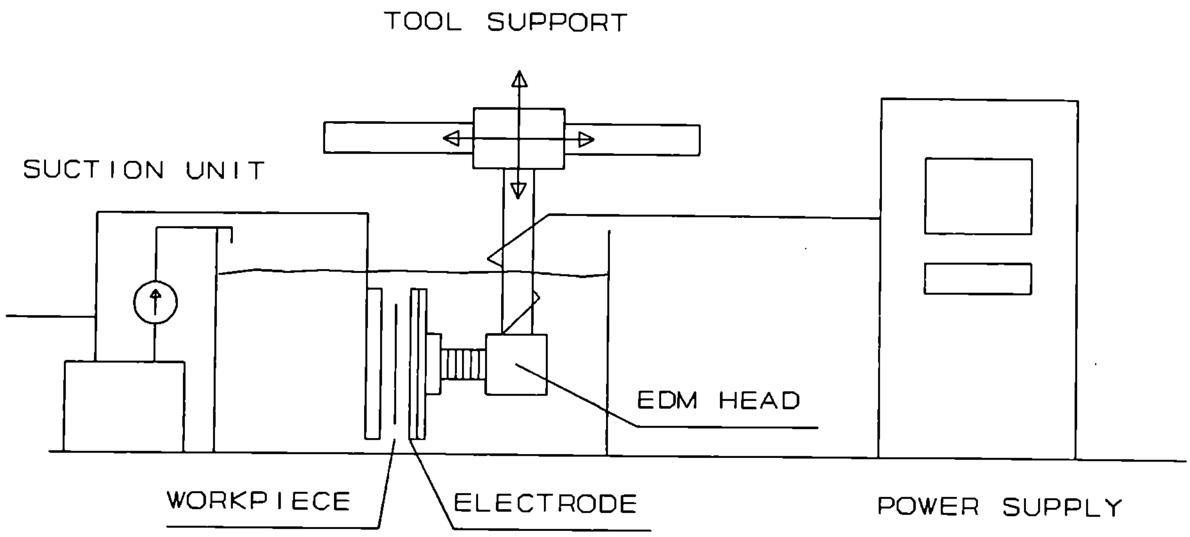


Figure 3: EDM equipment

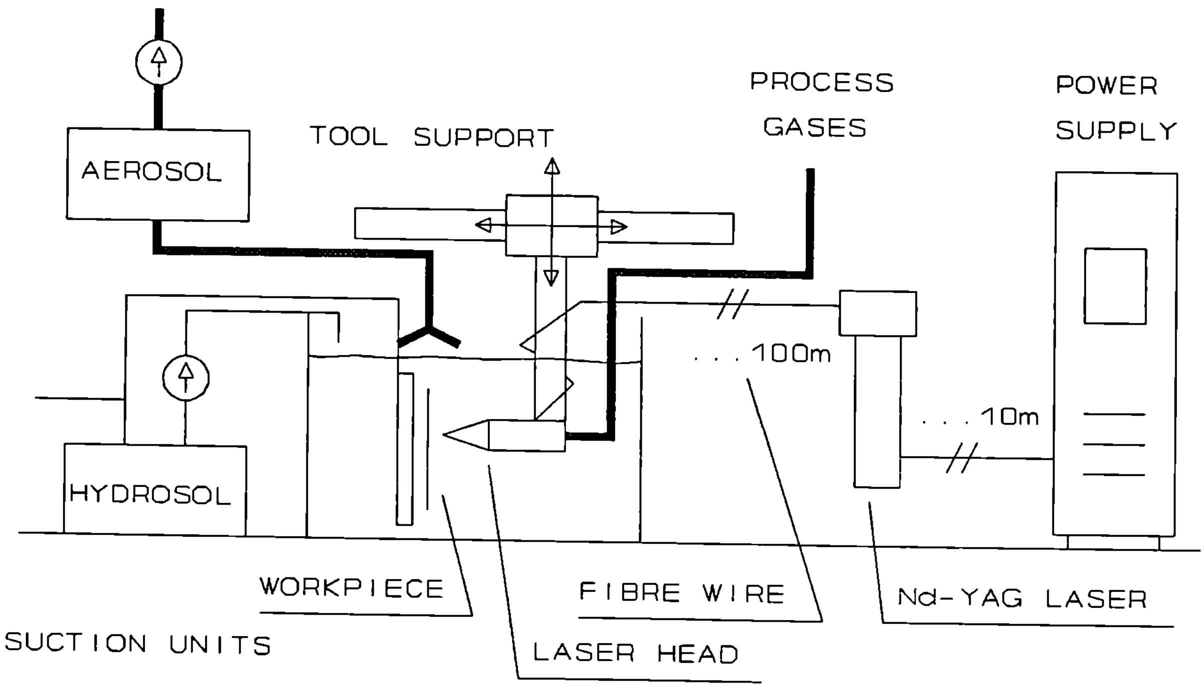


Figure 4: Laser equipment

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 - June 1994
Coordinator:	J S FEAUGAS, CEA-Valrhô
	Phone: 33/66 79 62 89 Fax: 33/66 79 66 61

A. OBJECTIVE AND SCOPE

The objective of the **present contract** was 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 of ferritic steel having a specific contamination in the order of 20 - 40 Bq/cm^2.

The objective of the **supplementary agreement** is to determine the U-distribution in the molten product during a large-scale melting campaign of approx. 100 t ($\pm$ 8 melts of 12.5 t each) of mild steel contaminated with uranium originating from the dismantling of the Pierrelatte reprocessing facility (Usines basse et moyenne).

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 the 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.**

SUPPLEMENTARY WORK PROGRAMME

B.1. Investigation of operational conditions including a minimum of experimental melts of U-contaminated steel
B.2. Series of experimental melts with increasing U-contamination
B.3. Analysis of results from B.1. and B.2.; preparation of industrial-scale melts
B.4. Industrial-scale melts of approx. 100 t of U-contaminated mild steel
B.5. Final evaluation and generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

In 1993, two melts a day were carried out during nine days, in order to show the feasibility of continuous melting. In December 1993, the ingot casting area was modified to produce cast iron containers and biological protections.

Progress and results

1. Operating conditions (B.2.2.)
   In 1993, the operating conditions were the same as defined in 1992. Two tests a day were done during 9 days in order to show the feasibility of continuous melting. The main problems encountered were related to a raise of ventilation temperatures (Tab. V) and filter plugging (Tab. III).

2. Melting 4,383 tons (B.3.)
   This melt was carried out from April 27, 1992 to November 30, 1993 with the recycling of ingots, which activity was superior to 1 Bq/g, and also with the recycling of dusts. The mass of scrap was 4,383 tons before melting, from which 4,070 tons of cast iron and 197 tons of total waste were obtained (Tab. I).

3. Radiological characterization of the melt product
   Before melting, it is very easy to determine the radioactivity present (G2/G3 primary circuit pipes) and its distribution between the "*Cesium*" and "*Cobalt*" nuclides.
   A measure of gamma doses flow, outside the container containing the pre-cut pipes, is made and then the total activity is calculated. During melting, samples of some grammes are taken for radiological analysis.

4. Radioprotection
   Radioprotection is carried out by CEA's health physics specialists. During the melts realized in 1993, specific problems appeared neither for the people working in this field, nor for the buildings. Measurement results are given in Table II.

5. Reuse/release of the melt product
   The ingots (25 kg) are stored in buildings near the furnace, because the activity levels for free release have not yet been decided.
   Since December 1993, no more ingots are cast but products for the nuclear industry, such as containers for radwaste transport and storage and biological protections.

TABLE I - Melting results

Melting with recycling	263
Number of melts	322
Materials to be molten	4,070 t
Graphite	214 t
Ferrosilicon	99 t
Melting	3,913 t
Ingots	2,851 t (4.6 Bq/g)
Blocks	1,062 t (9.11 Bq/g)
Remaining dust	26 t
Slags	171 t
Electric energy	8,225 kWh per melt

TABLE II - Working conditions

Atmospheric contamination	Melting zone: 0.01 to 6.4 Bq/m³ maxi
Surface contamination	Melting zone: 2 to 80 Bq/cm² Preparation zone: 1 to 160 Bq/cm²
Surrounding flow rate	< 10 to 20 MGy/h

TABLE III - Filters HE/THE plugging (mini/maxi in kPa)

	Two melts a day	Simple melt (1992)	Maxi limits
HE-NORTH	0.94 / 1.67	0.65 / 0.95	< 2.8 kPa
HE-SOUTH	1.09 / 1.65	0.87 / 1.22	< 2.8 kPa
THE-NORTH	0 / 0.25	0 / 0.18	> 2 kPa
THE-SOUTH	0 / 0.28	0 / 0.21	< 2 kPa

TABLE IV - Ventilation temperatures (mean in °C)

	Double melts	Simple melts (1992)
Hood	49	50
Water Jacket	238	195
Jet line	93	83
Cooler	49	47
Ingot casting	87	66
Filters	60	55

<u>Reference</u>:

- Intermediary reports Nos. 3 and 4 of the FI2D-0034 research contract
- Annual results of the 722/9/I/O/353 exploitation (April 1992)
- Annual results of the 722/9/I/O/377 exploitation (April 1993)

8.12 MELTING OF ALPHA-CONTAMINATED STEEL SCRAP AT INDUSTRIAL SCALE

Contractors: Siemens-KWU, SG
Contract No.: FI2D-0038
Work Period: October 1990 - April 1994
Coordinator: K H GRÄBENER, Siemens-KWU
Phone: 49/69/807 36 45 Fax: 49/69/807 47 19

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.

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. The specific contamination 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.

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 executed in close co-operation with 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

B.1. to B.5. had already been completed. B.6. terminated with the melting of 42 Mg of scrap in the second campaign. B.7., comprising the melting of steel scrap contaminated with thorium, could not be performed because no scrap was available. For B.8., the distribution of alpha nuclides in the walls of the crucible was estimated. Supplementary investigations have been proposed to the EC to improve the melting technique as well as the melt products.

Progress and results

1. Main melting programme (B.6.)

The melting tests of the first campaign of industrial-scale tests have shown that in the case of an initial specific U-235 activity of 0.12 Bq/g so much of the activity becomes concentrated in the slag that the limit of 2.4 Bq/g is reached there (this corresponds to 3 g/100 kg). Since, according to the definitions given in the German Radiological Protection Ordinance, such slag is no longer to be considered as being waste that contains nuclear fuel but rather as waste in the form of nuclear fuel, depleted uranium dioxide was added to the remaining 42 Mg of the second industrial-scale melting campaign. Addition of the depleted uranium dioxide resulted in most cases in the ratio of the mass of the isotope U-235 to the mass of the isotope U-238 in the slag being smaller than the ratio of these two radionuclides which arises in nature, which means that the slag can be handled as waste in the form of so-called "other radioactive substances".

Just as in the first campaign, the measurements revealed that no release of the nuclides occurs. The uranium is mainly retained in the slag. Eroded particles are drawn directly into the filters, even when the offgas path changes. Furthermore, it is possible with a feed system of this kind to ensure that only small quantities of radionuclides are entrained into the exhaust system and that there is no significant concentration of activity in the dusts.

These results prove the basic feasibility of returning the uranium to its natural condition in order to ensure safe handling and to fix the enriched uranium in the slag. The problems encountered in this connection in a related test programme were able to be solved. Under the conditions employed in these tests, radiation exposure of personnel as well as radiological impact on the environment can be ruled out.

The outputs and activities of the second alpha melting campaign are presented in Tables I and II.

2. Two large-scale melting campaigns using plutonium- and thorium-contaminated steel (B.7.)

No contaminated steel was available. It is therefore proposed that laboratory experiments be conducted in 1994 using artificially contaminated steel. These experiments would serve to provide information on the behaviour of plutonium and thorium.

3. Determination of alpha distribution in crucible material (B.8.)

Three samples removed from the wall of a crucible were analysed to determine their content of alpha-emitting nuclides. The crucible had been used for a melting campaign at Siempelkamp. The results revealed that only the layer of crucible material which had been in direct contact with the melt was contaminated (specific activity = 26 α Bq/g). The layers beneath this were only slightly contaminated (second layer = 5 α Bq/g and next layer = 0.5 α Bq/g).

4. Generation of specific data (B.9.)

The results showed that nuclide-specific measurements of extremely low plutonium contents in molten steel are very difficult to perform. The methods of analysis must be improved.

In earlier progress reports, detailed information was provided with regard to activity distribution, working time and personnel doses. At the present time, it is not possible to make any statements regarding the cost-benefit relationship of decontamination by melting due to a lack of sufficient experience.

5. Proposal to improve melting technique and melt products

The objectives of the supplementary research work are additional laboratory melting experiments to improve the decontamination factors as well as radiochemical analyses to obtain further information on the behaviour of Pu, Th and U isotopes during melting. This will be completed by large-scale melting tests of Pu-contaminated scrap as well as remelting experiments and evaluations of the melts from the 1993 campaign.

Table 1

Outputs and Activities of Second Alpha Melting Campaign

Melting day	AB Nr.:	Casting weight kg Sum	Results of Gamma-Measurements U-235 Bq/g	U-235 kBq	U-238 Bq/g	U-238 kBq	Mass Determination in g per g M U-235	M U-238	Mass	% Ratio % U-235	% U-238	Enrichment Depletion f	Total Activity kBq	Spez. Activity Bq/g
	Single Weights in kg		A U-238 kBq	A Th-234 kBq	A Pa-234m kBq	A U-234 kBq	A U-235 kBq	A Th-231 kBq	other FP kBq	other < 1 % kBq	Co-60 kBq	Cs-137 kBq		
01.03.1993	930109-111	2820	0,0029	8,178	0,089	250,98	3,6E-08	7,2E-06	7,22E-06	0,50413	99,4959	0,70	1020,36	0,36
	1020/1015/785		250,98	250,98	250,98	240,96	8,178	8,178	0	10,10	0	0,00		
02.03.1993	930112-114	2770	0,064	177,28	0,31	858,70	8E-07	2,5E-05	2,58E-05	3,11047	96,8895	4,32	8565,00	3,09
	1010/1005/755		858,7	858,7	858,7	5223,38	177,28	177,28	326,16	84,80	0	0,00		
05.03.1993	930115-118	2890	0,03	86,7	0,82	2369,80	3,8E-07	6,6E-05	6,65E-05	0,56568	99,4343	0,79	10333,13	3,58
	515/505/1025/845		2369,8	2369,8	2369,8	2554,53	86,7	86,7	393,49	102,31	0	0,00		
08.03.1993	Bl.v.01.03. melted 930119-122	2920	0,0023	6,716	0,095	277,40	2,9E-08	7,7E-06	7,69E-06	0,37506	99,6249	0,52	1053,95	0,36
	1085/1060/490/285		277,4	277,4	277,4	197,88	6,716	6,716	0	10,44	0	0,00		
11.03.1993	930123-127	2680	0,0085	22,78	0,255	683,40	1,1E-07	2,1E-05	2,07E-05	0,51566	99,4843	0,72	2906,40	1,08
	505/525/510/580/560		683,4	683,4	683,4	671,19	22,78	22,78	110,68	28,78	0	0,00		
12.03.1993	930128-130	2865	0,03	85,95	0,4	1146,00	3,8E-07	3,2E-05	3,27E-05	1,15281	98,8472	1,60	6451,90	2,25
	1130/1165/570		1146	1146	1146	2532,43	85,95	85,95	245,69	63,88	0	0,00		
15.03.1993	930131-133	2675	0,012	32,1	0,25	668,75	1,5E-07	2E-05	2,03E-05	0,74087	99,2591	1,03	3168,26	1,18
	1035/905/735		668,75	668,75	668,75	945,79	32,1	32,1	120,65	31,37	0	0,00		
16.03.1993	930134-136	2975	0,002	5,95	0,086	255,85	2,5E-08	6,9E-06	6,96E-06	0,36032	99,6397	0,50	964,31	0,32
	990/1000/985		255,85	255,85	255,85	175,31	5,95	5,95	0	9,55	0	0,00		

Table 2

Outputs and Activities of Second Alpha Melting Campaign

Melting day	AB Nr.:	Casting weight in kg Sum	Results of Gamma-Measurements U-235 Bq/g	U-235 kBq	U-238 Bq/g	U-238 kBq	Mass Determination M U-235	M U-238 in g per g	Mass	% Ratio % U-235	% U-238	Enrichment Depletion f	Total Activity kBq	Spez. Activity Bq/g
	Single Weights in kg		A U-238 kBq	A Th-234 kBq	A Pa-234m kBq	A U-234 kBq	A U-235 kBq	A Th-231 kBq	sonst. FP kBq	other < 1 % kBq	Co-60 kBq	Cs-137 kBq		
17.03.1993	930137-139	2775	0,0074	20,535	0,276	765,9	9,3E-08	2,2E-05	2,24E-05	0,41519	99,5848	0,58	2973,25	1,07
	945/955/875		765,9	765,9	765,9	605,04	20,535	20,535	0	29,44	0	0,00		
18.03.1993	930140-142	2760	0,0087	24,012	0,336	927,36	1,1E-07	2,7E-05	2,72E-05	0,40102	99,599	0,56	3572,97	1,29
	975/975/810		927,36	927,36	927,36	707,49	24,012	24,012	0	35,38	0	0,00		
19.03.1993	930143-145	2700	0,008	21,6	0,288	777,6	1E-07	2,3E-05	2,33E-05	0,43009	99,5699	0,60	3042,55	1,13
	950/1020/730		777,6	777,6	777,6	636,42	21,6	21,6	0	30,12	0	0,00		
22.03.1993	930146-148	2530	0,05	126,5	0,7	1771	6,3E-07	5,6E-05	5,71E-05	1,09851	98,9015	1,53	9761,57	3,86
	940/975/615		1771	1771	1771	3727,20	126,5	126,5	371,73	96,65	0	0,00		
23.03.1993	930149-151	3000	0,003	9	0,14	420	3,8E-08	1,1E-05	1,13E-05	0,33211	99,6679	0,46	1558,61	0,52
	1060/1055/885		420	420	420	265,18	9	9	0	15,43	0	0,00		
24.03.1993	930152-154	2770	0,004	11,08	0,1335	369,795	5E-08	1,1E-05	1,08E-05	0,46376	99,5362	0,64	1472,59	0,53
	945/960/865		369,795	369,795	369,795	326,46	11,08	11,08	0	14,58	0	0,00		
25.03.1993	930155-157	2555	0,005	12,775	0,167	426,685	6,3E-08	1,3E-05	1,35E-05	0,46341	99,5366	0,64	1698,83	0,66
	1075/920/560		426,685	426,685	426,685	376,40	12,775	12,775	0	16,82	0	0,00		
26.03.1993	930158-160	2675	0,00237	6,33975	0,087	232,725	3E-08	7E-06	7,05E-06	0,42182	99,5782	0,59	906,63	0,34
	920/955/800		232,725	232,725	232,725	186,79	6,34	6,34	0	8,98	0	0,00		
	930161-163	2408	0,007	16,856	0,237	570,696	8,8E-08	1,9E-05	1,92E-05	0,45718	99,5428	0,63	2264,87	0,94
29.03.1993	808/830/770		570,696	570,696	570,696	496,65	16,856	16,856	0	22,42	0	0,00		

8.13. DEMONSTRATION OF EXPLOSIVE DISMANTLING TECHNIQUES OF THE BIOLOGICAL SHIELD OF THE NIEDERAICHBACH NUCLEAR POWER PLANT (KKN)

Contractors: BE, Noell, Siemens-KWU
Contract No.: FI2D-0046
Work Period: November 1990 - June 1994
Coordinator: U FREUND, BE
Phone: 49/6196/936 419 Fax: 49/6196/936 499 or 199

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. Assessment of structural safety **(BE, Noell)**

B.4.1. Modelling of shield response to the blast transient loading (BE)

B.4.2. Modelling of building response by simple models and comparison to pre-evaluations at selected safety-relevant locations (BE)

B.4.3. Safety control for compliance with limiting values by test accompanying measurements (BE, Noell)

B.5. Final assessment of the blasting procedure **(BE, Noell)**

B.5.1. Technical feasibility and reliability (BE, Noell)

B.5.2. Compliance with safety regulations concerning radiation protection, radioactivity release, contamination/decontamination and structural safety (BE, Noell)

B.5.3. Comparison with other concrete dismantling techniques, such as sawing by diamond or wire saw, core drilling, possibly combined with sawing, high pressure water jet with abrasives (Noell, BE)

B.5.4. Setting up of guidelines and rules for general application of the bore hole blasting technique to other shield structure, and of cost estimates (BE, Noell)

B.6. Related investigations of general applicability **to various types of nuclear power plants (BE, Siemens-KWU, SSP).**

B.6.1. Building response by advanced modelling for the reactor building (BE, SSP)

B.6.2. Local damage, prediction of cracks and material failure (BE, Siemens)

B.6.3. Blast loading limits with regard to the integrity of light structures in close vicinity to the charge location (BE)

B.7. Generation of specific data **on costs, radioactive job doses, working time and secondary waste arisings, derived from the execution of items B.2. and B.6.**

C. Progress of work and results obtained

Summary of main issues

On-site preparations were concluded in April 1993. The explosive method application was started in May 1993. The predesigned test areas were dismantled by bore hole blasting in the period till September 1993. These test areas lie in three different locations of the biological shield column. The concrete is of B 45 quality grade and is heavily reinforced. The mean specific activity ranges from 0.04 Bq/g to 3 Bq/g. The free release limit is 0.37 Bq/g. The volume to be removed per blast ranges from about 1 to 4 m^3. The total mass of the explosive charges detonated per blast was between 0.6 kg and 3 kg. The charges were of different type of commercial explosives. The blast resulted in ejecting part of the concrete and shattering the remaining material in situ. The blast was contained inside the protection tent. Removal of the concrete was performed by hydraulic ram. The steel bars were cut by thermal lance. The packaging and radiological control were performed along with the ongoing commercial shield dismantling. Test data on air blast and building excitation were recorded. Dynamic loading stayed below critical limits.

Progress and results

1. Introduction

Following the detailed planning and licensing which was concluded in 1992 a subcontractor (Schöndorfer) was commissioned at the beginning of 1993 to perform the dismantling operations of the biological shield.

The subcontractor was responsible for supplying skilled labour and the tools necessary for dismantling. NOELL conducted the infrastructure preparations e. g. construction of the adjustable working platform, the stationary and mobile ventilation system and the dust retention tent.

All supplies and purchasings were performed by April 1993, so that dismantling operations could be started by the end of April.

The subcontractors on-site management was integrated into the main contractors (NOELL/NIS) dismantling organisation.

The contractual tests (EC part) were integrated into the global planning so that licensing and approval were achieved together with the commercial dismantling actions.

2. Final licensing and approval (B.5.2.)

The licensing papers had listed 21 approval specifications. Most of them concerned general construction site operations. Some were specific for the explosive shield dismantling:

- the work has to be performed by a licensed dismantling company.
- existing law and regulations for performing commercial blasting operations have to be obeyed. This was to be approved by a legally authorized expert.
- on-site storage room for limited explosives supply was to be operated under specific safety precautions.
- detailed description of the blast sequences including the local constructional conditions had to be submitted at least one week in advance for approval by the authorities before each blasting round.

- detailed reference of safety measures concerning the integrity of the ventilation system, specifically the filter units had to be given.

All requirements were fulfilled accordingly before starting of the blasting or during operation, respectively.

3. Site preparation of tools and measuring devices (B.2.2., B.2.2.)

The necessary hardware installations were carried out by NOELL in accordance with the regulations. In particular the adjustable working platform and the dust retention tent were constructed. The latter could be opened on top for heavy tool access by crane, in particular for the hydraulic ram. Personnel access to the control area was made through lock-doors. All openings through the shield wall were closed by steel plates and foil cover.

The control area preparations also included mounting of a ventilation pipe with connections for flexible ducts leading to the blast area. Personnel of the subcontractor was instructed about tool handling according to the regulations. The installation of blast wave pressure and structural vibration sensors was executed according to a measurement plan.

The working steps were laid down in detailed instruction sheets.

In Table I a listing is given of consecutive working steps related to the blasting tests.

4. Dismantling of the test sections (B.2.3.)

The test sections were dismantled in close cooperation with site personnel including the subcontractors specialists.

The locations of the 3 test sections can be seen in Figure 1.

4.1 Description of the test sections and blasts

Test section 1 had the shape of a ring segment with outer diameter of 10.1 m and inner diameter of 8.6 m and height 950 mm. The area to be blasted was 2.8 m by 1.1 m with a thickness of 0.8 m. The volume of the concrete to be removed was 2.9 m^3. The steel reinforcement was calculated from the construction drawings to be about 11 weight-%. In reality deviations occurred towards larger values. The concrete was of B45 quality with an average density of 2 g/cm^3. A total amount of 1.08 kg of explosive was detonated with a twin blast of 0.54 kg each. The concrete activation was below the free release limit of 0.37 Bq/g.

Test section 2 was located in the cylindrical wall at the same height as the reactor core, thus exhibiting the highest concrete activation level (about 3 Bq/g). Even so, it was low enough to allow manual operation of all working steps.

The heavily reinforced baryte concrete was composed of cement plus baryte aggregate yielding a density of 3.6 g/cm^3. Total amounts of explosive of 1.8 and 3.3 kg were detonated at 2 different positions, respectively. The explosive charges were distributed in a maximum of 24 bore holes.

Test section 3 was located around the D_2O-pipe feed through hole representing a very complex steel reinforcement pattern. The bore hole pattern was adjusted to these locally varying conditions. The total amount of explosive was 0.6 kg.

4.2 Supplementary work

Following a blasting round, the removal of the fractured and cracked concrete and removal of the reinforcement steel were performed. The concrete was removed by the hydraulic ram which was brought into the blast area by crane. The task of this ram was to set free the shattered concrete and to bend the reinforcement steel for easy cutting. The steel cutting was done by thermal lance to adequate size for container storage.

Packaging into containers occurred manually.

4.3 Blast loading

The building vibration was recorded at selected locations (reactor platform and foundation). The air blast was recorded at several positions inside and outside the confined blast area. The maximum pressures are below 20 millibars outside the confined blast area. The building vibration levels were at least one order of magnitude below the recommended values of the German standard DIN 4150.

5. Radiological conditions (B.5.2.)

From the accompanying dose rate measurements an average local dose rate value of less than 1 μSv/h was obtained. Due to the very low local dose rate and the low specific activity no radiation protection besides face filter mask was required for working personnel. The collective dose (external radiation only) was not further considered.

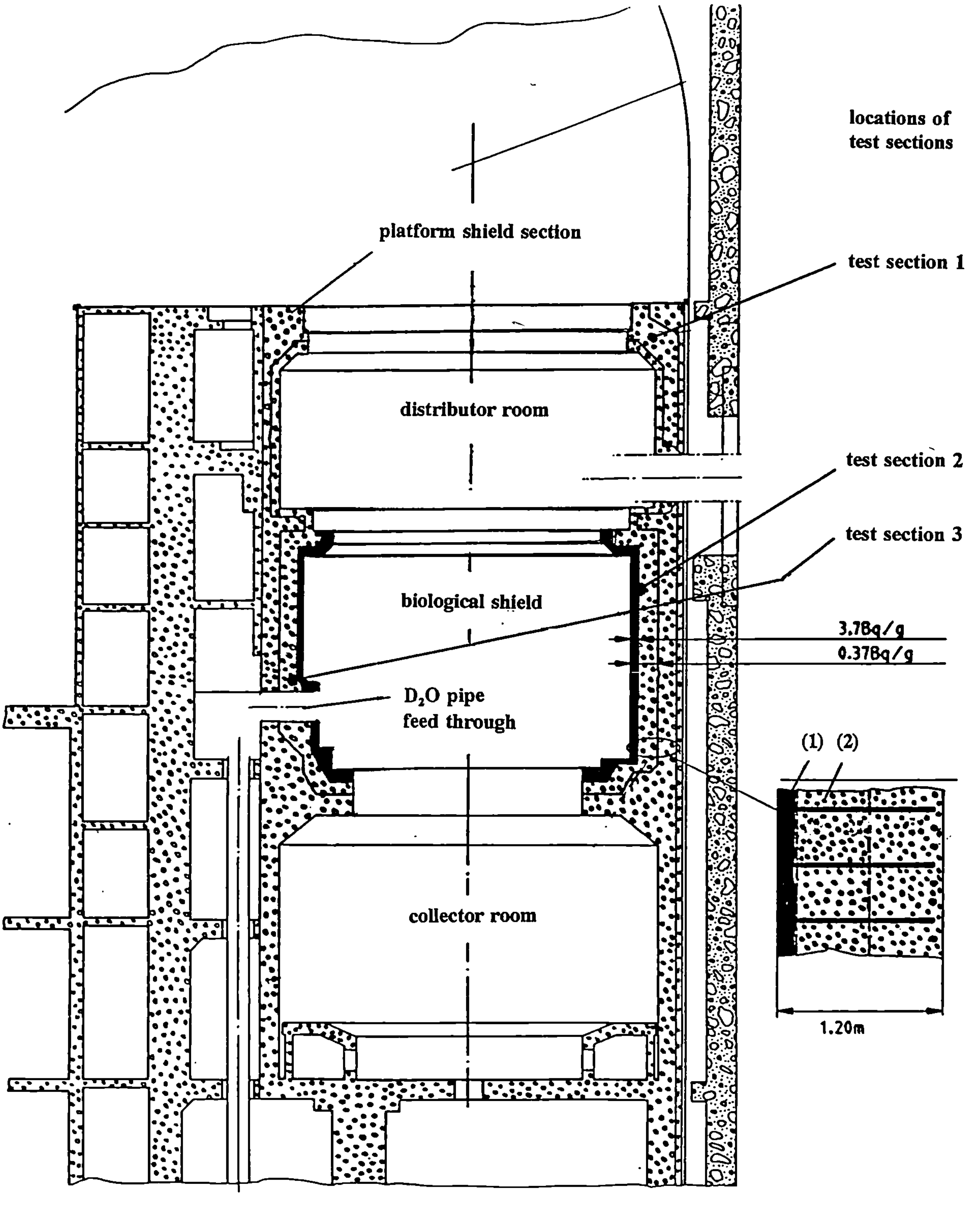
6. Preliminary conclusions (B.5.)

The explosive method has proven feasible and competitive to hydraulic tools. Dynamic loading was kept below damage levels with respect to air blast, vibrations were far below regulatory limits.

Due to the restrictions associated with explosives handling some time-consuming working steps cannot be avoided where no parallel work is permitted inside the reactor containment.

Table I: Working steps for the concrete blasting tests

Step No.	Tasks
1	Drilling of charge holes, core drilling, full volume drilling
2	Loading of charge cartridges
3	Plugging of charge holes
4	Wiring of electrical initiation cord
5	Test of sensor arrays, data acquisition (parallel to step 6 ff)
6	Covering of blast area by mats
7	Blast protection of ventilation duct and switch off ventilation
8	Two step clearing of personnel from the reactor containment
9	Explosive charge initiation and trigger of data acquisition
10	Control of blast result and possible damage
11	Radiological control
12	Return of personnel into reactor containment



(1) activated material, limit for free release 3.7 Bq/g

(2) additional activated material after reduction of limit for free release to 0.37 Bq/g

Figure 1: Concrete to be removed from the biological shield

8.14. DECOMMISSIONING OF THE B205 FUEL REPROCESSING PILOT PLANT

Contractors:	BNF plc Sellafield
Contracts Nos.:	FI2D-0050
Work Period:	May 1991 - June 1994
Coordinator:	D GAMBERINI
Phone:	44-9467-73667 Fax: 44-9467-27383 (74070)

A. OBJECTIVE AND SCOPE

The object of the underlying work programme is the dismantling of the B205 Fuel Reprocessing Pilot Plant. The plant was operated from 1957 to 1965. It processed uranium metal, totalling several hundred kilograms of fuel (typically 4,000 MWD/te). The product purification, solvent wash and sampling facilities were dismantled in the early 1970's leaving the MA cell nearly empty and the suite of HA cells untouched.

The aim of the project is to remove all of the contaminated structures from the laboratory (including the plinths that the cells are standing upon). It is ultimately intended to reuse the laboratory. This project will be used as a development project to demonstrate the techniques for dismantling this type of facility.

The contract covers dismantling work of the New Dissolver Cell, of the original HA Cell and of the Metal Cutting Cell. Information on the dismantling of the MA Cell (working period: August 1991 - March 1992) and of the New Primary Separation Cell (working period: July 1994 - September 1995) will be made available to the CEC.

This decommissioning project provides scope for testing a full range of techniques - for visual/radiometric inspection; remote handling; containment and shielding; decontamination of stainless steel, lead, concrete/brickwork; waste categorisation, segregation, monitoring and size reduction. Residual metal fuel, fuel cladding and historic dissolver liquor spillages provide authentic decontamination problems.

Estimated mean dose rates vary between < 10^{-2} mSv/h for the laboratory and 200 mSv/h for the cutting cell with hot spots in the latter area of up to 470 mSv/h.

Lessons to be learnt include operational effectiveness, reliability, "user-friendliness", secondary waste arisings, manpower needs, dose-uptake, etc. The data sought is fundamental to the evaluation of future large-scale decommissioning projects and invaluable feedback into technique development programmes. It is planned to effect industrial-scale evaluation of decommissioning techniques, thereby providing data to assist planning, cost estimation and implementation of subsequent major projects.

The work programme will complement, and involve co-operation with the parallel Danish project at Risø National Laboratory.

B. WORK PROGRAMME

B.1. Preparatory work including assessment and/or backfitting and installation of auxiliary equipment and access routes.

B.1.1. Removal and decontamination of redundant service lines, shielded liquor transfer line, fume hoods, internal wall etc.
B.1.2. Refurbishment of the cell ventilation/filtration system,
B.1.3. Establishment of waste decontamination facilities,
B.1.4. Establishment of waste handling/export facilities for LLW, ILW and PCM waste,
B.1.5. Installation of new lifting beams to support manipulators and of other new equipment where required.

B.2. Dismantling of the New Dissolver Cell

B.2.1. Installation and commissioning of the manipulator

B.2.2. Removal of all supplementary shielding as far as possible and construction of a modular containment for working area and manipulator maintenance area.

B.2.3. Removal of concrete panels and dismantling of the inner stainless steel skin by using the manipulator

B.2.4. Removal of all process plant equipment and the remainder of the cladding,

B.2.5. Clean-up/scabbling of all inside faces of the cell using the manipulator,

B.2.6. Removal of the remaining structure.

B.3. Dismantling of the Original High Active Cell

B.3.1. Installation and commissioning of the manipulator

B.3.2. Removal of all supplementary shielding as far as possible and construction of reusable modular containment, backed by lead brick as necessary, to form a working area and manipulator

B.3.3. Installation of waste handling arrangements,

B.3.4. Use of the manipulator to breach the lead brick wall and gain access,

B.3.5. Retrieval of the existing hoist from the cell to the maintenance area to be removed manually,

B.3.6. Dismantling and removal of all process plant using the manipulator,

B.3.7. Clean-up/scabbling of inside faces using the manipulator,

B.3.8. Removal of remaining walls and plinth.

B.4. Dismantling of the Metal Cutting Cell

B.5. Generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED.

Summary of Main Issues

This annual progress report summaries the work carried out from January 1st to December 31st 1993. The work carried out during this period covers work packages B1 Preparatory Work, and B.2.1 Installation and Commissioning of the Manipulator.

Decontamination Trials using Decoha and DSS 30 have been completed.

Clearance of all redundant equipment and modifications to the building structure (i.e. walls and ceilings) have been completed to allow for installation of the remote decommissioning facility.

The remote decommissioning facility has been constructed and commissioned at the manufacturer's works.

In addition to the above scheduled work a number of unanticipated discoveries have imposed additional scopes of work which has necessitated a re-scheduling of the project.

The dose uptake from one unanticipated scope of work lasting 4 weeks accounted for 48% of the project team's yearly total, dose uptakes from other activities have been within predicted values.

A schematic diagram of the facility is shown in Figure 1.

Progress and Results.

1. B.1.1 Removal and decontamination of redundant service lines, shielded liquor transfer line, fume hoods.

The floor and wall areas adjacent to the New Dissolver Cell, Metal Cutting Cell and Highly Active Cell have been cleared of all redundant equipment.

Decontamination trials on beta/gamma contaminated concrete have been undertaken on part of the floor in Lab 190B using Decoha and DSS 30 liquid decontamination agents. The project team have concluded that the process is useful for the removal of shallow contamination from concrete. However prolonged use of the process over high radiation areas can expose operators to high dose uptake.

B.1.1 is now completed.

2. B1.2. Refurbishment of the cell ventilation/filtration system

The ventilation ducts serving the New Dissolver Cell have been re-routed to allow for the installation of the remote handling equipment.

An additional ventilation system has been designed for local extract during decommissioning operations.

B.1.2 is now completed.

3. B.1.3 Establishment of waste decontamination facilities.

An adjacent laboratory will be made available for waste decontamination operations and trials during decommissioning of the major project facilities.

4. B.1.4 Establishment of waste handling/export facilities.

The waste handling/export facilities have been constructed and tested at the manufacturer's site. The facilities are being prepared for shipment to Sellafield.

A shielded storage compartment in Primary Separation 1 has had its active contents removed and the compartment has been demolished. This will allow the waste export track to be laid.

5. B.1.5 Installation of new lifting beams

Inspection of the installed lifting beams has indicated that this work package will not be required.

6. Unprogrammed preparatory work

The most significant unprogrammed item was the discovery that a load bearing wall had been removed from lab 190A by previous operators of the facility. This has had repercussions on the project timescale due to the requirement to design and manufacture a suitable steel beam that will reinstate the strength in the ceiling. The beam has been manufactured and installation work has begun.

During an attempt to manoeuvre a ceiling-mounted trolley into the Highly Active Cell the existing lead brick wall became unstable and had to be partially rebuilt. A brick wall was uncovered that was not shown in the plant record drawings. A special brick removal tool has had to be designed and built for use with the remote manipulator. This tool is currently undergoing trials.

Extensive contamination of the concrete floor in lab 190B was discovered during a decontamination exercise. Lead shielding plates were found under 50mm of concrete, these were removed and a dose rate of 50mSv/hr beta/gamma measured. A diamond tipped core drill was used to remove the contaminated concrete to a depth of 500mm in some areas. After removal of contaminated concrete the resulting hole was backfilled with lead loaded concrete. The dose uptake to the team from these decontamination operations over 4 weeks was 1191µSv, this is 48% of the yearly collective dose (Table I).

7. B.2.1 Installation and Commissioning of manipulator

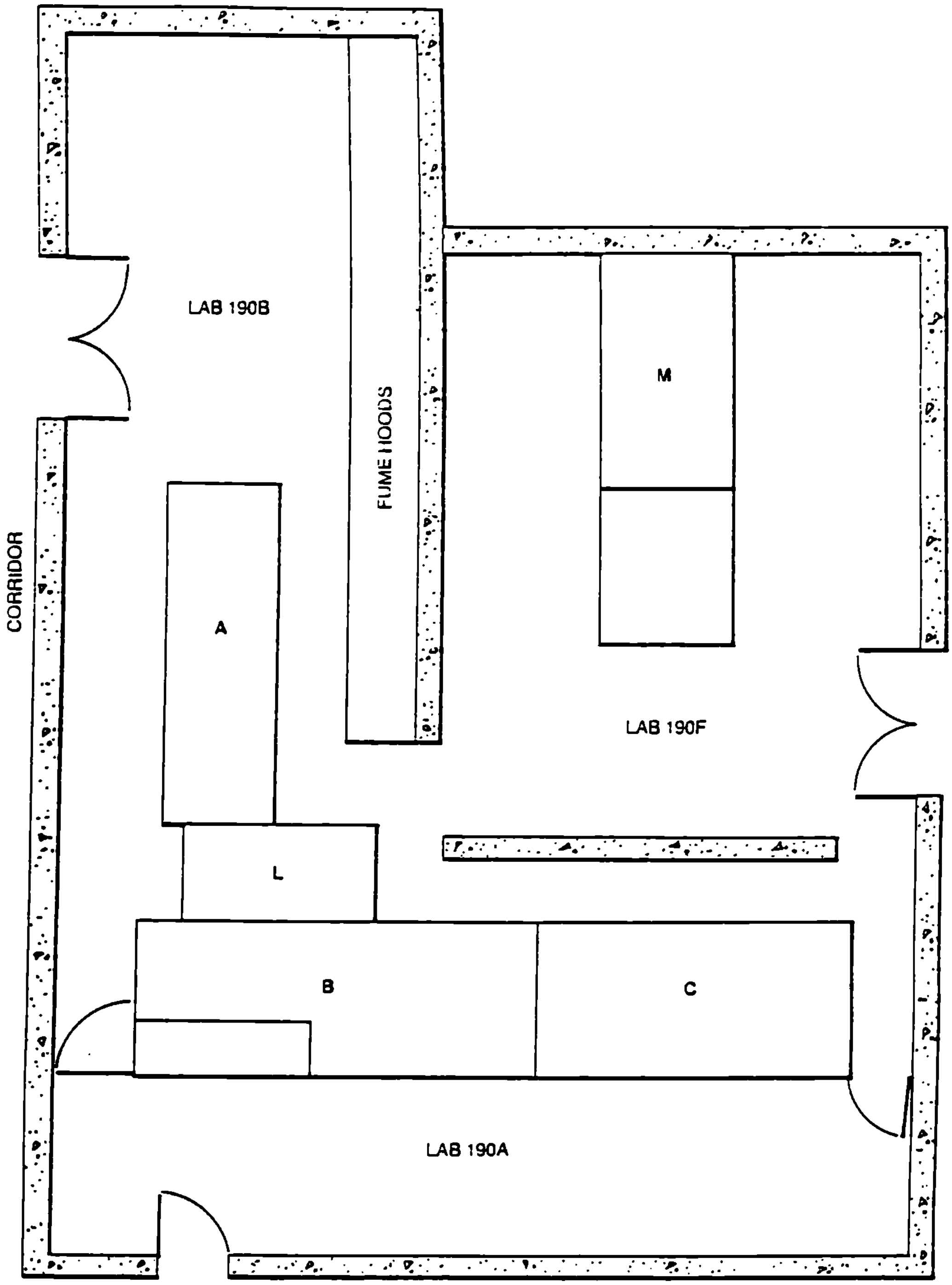
The manipulator, control system, transport trolley, waste handling and export facility have all been commissioned and tested at the manufacturer's site. Due to the unprogrammed preparatory works the transport of the equipment to Sellafield has been delayed. Therefore training and further tool development trials are being undertaken so as to optimise the control systems.

Table I: Dose Uptake to Project Team

Period	Collective Dose Uptake
1/1/93 - 30/6/93	2053µSv
1/7/93 - 31/12/93	438µSv

Table II: Waste Arisings

Period	Low Level Waste Arisings
1/1/93 - 30/6/93	$17.7m^3$
1/7/93 - 31/12/93	25.4m3



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: September 1991 - April 1994
Coordinator: C C FLEISCHER
Phone: 44-81-575 45 82 Fax: 44-81-575 40 44

A. OBJECTIVE AND SCOPE

The aim of the project is to demonstrate/investigate the efficiencies of the explosive cutting and microwave techniques developed under preceding CEC programmes, on the full scale biological shield structure of the decommissioned LIDO enriched uranium thermal swimming pool reactor based at Harwell in U.K.

The reactor was operated from 1956 to 1972 and was used for shielding and nuclear physics experiments. The reactor core was made up from uranium/aluminium plates clad in aluminium. The core was movable through the water into any position on the centre plane of three large aluminium windows or beam holes for heavy shielding experiments to be set up outside. Initial analysis of the LIDO biological shield indicates surface dose rates of up to 10 $\mu Sv.h^{-1}$ and contamination levels of up to 50 $Bq.g^{-1}$. It is anticipated at this stage, that the main source of activity is Eu-152, Co-60 and Cs-137, but further radiological assessment will be carried out during the programme. It is proposed that up to 10 m^2 of surface will be removed by the dismantling techniques under consideration.

The present work will concentrate on the use of the two dismantling techniques to achieve concrete removal to meet both decontamination and dismantling requirements. This will offer direct comparisons between the techniques and lead to the assessment of their respective economic efficiencies. Effort will be directed at regulatory and safety aspects involved in the large-scale application of the techniques. Practical problems associated with the full-scale use of the techniques will be identified and realistic solutions obtained.

This work programme will be harmonised and carried out in close technical collaboration with work being co-ordinated by Battelle Institute on the application of explosive techniques for the decommissioning of the biological shield of the HWR KKN at Niederaichbach in Germany and the work being carried out by ENEA-CRE on the application of the microwave technique. A separate programme is also being carried out by KfK, aimed at developing remote manipulator deployment systems for the microwave unit.

B. WORK PROGRAMME

B.1. Preparatory work (AEA, Taywood)

B.1.1. Establishment of operational control and safety documentation (AEA and Taywood)
B.1.2. Preparation of documentation for the control of operations on the LIDO structure (AEA)
B.1.3. Development and construction of a sealed sub-containment on the biological shield including associated ventilation and filtration systems (AEA)
B.1.4. Pre-test structural assessment (Taywood)
B.1.5. Activity assessment and coring for material samples (AEA and Taywood)

B.2. Large-scale technique demonstrations on non-active zones (AEA, Taywood)

B.2.1. Implementation of explosive techniques (Taywood)
B.2.2. Implementation of Microwave technique (AEA)

B.3. <u>Large-scale technique demonstrations on active zones</u> (AEA, Taywood)
B.3.1. Implementation of explosive techniques (Taywood)
B.3.2. Implementation of microwave techniques (AEA)
B.3.3. Recontamination assessments (AEA, Taywood)
B.4. <u>Aerosol characterisation</u> (AEA)
B.5. <u>Structural assessment</u> (Taywood)
B.6. <u>Theoretical assessment</u> (Taywood)
B.7. <u>Techniques assessment and conclusions</u> (AEA and Taywood)
B.8. <u>Collaboration with related programmes</u>
B.9. <u>Generation of specific data</u>

C. Progress of work and results obtained
Summary of main issues

Following commissioning tests on the ventilation system, the microwave tool and the containment structure against blast loadings, the programmed concrete removal from inactive and active sections of the LIDO biological shield was started. Very small quantities of explosives have been used to strip-off layers of concrete, in a controlled sequence, in both inactive and active sections of the biological shield. Concrete removal has been achieved in a series of layered strips by careful designing of explosive charge locations. The concrete removal has been achieved without causing any damage to sensitive equipment and parts of the containment structure and with blast pressure loads kept to a minimum. Throughout the series of trials, filter efficiency within the ventilation system was not found to have been impaired by either the blast pressure loads or the dust levels generated by the dismantling technique. Trials carried out on the biological shield walls using the microwave tool have indicated that the concrete may be too dry for the efficient use of the technique for the removal of concrete.

Progress and results
1. Development and construction of a sealed sub-containment (B.1.3)

The system is based on a mobile filtration unit (MFU) having two stages of HEPA filtration and two independent fans (see Figure 1). Glass fibre panels were used to enclose the top of the bioshield and in the fabrication of the man-entry system. Entrance to the bioshield is through a labyrinth formed by two man-entry ports (see Figure 1). A door, comprising of a high efficiency panel filters, is provided across the inner access port, which is closed during concrete removal. The extract system was required to maintain an average air velocity of greater than $1ms^{-1}$ through the outer man access port to the bioshield. To confirm this, commissioning trials were carried out during which a series of air velocity measurements were made across the outer man access port with the panel filter door both open and closed. Measurement made confirmed that the average air velocity remains $>1ms^{-1}$ with a single fan stage operating.

2. Large scale technique demonstration on non-active zones (B.2)

2.1 Implementation of explosive techniques (B.2.1)

As part of the safety case for the use of explosives as a technique for the trials on the LIDO bioshield, there was a requirement to commission the new containment structure built over the LIDO tank against blast loadings. From the commissioning trials, (see Figure 2) a conservative explosive charge weight of 24 grams was selected as a safe amount of explosive to be fired at one time without imposing high deflections on the glass fibre roof.

During the containment structure commissioning trials, deflection measurements were taken of the middle point of the roof's central glass fibre panel. In addition, pressure transients in the free airspace and on the walls of the containment structure and also the accelerations experienced by the walls and roof were measured.

Concrete removal trials in non active zones of the bioshield followed on after the commissioning trials. A charge layout pattern and firing sequence was designed to achieve the removal of $3m^2$ of surface area of the LIDO bioshield. Three separate sets of firings were carried out, each set arranged to remove a complete layer of the bioshield area under test. The tests were separated to ensure that measurements could be made to assist in determining the volume of concrete removed at each stage.

It was found that the proposed volumes of concrete could be removed, using explosives, in a safe manner. By ensuring that explosive charge weights are kept to the specified limits, then the glass fibre roof can quite easily withstand the blast pressures produced.

2.2 Implementation of microwave technique (B.2.2)

The microwave tool was commissioned in-situ with the microwave head flush against the surface of the bioshield. Unit commissioning involved microwave power leakage tests and radio interference tests with the requirement that the unit must produce less than 60dB above $1\mu V/m$ at 30m from the source.

On completion of the unit commissioning, a trial was carried out during which the microwave unit was slowly ramped up to full power (25kW) and applied to the test wall for a full 20 minutes. No detonation of the concrete appeared during this period and the microwave output was de-coupled.

The immediate conclusion drawn was that the concrete was too dry to produce enough steam to crack it. Moisture measurements taken from cores extracted from the bioshield wall under test confirmed the moisture content of the bioshield to be below 2%, which is considered too low to produce detonation from a 25kW microwave generator.

Large scale technique demonstration on active zones (B.3)

3.1 Implementation of explosive techniques (B.3.1)

Concrete removal trials have been carried out on active zones of the bioshield. Charge layout pattern and firing sequence followed similar lines to those used in the non-active zones. As before, firings for the different layers concrete removal were separated, in time, to ensure that measurements could be made for volume of concrete removed. The concrete removal rate was found to be of the order of $0.007m^3/g$ of explosive which is similar to what was obtained for the non-active zone trials.

3.2 Implementation of microwave techniques (B.3.2)

It was decided that since the moisture content of the bioshield in the active zones was likely to be the same as in the non-active zones, this trial will not be carried out.

3.3 Recontamination assessments (B.3.3)

In order to permit recontamination assessments to be easily made, the sections of the bioshield walls, outside the specific zone under test, were covered by PVC sheeting. On completion of the concrete removal trials mentioned under Section 3.1, the PVC surface was monitored. Although dust was found on the PVC, no detectable activity was found. The specific activity of the concrete is very low (<50Bqg^{-1}). The small amount of fine dust that settled on the walls was not of a sufficient quantity and mass to allow detection using standard contamination monitors. The heavier particles that fell to the ground were of a mass that allowed active detection albeit very low. The only measurable activity was found on the ground immediately in front of the section of the bioshield wall under test.

4. Aerosol characterisation(B.4)

Particulate sampling was carried out on the debris produced by the explosive technique during the non-active concrete removal trials. For this an Andersen (Cascade) Impactor was used to measure aerosol size in the airspace. In addition, a number of total (gravimetric) dust filters were operated within the bioshield and the measurements made used to calculate the total aerosol concentration in the air sampled (see Figure 3). A HIAC/ROYCO particle counting system was employed to monitor aerosol levels at the inlet to the filtration plant to confirm predictions regarding safe conditions for re-entry to the bioshield area. This instrument was also used to monitor the air discharged from the HEPA filter system. Particles were not detected in the exhaust flow which suggests that the filters had not suffered gross deterioration as a result of the overpressures caused by explosive detonations. More sensitive filter tests are to be employed to fully quantify the extent of any changes in filter performance.

5. Structure inspections (B.5)

Prior to the start of the trials, the LIDO bioshield outer structure was thoroughly inspected and all existing crack details noted. Throughout the trials the existing cracks on the structure were inspected to check whether they were undergoing any changes. No changes were noted on the bioshield outer structure throughout the trials.

6. Generation of specific data

Work in this period has been concentrated on the construction and commissioning of the containment structure and ventilation system, refurbishment and commissioning of the microwave tool, non-active and active trials using explosives. Given below is a summary of current estimates for specific activities.

Title of Working Step	**Duration (Days)**	**Manpower (man hours)**
Construction of sealed containment	90	2000
Explosive technique in non-active zone	75	1200
Explosive technique in active zone	50	1000
Aerosol analyses	50	300

FIGURE 1 - MOBILE FILTRATION UNIT WITH MAN-ENTRY TO BIOSHIELD IN THE BACKGROUND

FIGURE 2 - CONTAINMENT COMMISSIONING

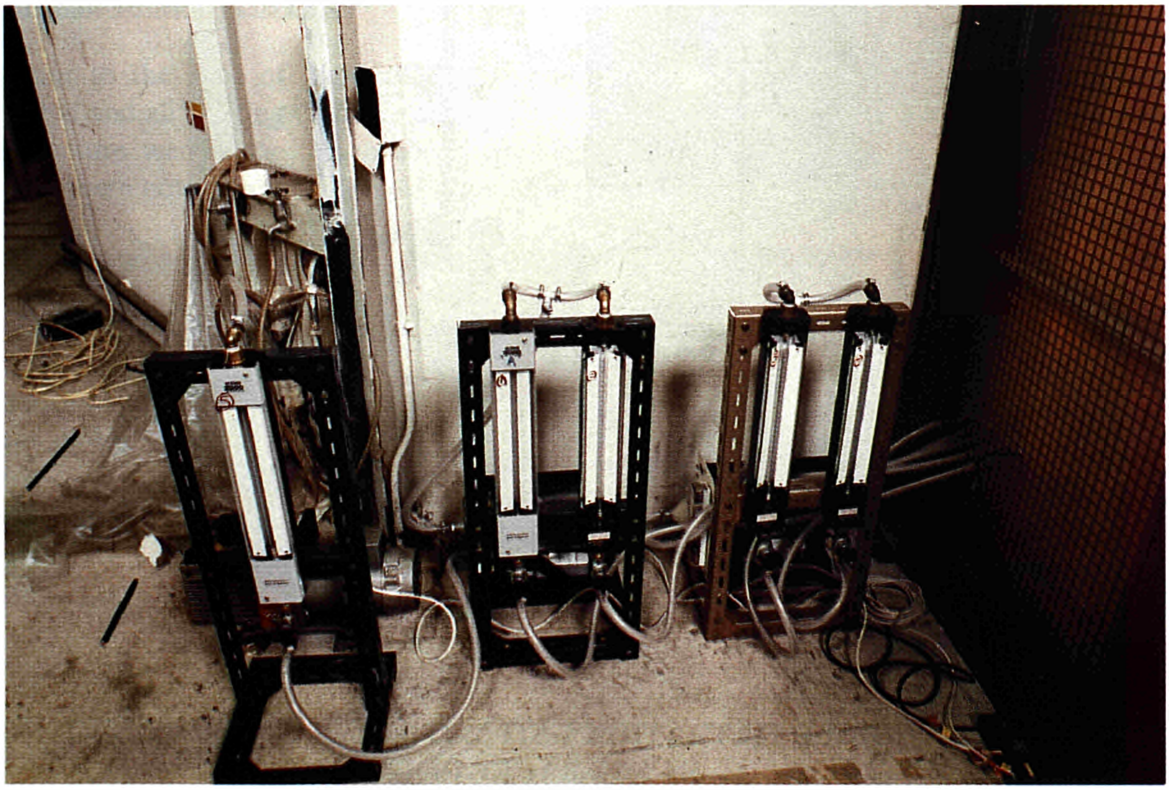


FIGURE 3 - AEROSOL ANALYSERS

8.16. FURTHER DEVELOPMENT AND OPERATION OF AN AUTOMATED LARGE-SCALE RADIOACTIVITY MEASUREMENT FACILITY FOR LOW-LEVEL DECOMMISSIONING WASTE

Contractors:	NIS, KKWR
Contract No.:	FI2D-0063
Work Period:	November 1991 - December 1992
Coordinator:	I AULER, NIS, Hanau
	Phone: 49/6181/1850 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

This project was completed in 1992. The final report is available as EUR Report No. 15444.

8.17. DEVELOPMENT, MANUFACTURING, COMMISSIONING AND TESTING OF AN AUTOMATED DEVICE FOR THE PROJECTION OF CHEMICAL GELS AND ITS APPLICATION TO G2/G3 REACTOR PIPES

Contractors: CEA-Valrhô
Contract No.: FI2D-0064
Work Period: December 1991 - June 1994
Coordinator: J R COSTES, CEA-Valrhô
Phone: 33/66 79 63 13 Fax: 33/66 79 64 22

A. OBJECTIVE AND SCOPE

This project is aimed at industrial-scale testing of an automatic machine designed to spray gel or foam compounds inside pipes with a contamination level of up to 200 Bq/g due mainly to Co-60. The envisaged decontamination performance is about 8 metric tons a week to less than 1 Bq/g; the scrap metal will then be melted down on the site for unrestricted release.

The innovative nature of this project lies in the automation of all steps of the decontamination process.

This contract will demonstrate that 450 tons of steel can be released for unrestricted use after automatic decontamination and provide a cost-effective alternative to any manual decontamination process.

The contractual work involves the internal decontamination of 450 tons of ordinary carbon steel pipes with diameters ranging from 0.5 m to 1.6 m, using an automatic decontaminant gel spraying machine. The dose rate is about 0.3 mGy/h at the pipe surface contact.
These pipes belong to the graphite-gas reactors G2 and G3, which are presently under stage 2 decommissioning.

The contractor is coordinating all decommissioning operations concerning the G2 and G3 gas-cooled reactors at Marcoule. The present state of knowledge is based primarily on CEC research contract FI1D-0003, covering small-scale manual gel spraying. The process will begin by spraying a highly alkaline gel or foam, followed by pressurized water rinsing. This will eliminate all greasy or oily deposits accumulated during 22 years of reactor operation.

In some cases, it may be necessary to follow this procedure by spraying with strongly acid gels or foams, followed by rinsing; this step may be repeated in exceptional circumstances.

B. WORK PROGRAMME

B.1. Supply and adaptation of equipment
B.2. Implementation and commissioning testing of the device
B.3. Preparatory work and preliminary active decontamination tests
B.4. Industrial decontamination of 450 tons of contaminated pipes
B.5. Analysis of results
B.6. Generation of specific data

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The first foam decontamination process tests using the SAMIT (Système d'Application de Mousses à l'Intérieur de Tuyauteries) device were conducted from February to June 93. The Télémécanique TSX 47-20 programmable controller operated without any problems using its initial programme.

No major difficulties were encountered when installing the workpiece on its cradle, covering the open ends, or positioning the chassis and spray arm. Two persons are required for these operations, and a third may be necessary to help install the front workpiece cover because of its size and weight (40 kg).

On the whole, the automatic control provisions and the sprayer itself (moving arm with alternating rotation) operate satisfactorily, although some improvements will be implemented in the next version.

Progress and results

1. Industrial decontamination (B4)

The foam spray and generation unit (Figure 1) uses standard equipment available from *Kew*, a Swedish manufacturer of high pressure spray cleaning devices. The foam injectors are capable of mixing low pressure reactants with the high-pressure (150 bar) water jet.

Our initial installation called for regulating the quantities of chemical reactants supplied to the injector by means of diaphragm metering pumps drawing fluid directly from the reagent tanks. The proper reagent proportions were obtained simply by adjusting the piston travel.

It is advisable to inject the surface -active agent before the water unit to avoid reactions with the concentrated acids or bases.

The electrically actuated valves were deleted because the internal electromechanical piston tended to jam in the presence of large quantities of viscous surface-active agent, as it is inevitable during testing.

The initial version used Kew foam injectors. It proved impossible to install two injectors in series because of the excessive pressure drop, which prevented adequate suction in the second injector. These injectors were not well suited to our process requirements: the injector was only capable of drawing 5% of the separately diluted fluid (i.e. 50-60 $l.h^{-1}$ since the initial flow rate of the Kew high pressure sprayers is 1150 $l.h^{-1}$).

This arrangement resulted in several ruptured diaphragms.

2. Process Effectiveness

All the tests were conducted with a single workpiece. It was therefore impossible to obtain a precise assessment of the action of the acid and alkaline foams on the surface contamination. The foam quality was never excellent, notably with regard to its adherence to the pipe wall. Nevertheless, two applications of weakly alkaline foam were sufficient to remove all the greasy film from the pipe.

3. Initial System Operation and Subsequent Modifications

Attempting to use standard industrial equipment for this application proved to be a mistake: it was unable to provide satisfactory operation during the initial tests and the test system was too complicated to allow rapid progress.

It became increasingly clear that spraying the foam under high pressure was also the wrong approach. The violence of the spray prevented the foam from spreading properly over the pipe surface, and significant losses were observed: the foam coalesced not at the top but at the bottom of the pipe.

An extremely simplified foam generating process was implemented at this stage in order to test a variety of foam compositions. The unit may subsequently be adapted for automatic operation according to the original SAMIT project.

Process reactants will probably be mixed manually using metering pumps and stored in a tank large enough for a single application. The tank is supported on a balance to facilitate product dosing.

This arrangement provides a number of advantages for the tests: it is simple and therefore reliable, can easily be modified and places no restrictions on preparing the process mixtures. On the other hand, the process solution must be entirely used before another is prepared (to prevent neutralization of the basic solution by the acid solution). This unit should make it possible to develop suitable foams, and may be used with the automatic version.

The problem observed with the ends of the pipe section should be eliminated by the use of a thicker foam sprayed at lower pressure with the next device.

The complete system on its carriage now resembles the following diagram. It comprises a reactant recovery pan, a foam preparation tank mounted on a balance, a compressed air spray pump and a high pressure rinsing pump.

4. Work in Progress

- Chassis and Spray Unit: the inclination of the workpiece on its cradle will be increased.
- Fluid unit: the new foam generation process will be implemented and tested.
- Controller: the controller will be reprogrammed for use with the new fluid unit and to combine separate operations. An alternating spray programme with an amplitude range of 30° to 90° will be implemented to rinse the workpiece ends on completion of the cycle. Provision will also be made to allow spraying with the arm moving forward or backward.

These are minor changes to the existing programme, with no fundamental differences.

5. Conclusion

Initial testing of the foam decontamination process using the SAMIT device between February and July 1993 showed that the spray system and automatic control system are now operational, although minor improvements will be implemented in the next version, but also showed that the reactant foam generator was not suitable for this purpose.

The initial foam generator design proved unsatisfactory, and was completely revised in July-August 1993 to simplify the unit and enhance its reliability. The redesigned unit is now nearing completion, and testing will begin again in November 1993.

The initial tests cannot be considered as a waste of time or money. The information they provided has already been implemented in redesigning a process with a very promising future. Plans for upgrading the SAMIT system are already under consideration.

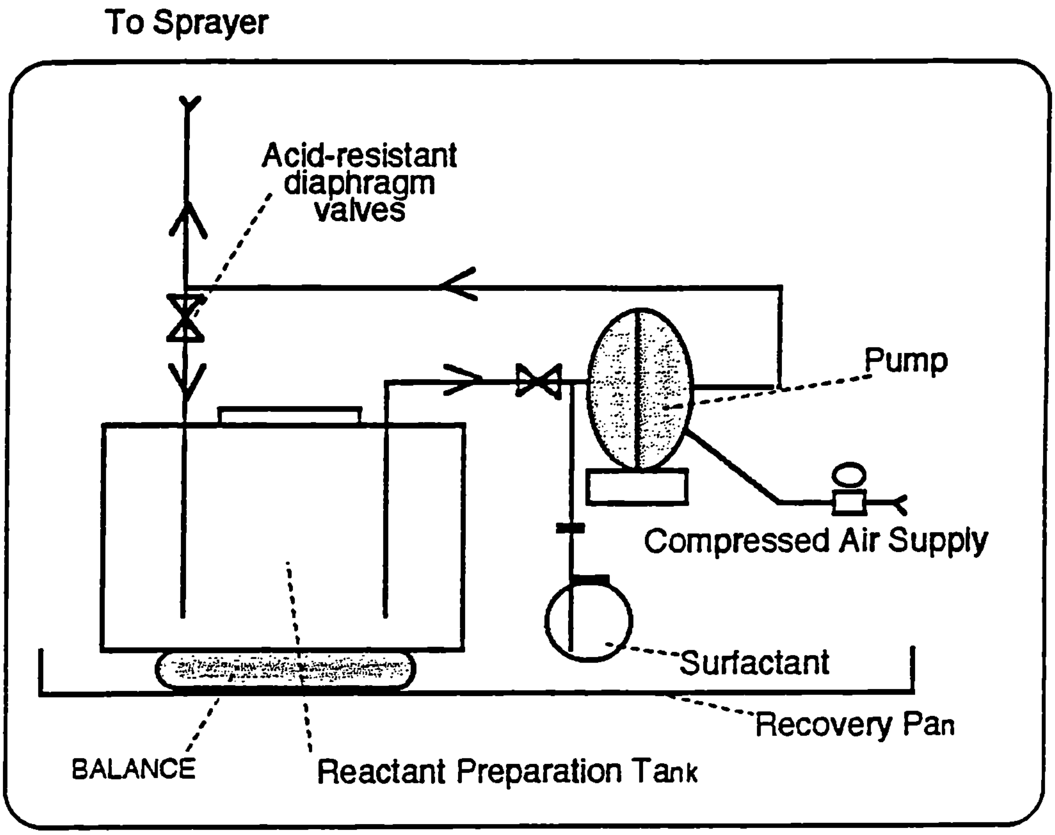


Figure 1: Simplified foam generating unit

8.18. DEVELOPMENT AND MANUFACTURING OF A FACILITY FOR THE DECONTAMINATION BY ELECTRO-ETCHING AND APPLICATION TO ALPHA RADIOACTIVE WASTE FROM THE "RM2" INSTALLATION

Contractors: CEA-Valrhô
Contract No.: FI2D-0065
Work Period: December 1991 - June 1994
Coordinator: J R COSTES, CEA-Valrhô
Phone: 33/66 79 63 13 Fax: 33/66 79 64 22

A. OBJECTIVES AND SCOPE

The "RM2" installation is a disaffected laboratory for post-irradiation fuel examination operated from 1967 to 1982 at Fontenay-aux-Roses. This project is aimed at the industrial-scale testing of a new drum-type nitric acid electro-etching process to be used for RM2 waste decontamination.

The innovative nature of this project lies in the decategorization of Pu-bearing waste. At present in France, only surface storage sites are available for nuclear waste. These sites can only accept beta-gamma waste with little or no alpha contamination.

The prevailing radiological conditions are as follows:

- specific alpha contamination: 3.7×10^3 to 3.7×10^5 Bq/g;
- dose rates: 10 mGy/h to 1 Gy/h.

This industrial decategorization process for alpha-bearing waste can be applied wherever necessary in the European Community (reactor decommissioning to Stage 3, decommissioning of research laboratories or fuel reprocessing plants). The end result will be a net reduction of waste volumes for underground waste storage.

The radioactive waste will be removed from the RM2 installation at a rate of 700 kg per month beginning in December 1991 and conditioned in special 200 l stainless steel drums which will be placed in waste drums and transferred to the Saclay waste treatment centre in type B casks for decontamination. Before the transfer to Saclay, the drum activity will be carefully measured by an automatic neutron counter. After decontamination, the waste will be reconditioned for transfer to the ANDRA facility.

The potential benefit is that, by thorough decontamination, 7 tons of metal waste provided for underground storage could be accepted for definitive surface storage. This would considerably reduce the total cost of this operation.

B. WORK PROGRAMME

B.1. Waste characterisation and conditioning
B.2. Adaptation of the electro-etching decontamination process
B.2.1. Design and manufacture of the decontamination device
B.2.2. Implementation and commissioning testing of the device
B.3. Preliminary decontamination tests on radioactive components
B.4. Industrial decontamination of 7 tons of radioactive waste
B.5. Analysis of results
B.6. Generation of specific data

C. PROGRESS OF WORK AND OBTAINED RESULTS

Summary of main issues

A total of 380 kg of RM2 wastes were decontaminated in five sessions in the drum prototype. The results were not very different from the expected results based on the laboratory investigations discussed in the second progress report.

The residual radioactivity was well below the surface disposal acceptance limits: the objective of waste decategorization is thus feasible.

It would be advisable to sort the waste according to its component materials, as the same decontamination procedure is not applicable in all cases.

Upcoming tests will be conducted without electrolysis, using only waste agitation in the acid bath followed by filtration.

Stainless steel wastes with the highest α contamination will be set aside for decontamination in the 37 l pilot unit in the spring of 1994.

Progress and results

1. Adaptation of the electro-etching decontamination process (B2)

The prototype is a commercial electrolyzer, normally used for electroplating of bolts or other small items. The basic unit was modified in several respects for the purposes of this contract: most of the copper current busbars were replaced by duralumin bars, and the drum itself is made of titanium sheeting; the current direction was reversed, and extra sensors were added.

The drum is rotated by an electric motor, and is partially immersed in the electrolyte contained in a polypropylene decontamination tank. In this position, the drum is placed between two stainless steel cathode plates positioned vertically on the walls. Electric power is supplied by a current generator rated at 1500 A.

After decontamination, the drum and its load of waste parts are immersed in a second tank to ensure complete rinsing of the metal waste.

A liquid filtration and transfer rack is used to filter the liquid waste across a *Pall* bag filter. The pneumatic pump and all the valves are controlled from a fully instrumented console located outside the process cell. The process equipment is located in a very large hot cell to allow manual intervention by operators with α-protective clothing when necessary.

2. Preliminary decontamination tests (B3)

The process wastes, coming from RM2 installation, are highly variable, consisting mainly of ferrous metals and aluminum alloys. In some baskets the average dose rate may reach 0.1 $Gy \cdot h^{-1}$ with peak values of 0.4 $Gy \cdot h^{-1}$. Most of the irradiation is due to ^{137}Cs, which accounts for over 90% of the βγ activity; the α activity is high, ranging from 7 $MBq \cdot kg^{-1}$ (0.2 $Ci \cdot t^{-1}$) to 180 $MBq \cdot kg^{-1}$ (5 $Ci \cdot t^{-1}$). Most of the items are plates or cylinders, but some are massive parts or major components.

First Test (Basket 02)

The first test was conducted with one-third of a drum of ferrous waste including machinery and electric motors (0.15 $Gy \cdot h^{-1}$) as well as less acceptable waste such as a gearbox containing 5 liters of oil.

The decontamination tank was filled with 1200 liters of 5N nitric acid. After 45 minutes of electrolysis at 1000 A (during which nitrous fumes were released on contact with the mild steel under the high current), the items were allowed to drain, then rinsed and sampled.

Second Test (Remainder of Basket 02)

The electrolysis conditions were modified to limit the release of nitrous fumes: the second test was conducted at 150 A for 70 minutes. After removing the large objects and adding a second batch of waste items, the electrolysis was repeated for 2 hours at 150 A. The parts were allowed to drain, then rinsed and sampled.

Following these two tests, the acidity of the electrolyte dropped from 5N to 3.8N.

Third Test (Basket 11)

For this test the waste was treated for 45 minutes at 900 A. As most of the waste consisted of stainless steel parts, no nitrous fumes were observed.

Fourth Test (Remainder of Basket 11)

The parts were electroetched for two hours at 1200 A.

At this point the acidity was only 3.4N, and a number of problems were noted, including leakage from an "O" ring seal and pollution of the electrolyte by oil and grease.

The decision was made to renew the electrolyte. When the decontamination tank was emptied, mercury (used to ensure good electrical contact on the drum rotation bearings) was found at the bottom of the tank. The plastic housing was partly deformed and was no longer leaktight.

The tests have been interrupted temporarily as a result of this incident, and the electrolysis current supply to the drum will have to be modified.

3. Interpretation of Radioactive Test Results

The test results are summarized in the following tables. Judging by the liquid waste contamination level, the initial activity in the drums appears to have been overestimated by a factor of 2 to 3. This means that the decontamination factors are also overestimated.

Nevertheless, three important points may be noted:

- Decontaminating mild steel at high current levels results in decomposition of the electrolyte with a loss of :

$$1920 - 4 \times 454 = 104 \text{ moles of nitric acid}$$

after allowing for the acid consumed by steel dissolution:

$$Fe + 4HNO_3 \rightarrow Fe(NO_3)_3 + NO\uparrow + 2H_2O.$$

High current levels (in the order of 1000 A) should therefore be avoided, and the parts treated for longer time periods (> 2 hours) at lower amperage. Under these conditions, electrolysis may even be unnecessary.

A decontamination factor of about 30 should be obtained.

- Decontamination of stainless steel is more effective, with decontamination factors of 100 to 200. The resulting waste is well below the acceptance limit for surface disposal.
- The current supply system must be redesigned without using mercury contacts.

Industrial tests balance

	Basket 02			Basket 11		
mass (kg)	220	mild steel		160	stainless steel	
	calculated	residual activity	DF	calculated	residual activity	DF
total alpha (Bq)	1.30E+10			1.89E+09		
average Bq/g	59091	580	102	11813	18.02	649
Total beta/gamma	1.90E+11			1.93E+10		
average bq/g	8.64E+05	24400	35	1.21E+05	530	228

Liquid effluent balance

Initial acidity (N)	5	Consumed moles :	
Final acidity (N)	3.4	1920	
Volume (l)	1200		
Radioactivity	measured	inferred	overestimed
Total alpha (Bq)	4.94E+09	1.49E+10	3.01
Bq/ml	4120		
Total beta (Bq)	1.116E+11	2.09E+11	1.88
Bq/ml	9.30E+04		

Iron balance

Iron (g/l)	21.2
Total iron (g)	1200
erosion	6.69 %
moles dissolved	454

8.19. DEVELOPMENT OF A ROBOTIC SYSTEM (TRT) FOR THE REMOVAL OF TUBES FROM A LATINA STEAM GENERATOR, WITH SUBSEQUENT MELTING AND RADIOLOGICAL CHARACTERIZATION

Contractors: Ansaldo Spa, Siempelkamp
Contract No.: FI2D-0066
Work Period: November 1991 - April 1994
Coordinator: M CIARAVOLO, Ansaldo/Genova
Phone: 39/10/655 87 05 Fax: 39/10/655 87 99

A. OBJECTIVES AND SCOPE

The work is related to the design, manufacturing and testing of a Robotic system to Retrieve Tubes (TRT) from a steam generator of Latina Magnox Power Plant owned by ENEL. The retrieved material will be characterized radiochemically, melted and characterized again with a view to reuse.

The use of TRT will allow the reduction from 40 mSv to about 2 mSv of the radiation dose to workforce and a significant cost reduction mainly due to the reuse of obtained material.

The objective of the proposed project is the detailed study of a dismantling technique based on a robotic system and characterisation of the material obtained after melting. The Magnox steam generators are the largest contaminated plant items although activity levels are low.

The results of the work should strengthen the cost-effectiveness of reusing material from decommissioning. Moreover, it will allow to get detailed information about the radiological aspects of this technique and be applicable to other Magnox or gas/graphite type reactors.

B. WORK PROGRAMME

B.1. System requirements definition and studies (ANSALDO with ENEL support)
B.1.1. Analysis of the robotic system (TRT) layout and of the environmental and radiological conditions (in cooperation with ENEL)
B.1.2. Functional requirements definition of the TRT
B.1.3. TRT design at system level (Two TRT designs will be investigated)
B.2. Mock-up design and manufacturing (ANSALDO)
B.2.1. Design of a SG tube nest mock-up based on real geometric requirements
B.2.2. Mock-up manufacturing and installation in the Ansaldo testing facility
B.3. TRT design and manufacturing (ANSALDO)
B.3.1. TRT detailed design based on B.1. and control system design (Hardware and Software)
B.3.2. Mechanical part manufacturing, commercial part purchasing and assembly and integration of all parts.
B.4. Testing of the TRT on the mock-up (ANSALDO)
B.5. Site preparation (ANSALDO with ENEL support)
B.6. TRT operation and material transportation (ANSALDO + ENEL & SIEMP. support)
B.6.1. TRT installation on the SG made available by ENEL
B.6.2. Tube cutting and removal; charging of cut material in the transport containers
B.6.3. Chemical and radiological characterization - Transportation to Siempelkamp, Krefeld
B.7. Restoration of the site
B.8. Radiochemical analysis (SIEMPELKAMP)
B.9. Melting (SIEMPELKAMP)
B.10. Radiochemical analysis (SIEMPELKAMP)
B.11. Generation of specific data

C. PROGRESS OF THE WORK AND RESULTS OBTAINED

Summary of main issues

The activities performed in this period saw the completion of the manufacturing activities related to both the Tube cutting and Retrieve Robotic system (TRR) electromechanical part and control system, and the contamination containment system which had been erected on site near the steam generator, object of the intervention.

A campaign of qualification tests was conducted on a specially built mock-up after the assembly and integration of the system at the workshop, in particular to verify the capability of the system to perform the required operations and to provide training for the operators.

At the end of the above tests the TRR was transported to the Latina plant and installed inside the contamination containment system and the steam generator.

In the meantime the licence for melting the tubes in Siempelkamp CARLA plant was obtained and the scanner for the radiological characterisation of the drums before transportation designed and manufactured by Siempelkamp.

Progress and Results

1. Mock-up design and manufacturing (B.2)

The full-scale Steam Generator (S.G.) tubes bundle mock-up, to be used for shop testing of the cutting and retrieval device was ordered and built.

2. TRR design and manufacturing (B.3)

On the basis of the detailed design performed in the previous period, the manufacturing of the dismantling system was completed, as well as commercial parts purchasing and integration.

Picture 1 shows the TRR during the assembly phase.

The control system was also completed: the hardware consists of a control panel (see Picture 2) housing both the power drivers and the Programmable Logic Controller with the axis controller boards. The software allows control of execution of the various operations in a semi-automatic mode, managing the proper execution of the sequence while supervising safety interlocks, and the verification of the various machine configurations during the cycle by means of a mimic panel.

A double camera Closed Circuit TV system is mounted on the TRR to give a visual feedback to the operator during the positioning and cutting operations.

3. Testing of the TRR on the mock-up (B.4)

An extensive test campaign on the S.G. tube bundle mock-up was conducted to verify the capability of the TRR to perform the positioning and cutting operations in an environment simulating the real conditions. The proper execution of the physical integration between the device and the control panel was checked and some problems related to disturbances and noise were solved.

The performance of about thirty cuts demonstrated a good reliability of the machine, giving at the same time the opportunity of personnel training, in particular as far as the knowledge of the possible problems which can be encountered during the real cutting campaign, and the actions to solve them are concerned.

Picture 3 shows the TRR in operation on the tube bundle mock-up during the workshop testing phase.

4.Site preparation (B.5)

The contamination containment system, which essentially consists of a box providing a dust-tight environment during tube cutting and a controlled working area where personnel performs cut tubes handling and TRR routine checks and maintenance, was designed and installed on the SG object of the intervention after the completion of site preparatory works, such as scaffolding removal and modification and cleaning of the site, in charge of ENEL.

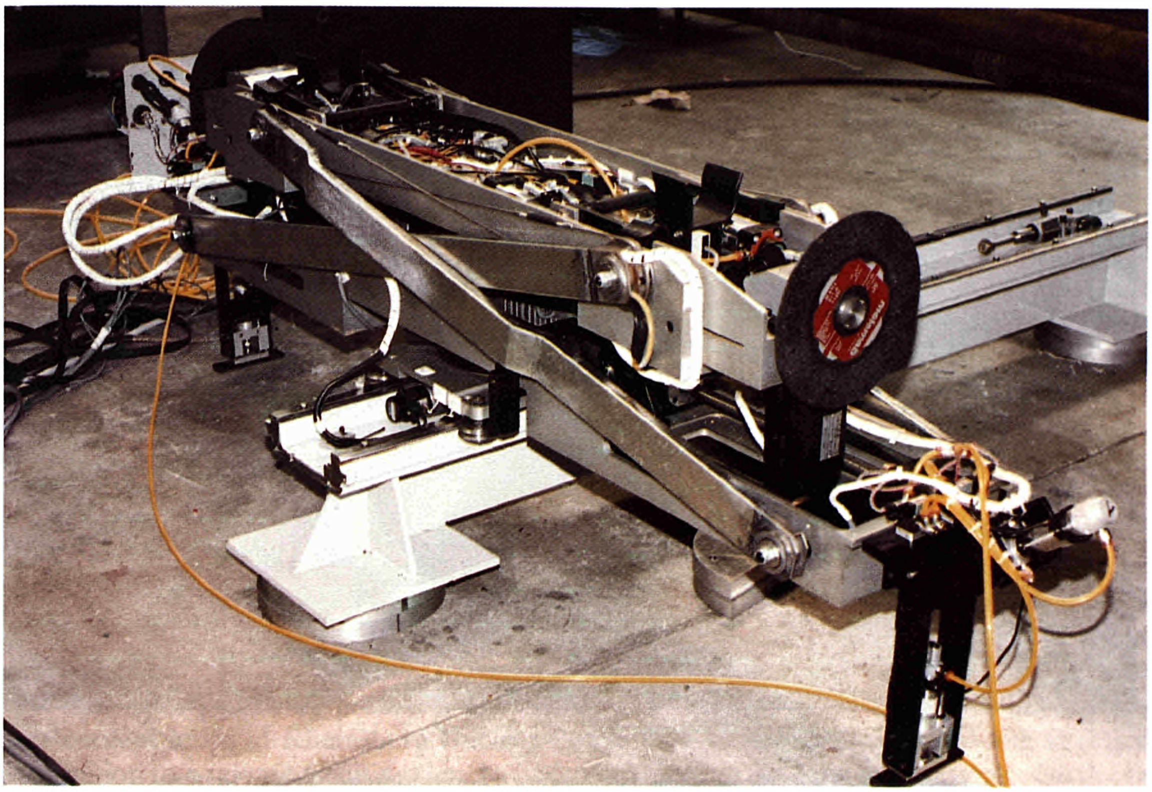
5.TRR installation (B.6.1)

After the testing campaign on the mock-up, the TRR was transported to the Latina plant where it was installed on the Steam Generator made available by ENEL; the guiding rails were mounted inside and aligned with the manhole penetration, the cutting device was re-assembled and all cables from the control panel to the TRR connected and tested.

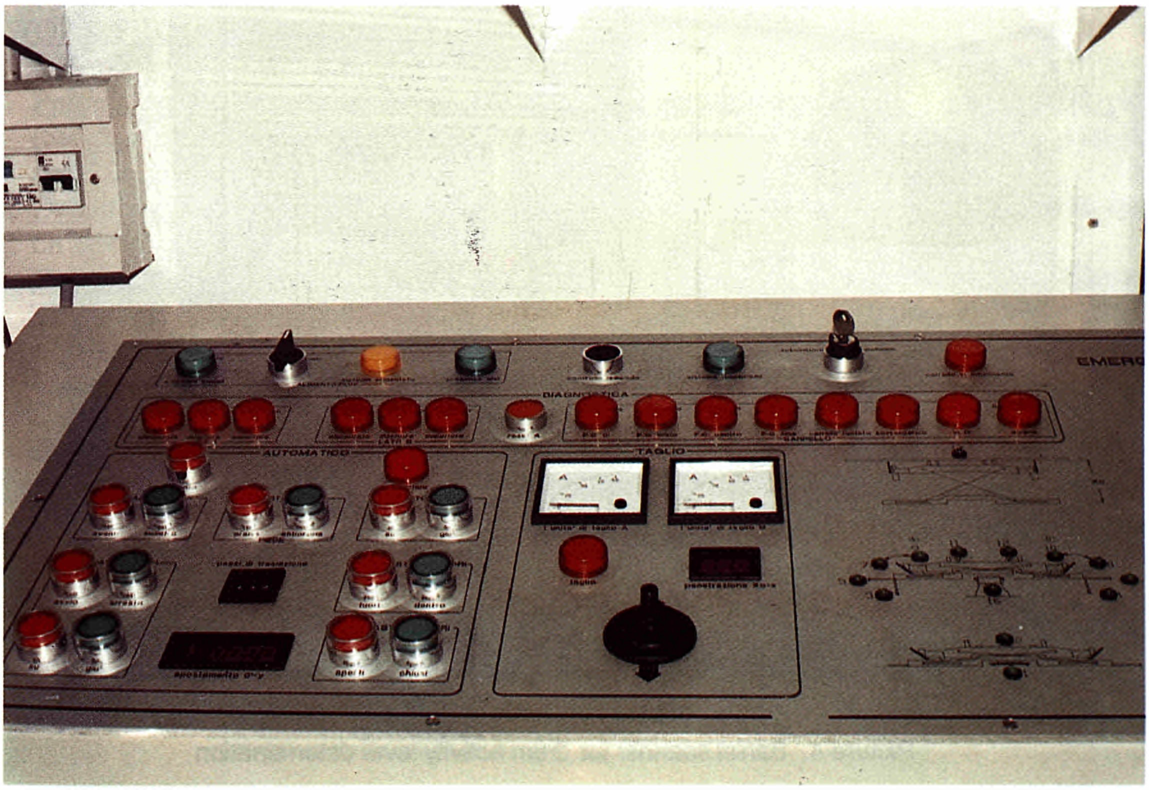
6.Radiochemical analysis (B.8)

The results of the previously performed radiological characterisation of tube sample material were presented to the Authorities to obtain the licence for melting the scrap metal coming from Latina steam generators in the Siempelkamp foundry, the licence was issued under the condition that the "Zentralstelle fur Sichereitstechnik" will attend the melting campaign to make measurements of the atmosphere around the furnace.

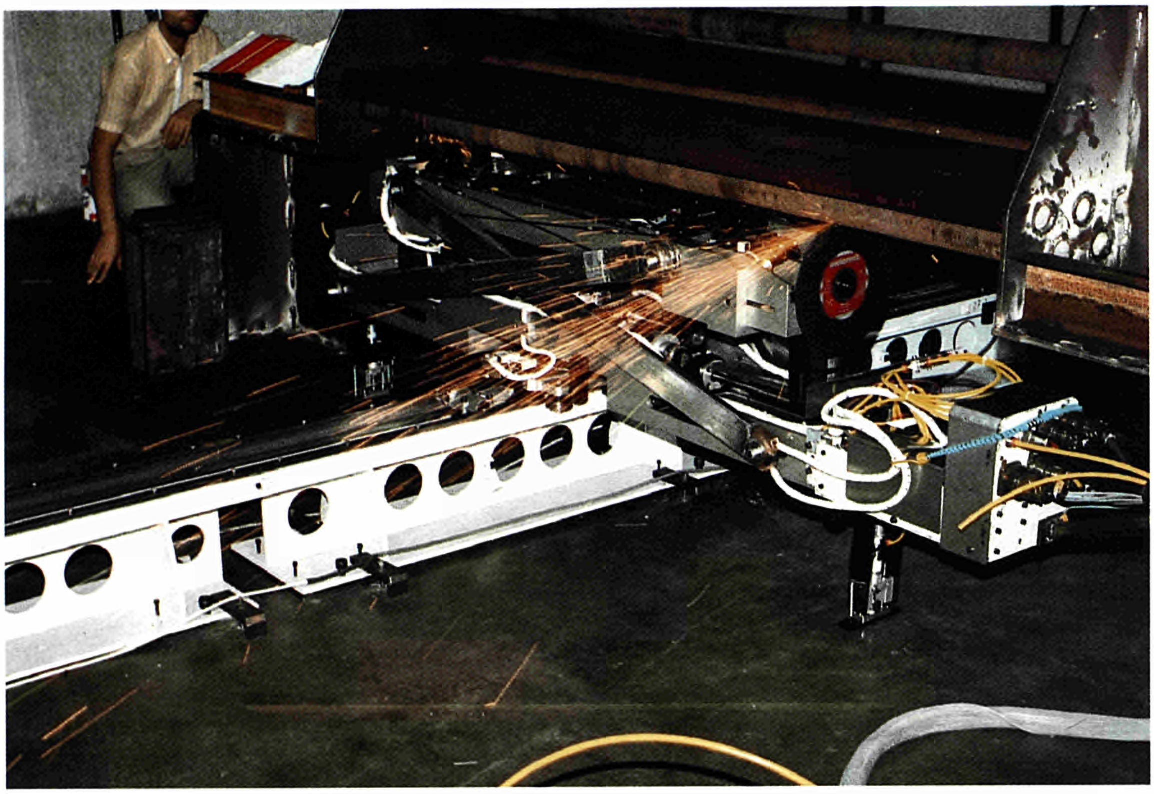
The barrel scanner for checking the activity levels (about 38 Bq/g from previous analysis) of the 200 l drums containing the cut tubes was designed and manufactured (see Picture 4), as well as the pattern for the lead radiation shielding, which will allow the lead radiation shielding itself to be cast.



Picture 1. TRR during the assembly phase



Picture 2. TRR control panel



Picture 3. TRR in operation on the mock-up during workshop testing

Picture 4. Barrel scanner for drum activity level determination

8.20. ASSESSMENT OF DECONTAMINATION PROCEDURES FOR WWER-PWRs WITH A VIEW TO MINIMIZE THE GENERATION OF WASTE

Contractors: Siemens-KWU, EWN/KKW Rheinsberg
Contract No.: FI2D-0067
Work Period: December 1991 - June 1994
Coordinator: H WILLE, Siemens KWU Erlangen
Phone: 49/9131/183 339 Fax: 49/9131/182 821

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 BR3 are also to be treated. These parts, among others, are to be decontaminated in accordance with the requirements to levels permitting unrestricted release or reaching at least remelting conditions.

The most promising processes shall be applied for the decontamination of one of the reactor coolant loops at Rheinsberg Nuclear Power Plant. This treatment will aim at reducing contamination to a release (melting) level without dismantling the loop components. On the basis of the investigation results, a concept is to be established for the decontamination of WWER-type PWR reactors, illustrated with reference to Rheinsberg Nuclear Power Plant.

B. WORK PROGRAMME

B.1. Process evaluation of decontamination processes permitting subsequent unrestricted release (Siemens): selection of the most effective process for the decontamination of component surfaces.

B.2. Decontamination of removed components: the process selected in para B.1. shall be tested on laboratory samples and full-sized components.

B.3. Decontamination on one primary loop of the Rheinsberg NPP (Rheinsberg and Siemens) The in-situ decontamination of one of the three primary loops of the reactor will consist of:
- pre-treatment with APCE/CORD to remove oxide layer;
- succession of oxidation and pickling cycles as determined in B.1. and B.2.
- determination of residual activities.
- treatment and conditioning of process waste.

B.4. Overall concept for the decommissioning of a WWER nuclear power plant (Siemens and Rheinsberg). Based on the experience and the results of the preceding work programme, an optimized concept for the decommissioning of reactor components for WWER NPPs shall be developed.

B.5. Generation of specific data: Specific data on costs, worker exposure, working time and waste arisings will be derived from the execution of items B.2. and B.3.

C. Progress of work and results obtained

According to the work programme the investigations foreseen under B1 and B2 are finished. The decontamination of the primary loop of the Rheinsberg NPP is now foreseen to be performed in March 94, after the reception of the permission by the end of '93.

Results

1 Decontamination Tests with Components

1.1 Tests at EWN, NPP Rheinsberg

In the NPP Rheinsberg decontamination tests in a large decon vessel were performed. The vessel with a volume of about 13 m^3 was filled with components removed from the primary loop. The material of the components is X 8 CrNiTi 18.10. They were removed during major repair activities in 1982 and 1986. The total weight is about 5000 kg.

The components were subjected to a full system decontamination before removal, the contamination was:

	Piping	Valve housing
Contact dose rate	~ 5 µSv/h	~ 460 µSv/h
Removable contamination	~ 20 Bq/cm^2	~ 200 Bq/cm^2
Co-60 specific surface contamination	~ 25 Bq/cm^2	-
Mass-specific total activity	~ 51 Bq/g	~ 600 Bq/g

The individual components and the location during service are shown in Fig. 1.

The decontamination was performed in two cycles, consisting of a preoxidation and a decontamination step. The temperature was < 90 °C and the concentration of the chemicals ~ 15 000 ppm.

The results obtained are shown in Fig. 2. The numbers shown are the maximum values measured. The final mass-specific contamination shows that simple geometric parts can be decontaminated to an activity level of < 1 Bq/g which allows further reuse. More complicated components like valves need to be segmented before decontamination to reach such low contamination levels.

The waste treatment by evaporation resulted in ~ 1 m^3 of solid waste after fixation in concrete. If the components were disposed off directly, a volume of 4.1 m^3 would be required. For a large scale application exists the possibility to reduce the amount of waste generated remarkably by UV decomposition of the decontamination chemicals.

1.2 Tests at SCK/CEN Mol

The decontamination tests at SCK/CEN Mol were performed in a small decontamination loop. (Fig 3). The test vessel for the samples is equipped with ultrasonic heads and is used to examine the influence of ultrasonic on the decon result. The general procedure is the CORD process with high oxalic acid concentration during the decontamination step. Each cycle comprises the classical CORD steps i.e. oxidation with permanganic acid, reduction, decontamination with oxalic acid and decomposition of oxalic acid with UV-light. However, the procedure was slightly modified for the lst step. Instead of using UV together with H2O2, the solution is decomposed by slow addition of H2O2 in the presence of Fe^{3+}. The CO2 in the off-gases is continuously monitored during the oxidation process with an infra red photometer, so that the end of the reaction can be observed. After decomposition of the oxalic acid, the absence of cobalt complexants in the solution is verified and the liquid waste can be evacuated as standard liquid waste to Belgoprocess where it is conditioned by precipitation with ferriferrocyanide.

The CORD process was used with or without Ultrasonic (US) treatment in two cases:

Pieces of 1" SCH160 pipes removed from the purification system during the dismantling of the Regenerative Heat Exchanger; these pipes are only slightly contaminated since they were decontaminated during the BR3 Full System Decontamination (FSD) of 1991.

Pieces of 3"SCH160 pipes removed from the partly dismantled Spray In pipe; this line was not decontaminated during the FSD of 1991 and is still covered by the original crud layer.

1.2.1 Tests with 1" SCH160 pipe pieces

Three tests were performed; one with three 50 cm long pieces without US, the same test with US and finally a test with US with 7 pieces (the maximum) in order to test the geometry effect. Fig 4 gives the evolution of the activity in the solution during the oxidation and reduction steps. There is a marked difference between the tests performed with or without US. Without US, there is nearly no activity release during the oxidation step. The release of activity is observed only during the decontamination step. When the ultrasonic transducers were turned on, a significant release of activity occurs already during the oxidation step. Thanks to this enhancement of the oxidation attack, the residual contamination on the piece after treatment is significantly lower when US is used.

The level of the residual contamination is influenced by the number of pieces submitted to the decontamination, the higher the number of pieces in the bath, the lower the US energy per piece, and the lower the decontamination efficiency.

The final residual contamination reached after one cycle is lower than 1 Bq/g for the tests using US and slightly higher when the US is not used.

1.2.2 Tests with 3" pipe pieces

With the 3" SCH160 pipe (OD 89 mm, e=11.1 mm) in comparison with the 1" SCH160 (OD 33 mm, e=6.3 mm), only 2 pieces maximum can be placed inside the test vessel.

Three tests were also performed; one without US with one piece, the same test with US, and one test with 2 pieces and US. Due to the higher crud content, 2 to 3 cycles were necessary to remove completely the contamination layer.

The same observations as for the 1" tests can be made:
The use of the US enhances the reactions, which results in a higher release of activity during the oxidation and the decontamination steps. The same final residual contamination is obtained after 2 cycles with US as after 3 cycles without US.

For 2 pieces treated, the efficiency of the combination US/CORD is somewhat lower due to the absorption of US in the thick pipes. In the US test with the two pipes, the pipes were not completely similar, the first one was a straight pipe of 50 cm long and the second one was a straight pipe 50 cm long with a 45° elbow welded at one end. The pipe with the elbow was slightly less decontaminated in the elbow itself probably due to the presence of the welds.

1.2.3 Conclusion of the decontamination test on cut pipes

The study has clearly shown the positive effect of the combination of US power together with the CORD chemistry. The improvement in the decontamination effect allows slightly contaminated pieces to reach very low residual contamination in only one cycle. For higher contaminated pieces with a thick crud layer, it is still necessary to perform several decontaminations. However with the combined effect of US and CORD, the number of cycles can be reduced.

D. Outlook

The decontamination of the regenerative heat exchanger of the BR3 will be performed during the first quarter of 1994. In Rheinsberg the decontamination of one loop is planned for March 94. Subsequently the evaluation and the concept for a VVER Reactor decontamination will be developed.

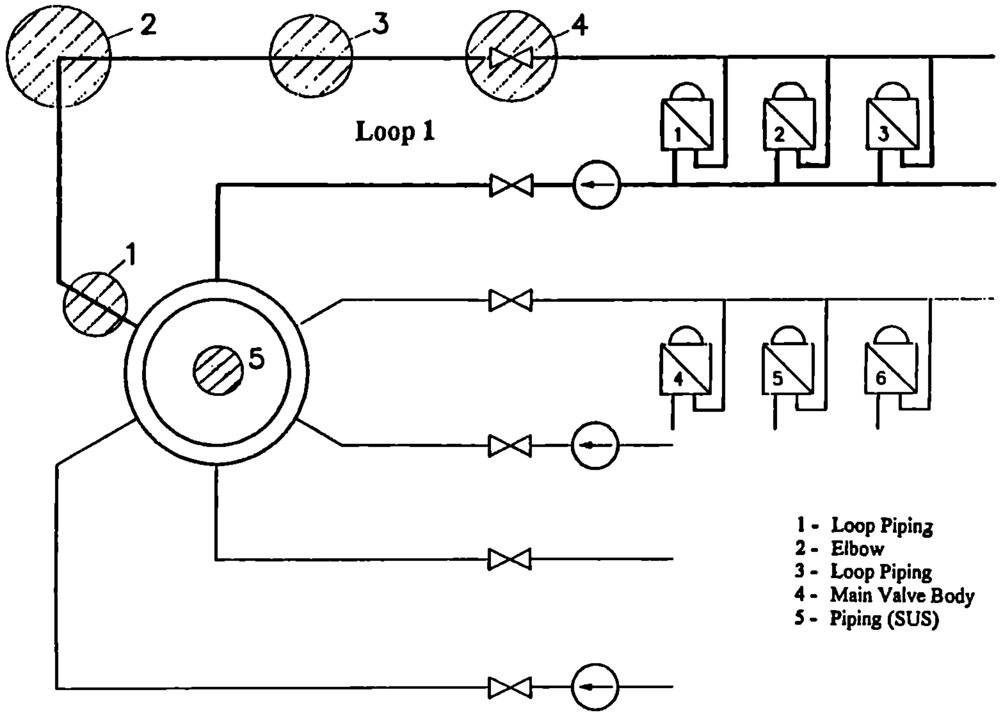


Fig. 1: Location of the Decontaminated Parts during Operation of VVER 70

Part	Time of Dismanteling	Mass [Mg]	Surface [m²]	Co-60 [Bq/cm²] before Deconta-mination1)	Co-60 [Bq/cm²] after Deconta-mination1)	Co-60 [kBq] after Deconta-mination	Mass Specific Activity4) [Bq/g] after Deconta-mination	Contact Dose2) [µSv/h] after Deconta-mination
Loop Piping	05.03.1986	0.400	3.72	< 11.4	< 7.08	< 263.38	< 5.7	< 6.0
90° Elbow	05.03.1986	1.21	4.9	< 19.1	< 0.26	< 12.7	< 0.09	< 0.2
Loop Piping	05.03.1986	0.292	2.12	< 7.2	< 1.09	< 23.1	< 0.7	< 0.3
Main Valve	14.04.1982	3.0	8.4			< 41 400³⁾	119³⁾	< 92.0
Piping System SUS	05.03.1986	0.098	4.34	< 24.8	< 0.1	< 4.3	< 0.3	0.07

1) maximum value
2) maximum value at a background of 0.07 µSv/h
3) calculated value from the maximum contact dose measurement
4) calculated according to the nuclide distribution
(Co-60 = 11.6 % of Total activity)

Fig. 2: Results of Component Decontaminations at EWN KKW Rheinsberg

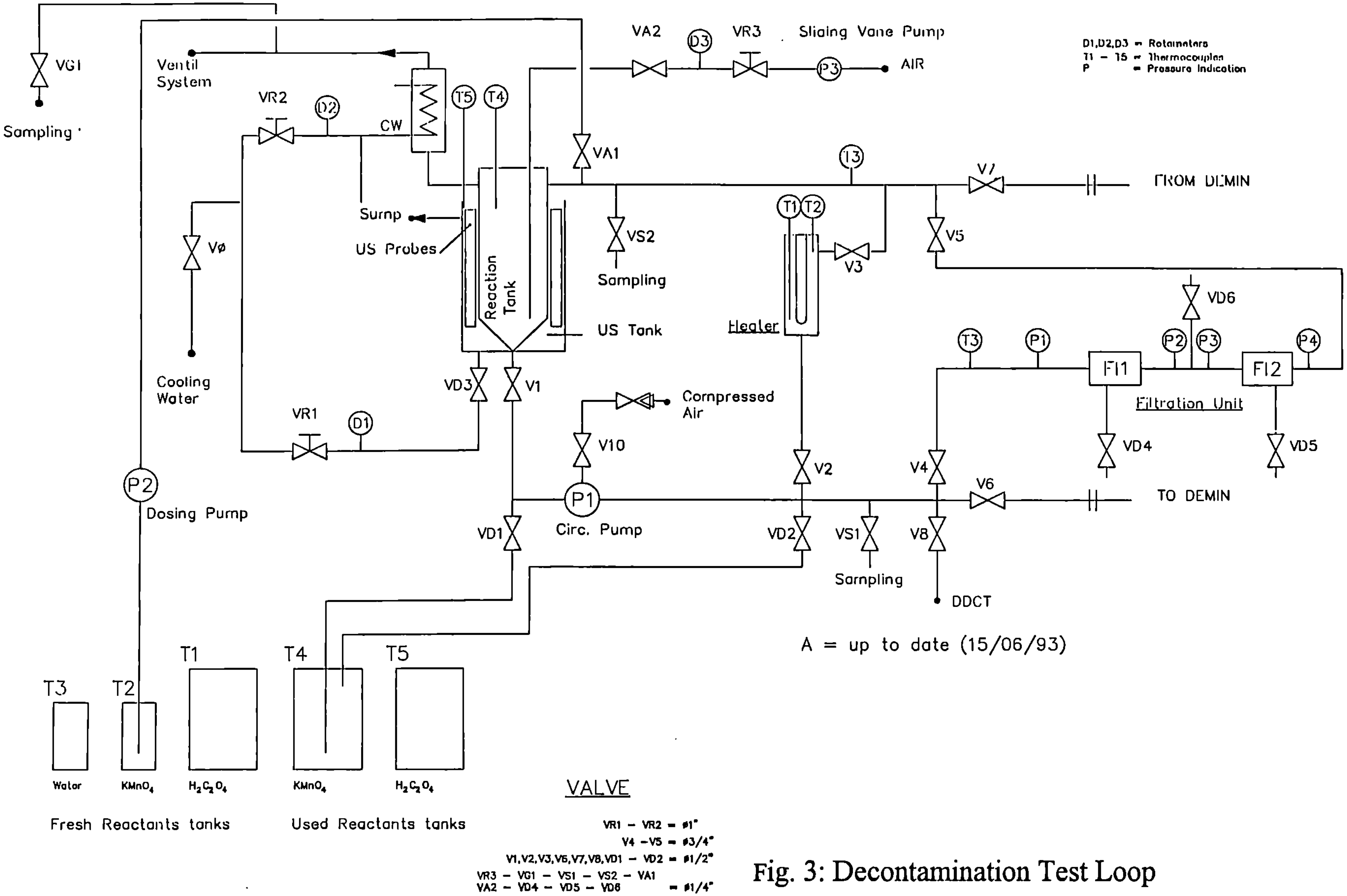


Fig. 3: Decontamination Test Loop

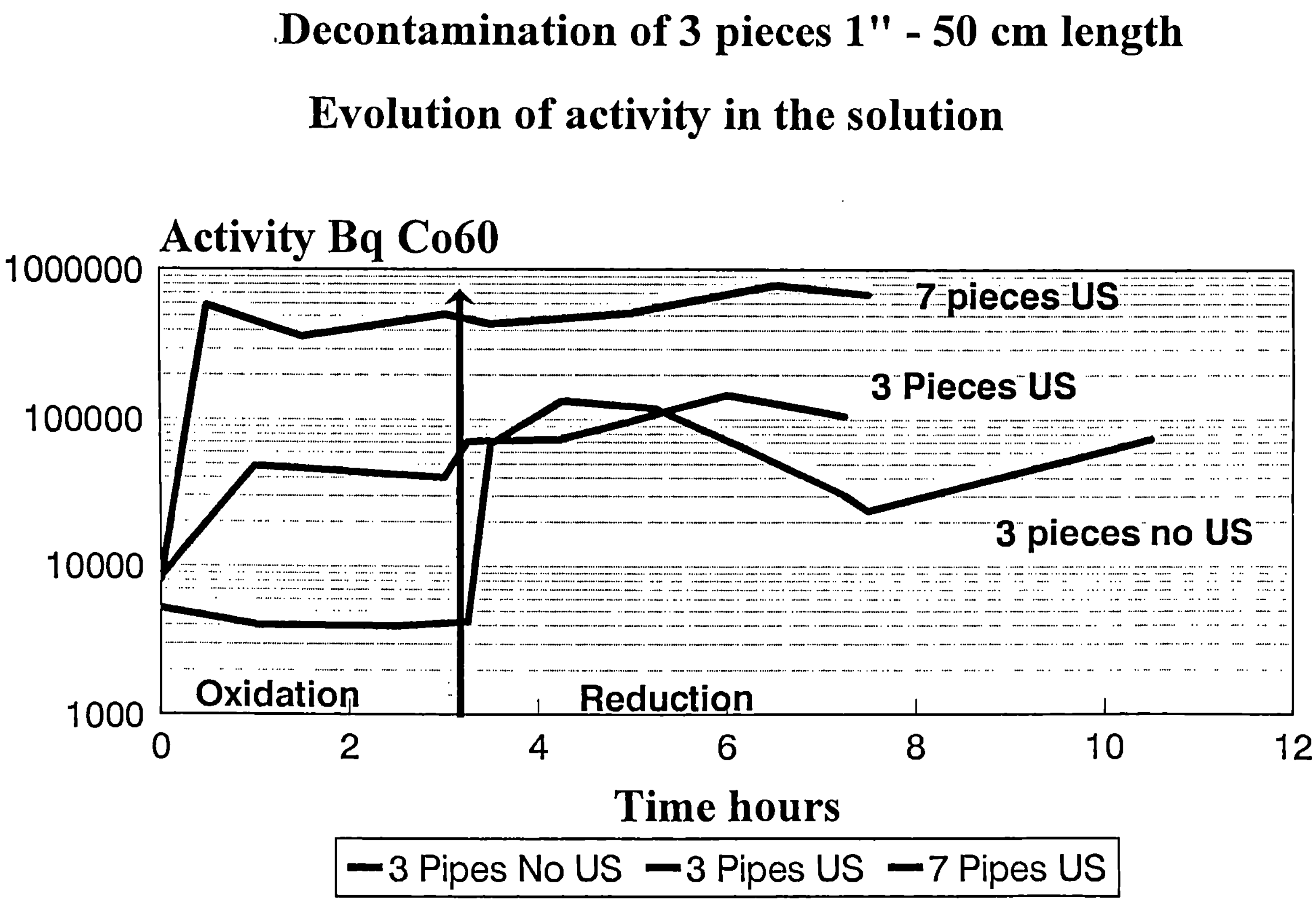


Fig. 4: Decontamination of 3 pieces 1" - 50 cm length
Evolution of activity in the solution

8.21. DESIGN, MANUFACTURING AND APPLICATION OF A TELEOPERATED DEPLOYMENT SYSTEM BASED ON THE "NEATER" ROBOT

Contractors: AEA Technology
Contract No.: FI2D-0068
Work Period: November 1991 - April 1994
Coordinator: G V COLE, AEA Technology Harwell
Phone: 44/235/821 111 Fax: 44/235/436 138

A. OBJECTIVE AND SCOPE

The work relates to the construction and operation of a full-scale teleoperated cutting, monitoring and decontamination system to remove highly contaminated equipment. The project includes the proving of the system in an inactive mock-up followed by size reduction of contaminated equipment in the DIDO High Activity Handling Cell, removal of items, monitoring and decontamination.

Advantage will be taken of techniques and equipment developed under earlier work in the Programme for Decommissioning of Nuclear Installations (e.g. FI2D-0012), as well as of other initiatives (e.g. Teleman).

The range of applications covers all types of nuclear facilities. Specific benefits of the project will be:

- a reduction in occupational exposure to radiation, due to automatic and remote manipulation;
- a reduction in background radiation levels in the vicinity of the work areas;
- an improvement of the Quality Assurance of operations and waste accountancy;
- improved safety and protection of operators;
- a reduction in the cost of decommissioning activities due to e.g. more rapid work completion;
- improvement in the awareness of remote technology for decommissioning.

The system will be based on already available components such as the Nuclear Engineered Robot System (NEATER), the Telerobotic Control System (HTC) and the Stereoscopic Television System (TV3).

B. WORK PROGRAMME

B.1. Design, specification and commissioning of the robot and auxiliary equipment support frame
B.2. Design, specification and commissioning of the decontamination equipment
B.3. Design, specification and commissioning of the tools and tool change system
B.4. Design, specification and commissioning of the control room
B.5. Service requirements and interface connection will be specified and survey services provided.
B.6. Design, specification and commissioning of the viewing system, including pan/tilt unit and controls, cables, lighting.
B.7. Design, specification, construction and commissioning of the mock-up area
B.8. Preparation of safety case and obtention of approval for robotic safety, active area safety and of operational method statements and procedures.
B.9. Site-specific activities including delivery of equipment, installation of control room, etc.
B.10 Removal, monitoring and decontamination of in-cell equipment, waste management, removal of robot system and cleaning up work area.
B.11 Specific data on costs, worker exposure, working time, waste arisings and fissile material recovery will be derived from the execution of work.

C Progress of work and obtained results

Summary of main issues

Activities B1 to B9 are complete. The main variance was that mock-up trials and operator training in B7 was extended to incorporate operator suggestions and to fully test the equipment before radioactive installation.

Activity B10 is ongoing and about 80% complete. The project will be complete before the end of March 1994. A summary of human resources utilised is given in Figure 1.

Progress and results

1. Inactive Work

The equipment design, construction and testing, B2 to B6, progressed with no significant problems or delays.

The mock-up, B7, after construction was used to prove the system was fit for purpose, to prove procedures and to train 3 operators. The operators were from the Reactor Decommissioning Team and were familiar with DIDO HAHC operation and radiation work principles. Training included:

- System operation
- Size reduction techniques
- Monitoring techniques using the robot
- Decontamination techniques
- Installation sequencing

The operators made several suggestions to improve telerobotic system operations. These were noted and, where practicable, incorporated into the system.

When fully satisfied, the decision was made to install the equipment into the High Activity Handling Cell.

2. Active Work

Installation, B9, took 15 working days against an estimated 25 days. The extra time to practice installation in the mock-up was justified by the installation being almost trouble free. The only significant event was that equipment was manhandled into the cell maintenance bay. It had been planned to use the existing in-cell trolley, but this broke down and needed repair. Equipment commissioning was finished in 3 days and the equipment was ready for use. Active decommissioning work, B10, started on 2 August 1993. An initial detailed survey using the robot revealed the bench top was mostly Low Level Waste [LLW] category and, therefore, bench top decontamination was not necessary. A cell visual examination, using a robot deployed camera, showed the bench was considerably more substantial than had been indicated. The underframe comprised: 75mm 'I'beam, 100 mm channel, and 50 mm angle mild steel sections. A 2.5 mm thick tray was resting on a 6 mm thick bench top. The tray was only held in place by the bolts securing the saw rig and vice bench machinery. The direct viewing was very poor through zinc bromide cell windows. The operators preferred the remote cameras supplied with the system.

A grid of 50 mm holes was drilled in the bench top. The saw was deployed to cut between the holes and produce 100 mm square waste pieces. Many pieces fell to the floor when cut. NEATER, deploying the gripper, was used to retrieve the dropped pieces. The bench support structure was size reduced, monitored and posted out. As work continued, the remaining structure became more unstable.

Each piece of waste was monitored using the telerobotic system. In this way much less ILW(R) resulted than anticipated. A comparison of waste volumes arising is given in Figure 2. Most of the fines generated fell to the floor. Fines were collected using a robot deployed vacuum cleaner.

After the main bench had been size reduced and general housekeeping carried out, the second, vacuum bench was moved to the cell centre using the overhead hoist and the robot. The vaccum bench was much less complicated than the main bench. A radiation survey showed the whole bench was LLW. It was cut into larger pieces for disposal into a LLW ISO container.

During decommissioning hot spot areas were discovered that had been covered by lead bricks. These bricks were removed and the hot spots cleaned. Tool breakages were significantly less than would have been expected if traditional methods had been used. Manual intervention to the system was minimal.

At the start of active work the cell background level was 20 mSv/hr. By December it had reduced to 2mSv/hr. The cell is now nearly empty. The only telerobotic operations left are to decontaminate and clean the floor. Once done, the background levels within the cell are expected to be below the target of 0.2 mSv/hr.

The sequence of active work is illustrated in Figures 4 to 7.

3. Radiation Doses

Installation

Total Dose received by staff during the system installation and commissioning was 3.27 mSv, compared with the estimated figure of 1.7mSV. At the time, no mechanism was in place for Health Physics [HP] staff to record dose up-take against a specific project. Although it is not possible to state exactly this dose it is estimated that that a 300 μSv dose was received by HP staff and this has been included in the above total.

The total also includes activities carried out that were extra to the telerobotic installation but were recorded under the scheme including: cleaning, covering the maintenance bay floor with polythene sheeting, maintenance to existing trolley bus bar, applying strippable coating to maintenance bay walls, and repair to existing trolley.

Operations

The total dose received during operations is 4.2 mSv. This figure is made up of:

Operating telerobotic system	-	1.2 mSv
Waste handling	-	1.0 mSv
Telerobotic equipment maintenance	-	0.9 mSv
Man entries to tackle inaccessible fixings and features.	-	1.1 mSv

This compares with an estimated dose of 3.4 mSv which did not include for any waste handling. If handling is included, the estimate increases to 4.4 mSv.

The total dose received to 31 December for telerobotic system installation and operation is 7.47 mSv 1, see Figure 3. The dose estimated for carrying out the work, hands on, in a traditional approach was 100mSv.

Construction		Operations	
Staff Description	Mandays	Staff Description	Mandays
Project Manager	30	Supervisor	30
Senior Project Engineer	90	Non-technician	150
Project Engineers	200		
Technicians	180		
Craftsmen	120		

Figure 1: Human Resources Utilised

Waste Type	Actual waste volumes using Telerobotic methods (litres)	Estimated waste volumes using traditional techniques (litres)	Estimated waste volumes using MSM's to Plasma Arc cut bench (litres)
ILW(R)	60	60	900
LLW	1580	8000	740
			[Original method proposed before oil spill]

Figure 2: Waste Volumes

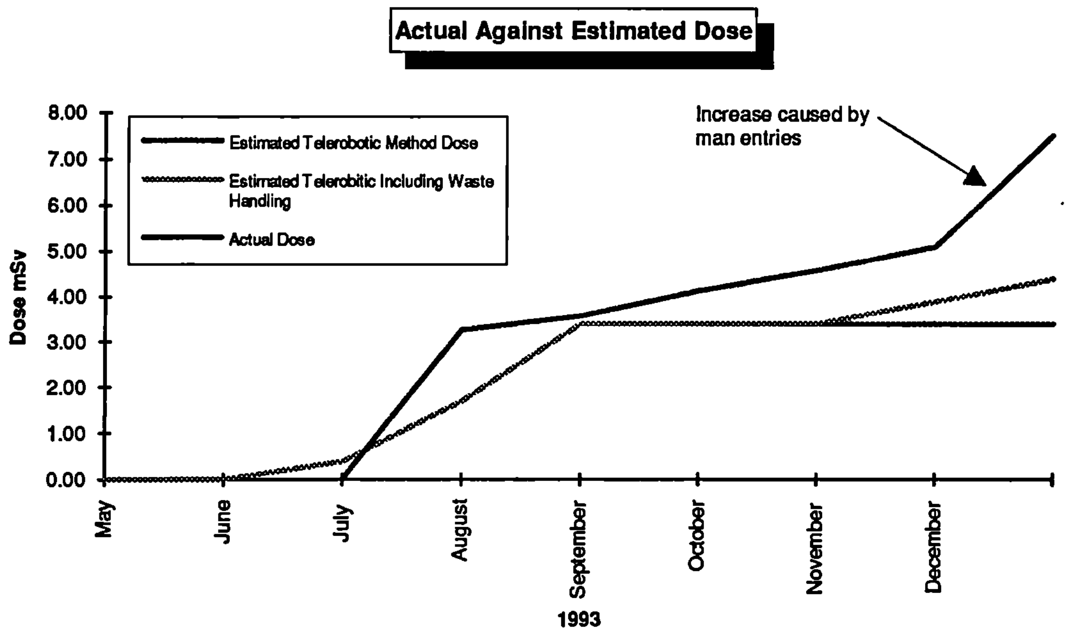


Figure 3: Graph Comparing Actual against Predicted Dose Levels

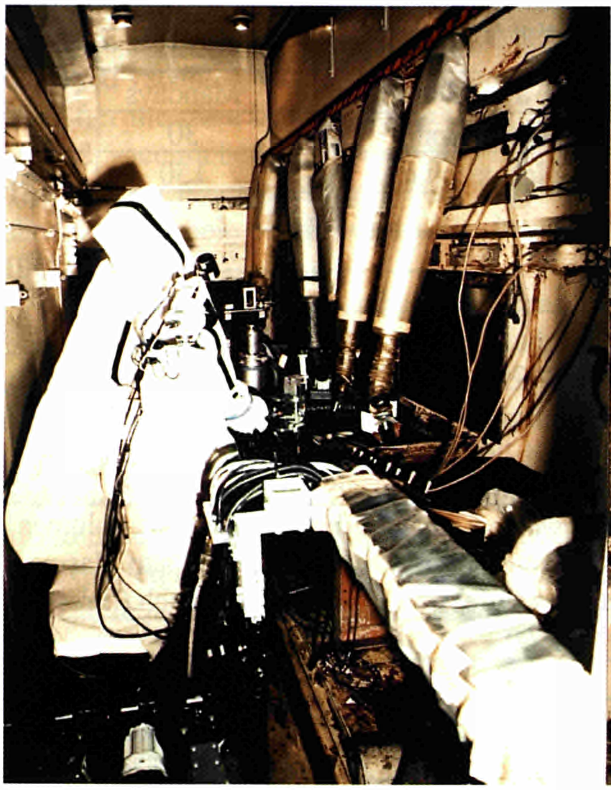


Figure 4: NEATER First Installed
2 August 1993

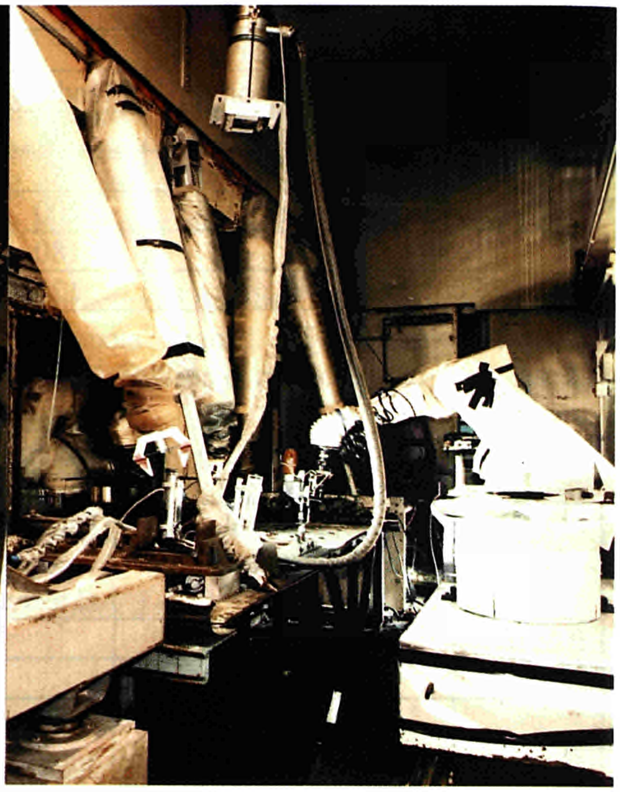


Figure 5: Size Reduction Main Bench
24 August 1993

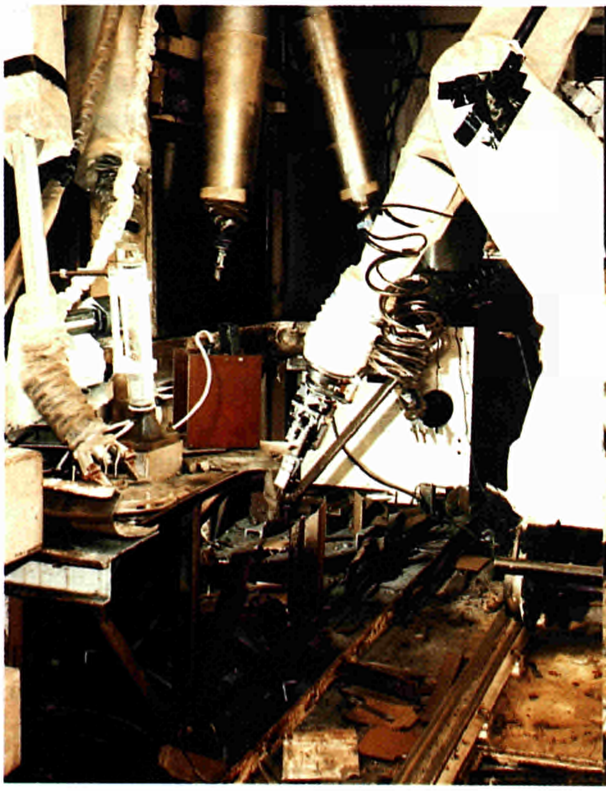


Figure 6: Main Bench Supports
15 September 1993

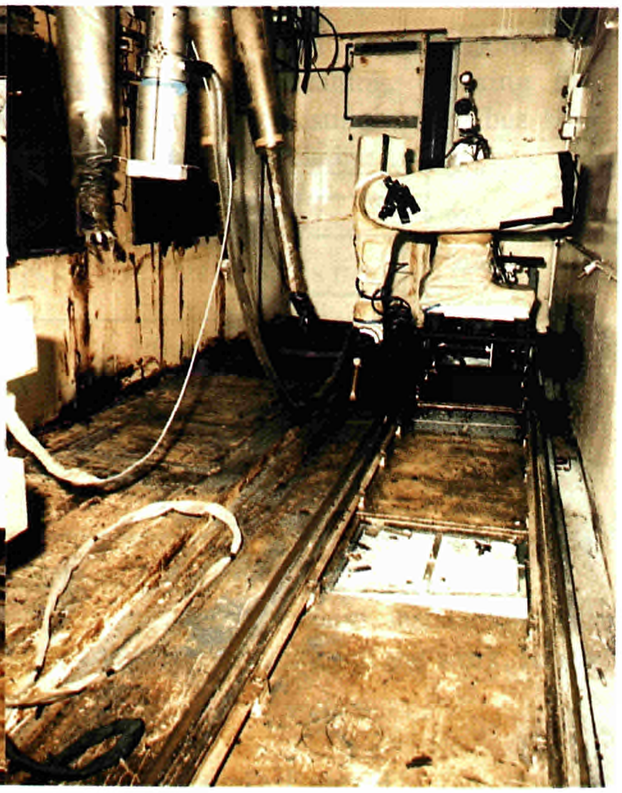


Figure 7: Vacuuming Floors
15 December 1993

8.22. IN-SITU DECONTAMINATION OF THE TUBE BUNDLE FROM A STEAM GENERATOR OF THE DAMPIERRE PWR AND SUBSEQUENT WASTE TREATMENT

Contractors: Framatome, EdF, CPN Dampierre
Contract No.: FI2D-0069
Work Period: January 1992 - April 1994
Coordinator: G DORIMINI, Framatome, Chalon s/S.
Phone: 33/85 96 30 76 Fax: 33/85 96 35 54

A. OBJECTIVE AND SCOPE

The present work concerns the "hard decontamination" of a tube bundle installed in a steam generator (SG) removed earlier from the DAMPIERRE 900 MWe PWR, in order to reduce significantly the radioactive dose rate before the final dismantling of the steam generator.

The SG in question is one of the original SGs designed and built by Framatome. They were removed from service in 1990 because of the large number of plugged tubes (approximately 10%) and the numerous primary-to-secondary leaks apparently due to stress corrosion cracking attack (SCCA).

The SGs have been stored since 1990 in a building designed for this purpose and the radioactivity present in each SG, at this time, is estimated to be about 100 Ci (essentially Cobalt 60). The dose rate on the outer surface of the dry SG varies from 0.05 mSv/h to 0.5 mSv/h.

The proposed decontamination process provides using a combination of nitric acid and Cerium nitrate with regeneration of the Cerium (Cerium 3+, Cerium 4+) by injecting ozone during the decontamination operation. The decontamination operation is performed at low pH and at ambient temperature.

After neutralization and precipitation of the decontamination solution containing the removed activity, the residue will be dryed in order to fix it in solid form for storage and disposal.

The objectives of the method selected to decontaminate the SG bundle made of Inconel 600 material are:

1. obtaining a high decontamination factor DF equal to approximately 1000;
2. decontamination at atmospheric pressure and at a temperature of less than 60°C because of doubts concerning tube integrity and the risk of leakage into the secondary side;
3. minimize the volume of generated secondary wastes;
4. filtration of sludge and residues generated by the decontamination process, if possible, using a standard process.

The programme will be implemented jointly between: Framatome (FRA), Electricité de France (EdF) as partners with the assistance of the plant owner Centrale Nucléaire de Dampierre (CND) and KWO, owner of Obrigheim NPP which will provide some participation.

B. WORK PROGRAMME

B.1. Definition of the decontamination and of the waste treatment process

B.1.1. Definition of the decontamination process based on the existing oxide layer thickness and the needed time to remove this oxide layer to obtain a DF of approximately 1000 (FRA).

B.1.2. Implementation of corrosion tests and laboratory studies for the assessment of the corrosion resistance of the equipment in the decontamination loop (EdF).

B.1.3. Establishing of the procedure for the in-situ decontamination (FRA and EdF).

B.1.4. Definition of the conditioning of effluents and waste treatment arising from the decontamination of the SG bundle and of shielding and protection requirements (EdF and CND).

B.2. Procurement and adaptation of equipment

B.2.1. Adaptation of existing equipment formerly used with an AP Citrox process (EdF)

B.2.2. Fabrication of the additional decontamination equipment, taking into account the characteristics of the provided decontamination unit, followed by commissioning testing (EdF, FRA)

B.3. Decontamination operations and liquid waste treatment

B.3.1. Preparatory operations on site, including preparation and qualification of operating specifications and procedures (FRA, CND)

B.3.2. Implementation of the decontamination operation (FRA)

B.3.3. Liquid waste treatment and conditioning (FRA, EdF)

B.4. Analysis of results including the estimation of the residual activity and the definition of an industrial-scale decontamination procedure (all)

B.5. Generation of specific data: Specific data on costs, worker exposure, working time and waste arisings will be derived from the execution of items B.2. and B.3. (all).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

The work carried out during the first half of the year essentially concerned the finalization of the design and changes to bring into the Alkaline Permanganate Citrox loop, to assure its compatibility with the new ozonation system, the suction cup and the steam generator (SG) storage building. The possibility of a new ozonation system was studied with the Trailigaz Company, to ensure the production of 1500 g/h ozone, and to optimize the dissolution in the liquid.

The remainder of work performed during the second half of 1993 essentially concerned the preparatory operations on site, including preparation and qualifications of official documents requested by French authorities for an ICPE (*Installation classée de protection et d'environnement*") facility.

Progress and results

Definition of the decontamination and waste treatment processes (B.1.)
Effluent conditioning and treatment (B.1.4.)
The effluent treatment was subjected to changes, following conclusions of the economic and technical analysis and waste products to be treated.
It was finally decided to evacuate the effluents to the CEA for treatment.

Procurement and adaptation of equipment (B.2.)
Manufacture of additional equipment (B.2.2.)
Additional modifications made on the loop related to:

- the new system for ozonation production, control and destruction
- the liquid waste storage and the suction cups (Fig.1).

Ozonation system

Framatome designed and manufactured a special vessel to efficiently dissolve ozone in liquid (Fig. 2).

Liquid waste storage and filtration

New preconfined filters were chosen by EdF to minimize the radiological impact

Suction cups

To limit the total dose, it was first necessary to limit the decontamination to only three lots of 336 SG tubes each, then to develop a new concept to attach the suction cup on the tube sheet.
The actual dosimetric estimation made for the use of this suction cup with the new attachment system and recycling of the decontamination solution, for three lots of 336 tubes each, with a preconfined filter and liquid waste evacuated to the CEA, is about 16 rem (or 0.16 Sv).

Decontamination operations and liquid treatment (B.3.)
Preparating operations on site (B.3.1.)

SG storage building

EdF and CND carried out considerable work in the SG storage building to install the decontamination loop and other equipments: retention pit, platform, shielding, anticorrosive protection on the ground and an access hatch.

SG mock-up

To be able to validate the suction cups and the procedure, Framatome performed a training programme and fatigue strength measurement, using the SG mock-up at Chalon/Saône.
An extensive training programme was carried out in the contractor's manufacturing training centre for installing the suction cups in the primary channel head of steam generator (Fig. 3).

DAMPIERRE STEAM GENERATOR TUBE BUNDLE HARD CHEMICAL DECONTAMINATION

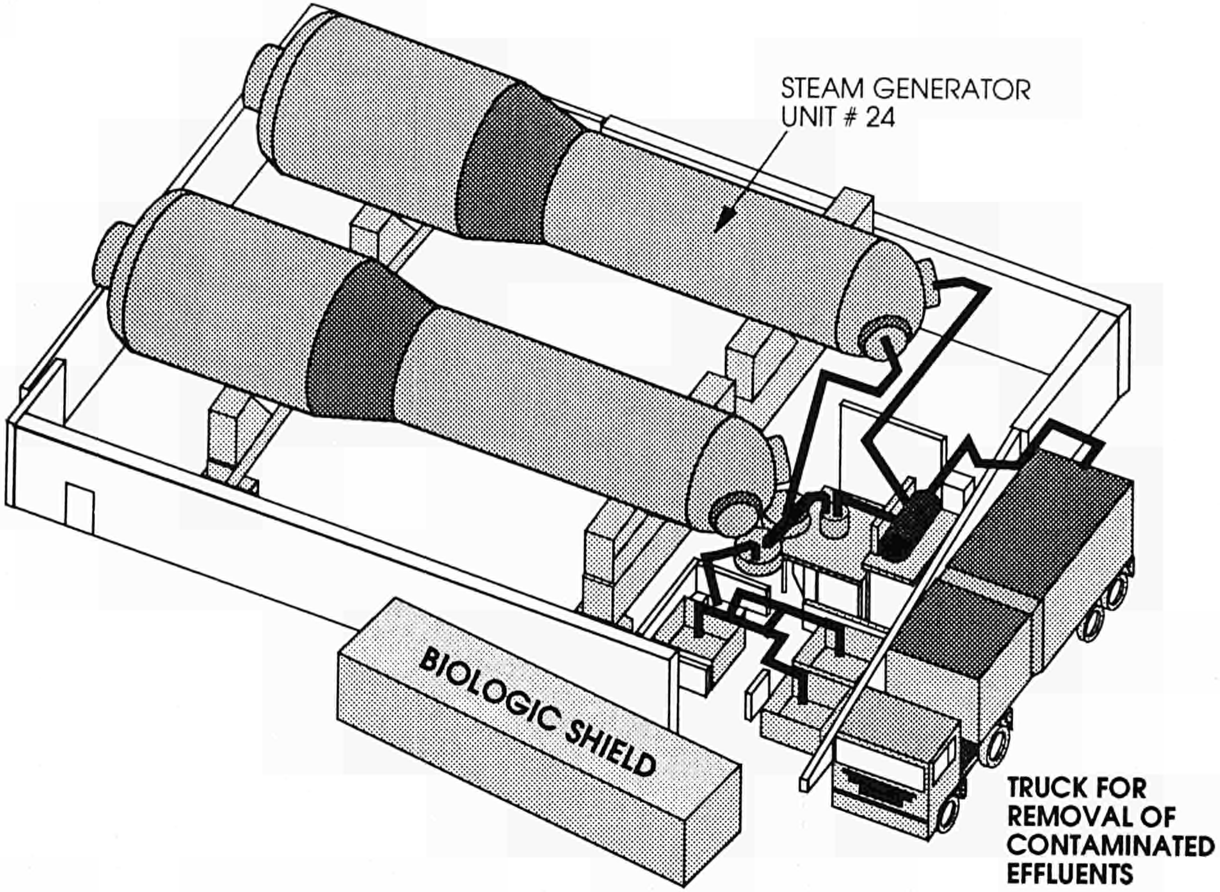


Figure 1.

Figure 2. **Ozonation system**
Ozoner (blue cover) and liquid oxyden storage tank

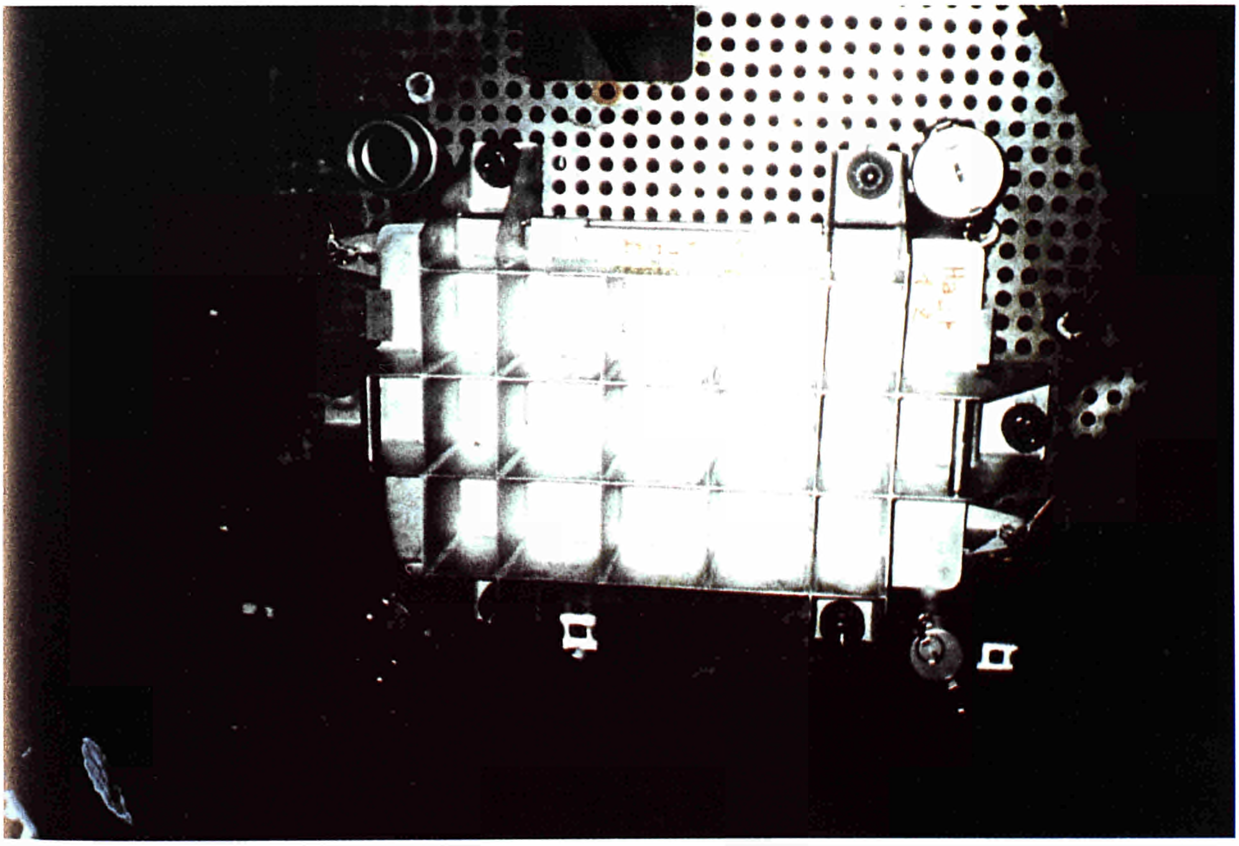


Figure 3. **Isolation suction cups**

8.23. DECOMMISSIONING OF THE DRY GRANULATION PLANT USING MACHINE ASSISTANCE

Contractors: BNF plc Sellafield
Contract No.: FI2D-0071
Work Period: December 1991 - June 1994
Coordinator: B ROSE, BNF Decommissioning
Phone: 44/9467/75480 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

Summary of Main Issues

During 1993 work proceeded on Work Packages B2 Detail Design, Planning and Safety Studies, B3 Delivery and Commissioning of Remote Equipment, and B4.2 Manual Decommissioning of First Floor Area.

The programme was revised in view of continuing development of the remote equipment and a decision taken to deploy the machine at ground floor only because of acces problems.

Detail design and manufacture of the Remote Handling Machine (RHM) was completed to programme and currently works testing is continuing, prior to the machine being delivered to the test area for inactive testing (B3).

A tooling system for remote cutting operations based on a circular saw has been developed separately and is being manufactured at present.

Investigation of plasma arc cutting has continued as part of a parallel development programme and, dependant on progress, will be tested on the RHM.

Safety Studies are complete but work continues with approval procedures with regulatory bodies.

Installation of the containment and ventilation systems for first floor decommissioning has started, as part of area preparation for B4.2 Manual Decommissioning First Floor.

Table I : Work Programme

WORKING PHASE	EARLIEST START	EARLIEST FINISH
B1 STUDY OVERALL DECOMMISSIONING SCHEME	2 - JAN - 92	30 - SEP - 92
B2 DETAIL DESIGN, PLANNING AND SAFETY STUDIES	30 - SEP - 92	31 - OCT - 93
B3 DELIVERY AND COMMISSIONING (INACTIVE TESTING)	31 - OCT - 93	31 - MAR - 94
B4.1 DEPLOYMENT AND ACTIVE OPERATION AT GROUND FLOOR (ACTIVE OPERATIONS BY SEP 94)	31 - MAR - 94	31 - MAY - 95
B4.2 MANUAL DECOMMISSIONING FIRST FLOOR	28 - SEP - 93	7 - JUL - 94
B5 & B6 GENERATION AND EVALUATION OF SPECIFIC DATA	31 - OCT - 93	31 - MAY - 95

Progress and Results

1. Planning and Safety Studies (B2)

1.1 Refer to revised Work Programme Table I for details. A review of strategy earlier in the year indicated that dose reduction benefits in using remote methods were outweighed by access and installation problems at first floor level. It was therefore decided to employ the RHM at ground floor level only, where improved access will allow a clearer assessment of remote methods. This provides an opportunity to compare directly manual operations on the first floor with manual/remote operations on the ground floor.

1.2 Because operations at first floor level are manual, they can proceed independently of remote equipment development. Work on the first floor has therefore been brought forward and area preparation commenced in November.

1.3 Safety Studies are complete and submissions being made to the regulatory bodies for agreement to proceed. Of particular interest to this report is an assessment of the risks of using hydraulics in a plutonium environment, which will impose operating rules on the use of the RHM in the active area. This will form part of an application for agreement to install and commission the RHM in the active area.

2. Detail Design (B2)

Refer to Figures 1 and 2

2.1 Detail design and manufacture of the RHM were completed to programme in October. The machine consists of a carrier system (or deployment machine), manipulators, machine and manipulator control systems, hydraulic power pack and a television viewing system. Works testing is currently in progress, principally on the manipulators and their control system, in readiness for relocation of the machine to the test area for inactive testing(B3).

2.2 The tooling system selected for use with the RHM is based on a circular slitting saw. The cutting tool, which is handled by the manipulator, consists of an electrically driven rotating blade attached to a plunge arm. This can be mounted on a range of modules that hold the tool in place during cutting operations. These are a magnetic base, claw base and a track with carriage and magnetic feet. Use is made of special alloys to reduce weight. Tool system development was not at a suitable stage during RHM specification to incorporate it into the machine control system and the tool therefore was designed with an independent control system. One of the principal aims of the inactive testing (B3) will be to combine the operation of these to produce a working system for remote cutting operations. Manufacture of the tooling system is proceeding at this time.

2.3 Investigation of remote plasma cutting techniques has been the subject of a continuing development programme. The specific needs of this project are being addressed with a view to carrying out development and testing of plasma cutting with the RHM. This includes solving the problems of controlling the manipulator to provide the required motion, handling of the torch and its power and air supplies and control of fume.

2.4 Provision of other equipment for the decommissioning facility, principally containment, ventilation, waste drum handling and structural equipment, is proceeding to programme. Technical problems with the design of a plutonium piece assay monitor may cause delay to first floor manual work.

3. Delivery and Commissioning of Remote Equipment-Inactive Testing (B3)

3.1 Provision of a mock-up of the ground floor glovebox and cell is complete, awaiting installation of the RHM on completion of works testing.

3.2 A Test Schedule details the objectives of the tests, which broadly are:

- to bring together the separate elements of the deployment machine, manipulators and tooling, each with their own control systems, to produce a complete working system.
- to develop practical cutting and waste handling techniques.
- to develop a sequence of dismantling operations on the mock-up glovebox as a rehearsal to active operations.
- to train operators in the use and maintenance, installation and commissioning of the RHM.
- to obtain regulatory body approval for installation and commissioning in the active area.

4. Manual Decommissioning First Floor (B4.2)

4.1 Practical operations began in November with preparation of the area for decommissioning. A large proportion of low level waste has been removed from the exterior of the gloveboxes at both floor levels. Installation of the containment is continuing around the first floor gloveboxes. Preparations for installation of the containment ventilation system that will serve both floor levels is underway. Cyclone/bag filters and a fan unit for plasma fume collection, a shower entry unit and new access at first floor level are being installed. This work will proceed independently of RHM development and testing. Results of decommissioning operations in terms of duration, dose uptake, etc will be compared with remote operations on the ground floor.

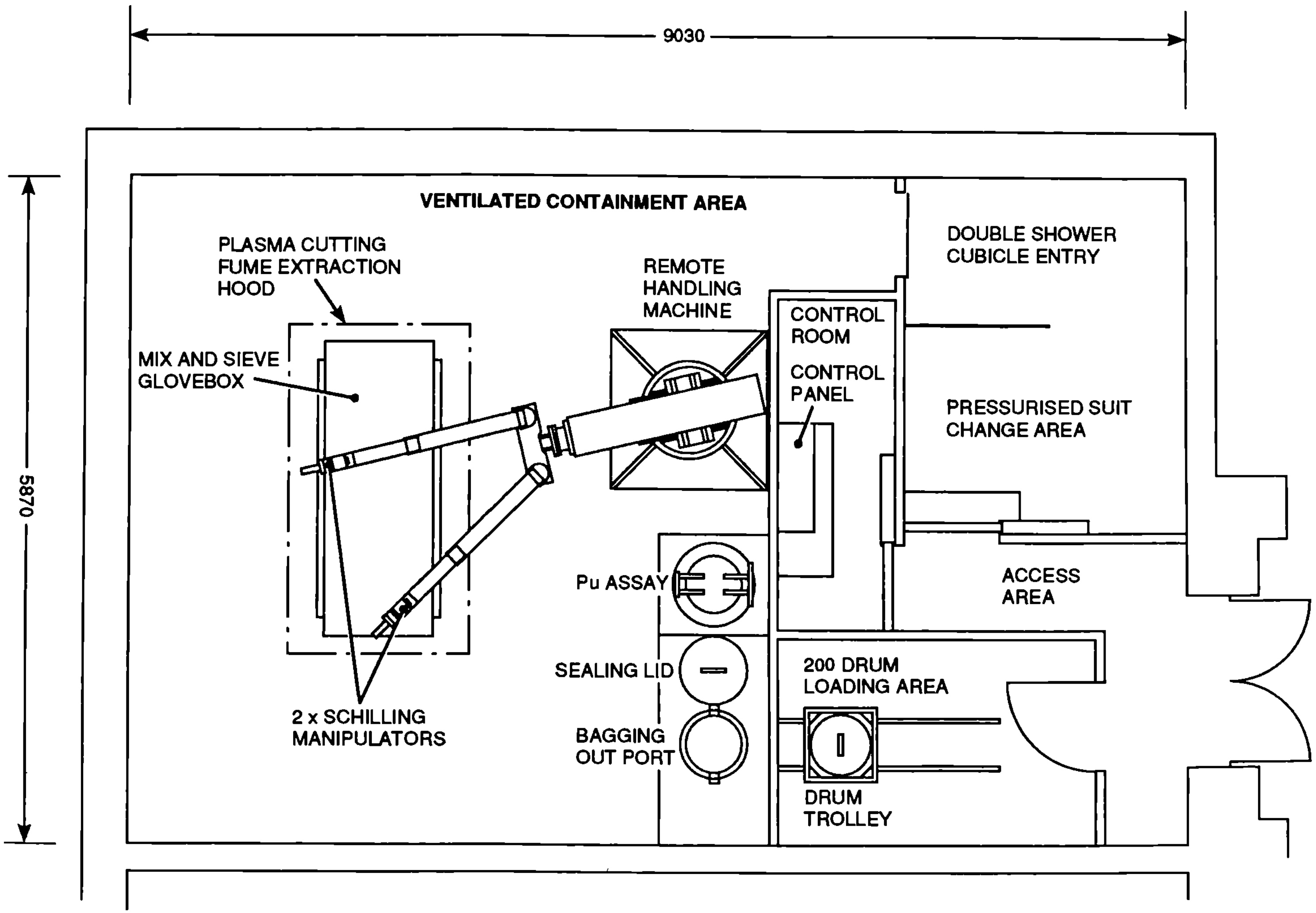


Figure 1 Ground Floor Plan

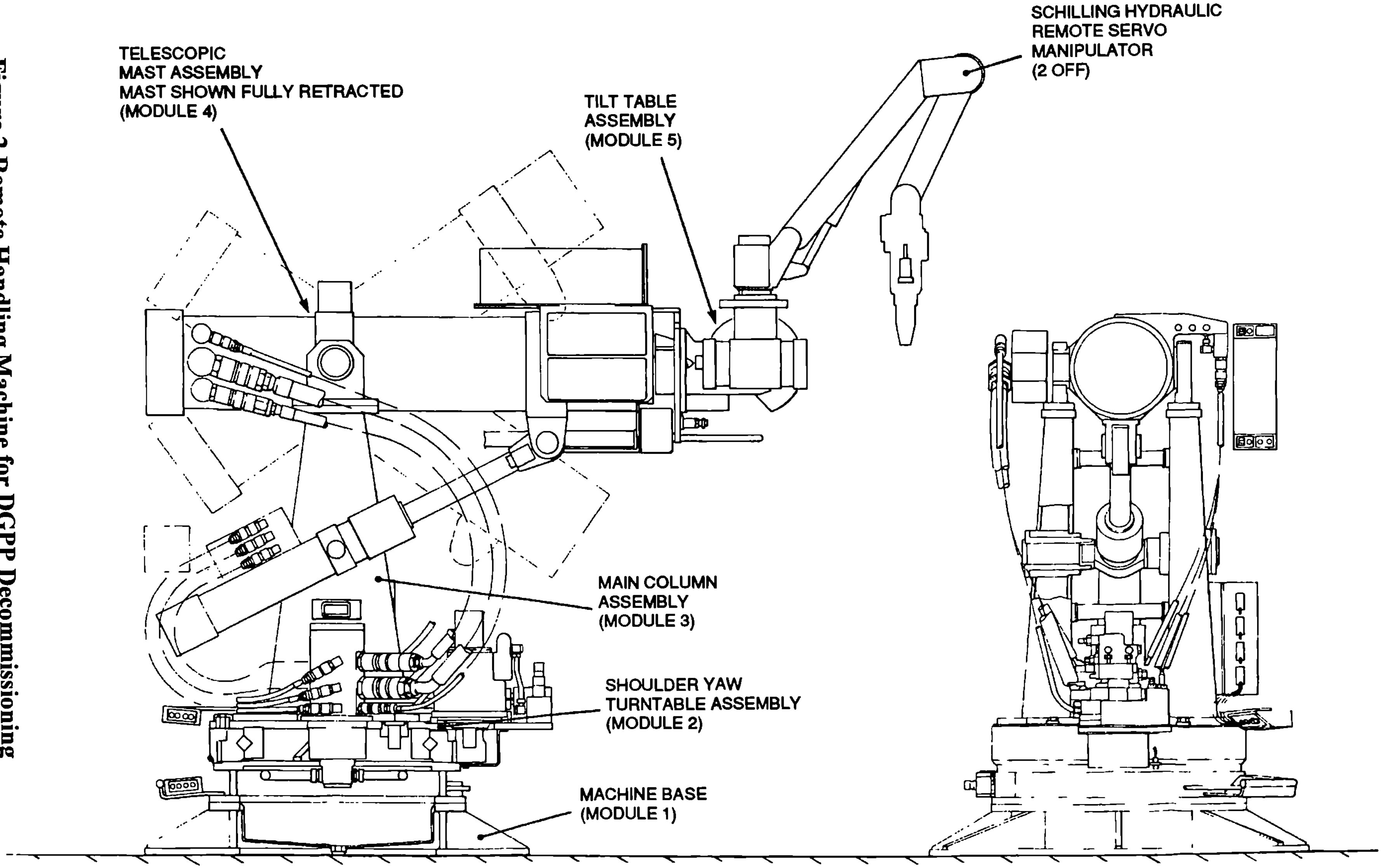


Figure 2 Remote Handling Machine for DGPP Decommissioning

8.24. REMOTELY OPERATED UNDERWATER DISMANTLING BY PLASMA ARC AND CABLE SAW OF A STRUCTURE IN A COGEMA STORAGE POND

Contractors: COMEX NUCLÉAIRE, COGEMA, DIAMANT BOART
Contract No.: FI2D-0072
Work Period: January 1992 - December 1994
Coordinator: J BLIGHT, COMEX Nucléaire, Marseille
Phone: 33/91 29 13 00 Fax: 33/91 25 30 39

A. OBJECTIVE AND SCOPE

The objective of this project is to demonstrate the radiological, technical and economic feasibility of using divers to install remotely controlled cutting and decontamination equipment in a nuclear environment.

The demonstration consists in the underwater dismantling by plasma arc or mechanical tools of a number of stainless steel anti-shock mattresses installed in the bottom of the Pond N° 900 at COGEMA, La Hague after initial decontamination. The specific radioactive inventory is in the order of $3.7.10^3$ Bq/l, specific contamination is about 200 Bq/cm² beta and the dose rate is estimated in the range of 1-30 mSv/h. The contractual work will consist of the preparation of equipment and procedures followed by an industrial-scale dismantling operation.

The project will provide the information required to judge whether the use of divers represents a technical and economically viable alternative to either "dry" manual or telemanipulation techniques for the decommissioning of nuclear installations and will permit a definition of the conditions in which the use of divers would represent the optimal solution. The project will also result in a validation of diving equipment and procedures for nuclear environments.

The results of the study should be used for both decommissioning and maintenance operations for nuclear power generation and fuel reprocessing plants. The techniques to be demonstrated could be used for installations which are normally immersed or which could be immersed specifically to permit the use of divers during maintenance or decommissioning. Typical decommissioning tasks would include decontamination, dismantling and assistance during removal.

Water is an effective biological shield and as such, intervention under water will permit a reduction in dose uptake when compared to manual intervention in an air atmosphere. Real time monitoring systems will ensure that the diver is subjected neither to contamination nor to excessive dose rates.

The project will be carried out by COMEX NUCLÉAIRE (CxN), COGEMA and DIAMANT BOART and will produce specific data on subjects such as dose levels (individual and total) and times and costs for specific tasks, which can be used in the planning of larger-scale decommissioning operations.

B. WORK PROGRAMME

B.1. Preparatory work

B.1.1. Modification of diver equipment, including safer leak-tightness and provision of real-time dose rate measurement with on-line calculation of integrated job dose at exposed parts of the diver, review and upgrade of existing procedures and regulations for divers in ionizing environment.

B.1.2. Implementation of comparative underwater cutting tests on flat plates with plasma arc and mechanical sawing, including the assessment of systems for collection of cutting waste and for an easier decontamination (CxN).

B.1.3. Assessment of cable sawing for metallic structures, also aiming at an application to components of the KRB-A reactor (CxN, Diamond Boart).

B.1.4. Development and testing of systems for the decontamination/cleaning of walls and bottoms of the storage pond by jetting or brushing (CxN).

B.1.5. Preparation of documents for the licensing and operation of all equipment at the storage pond for approval by COGEMA (CxN, COGEMA).

B.2. Execution of the dismantling work including the installation of the equipment, radiological mapping and pre-decontamination of the pool, and the conditioning and removal of dismantling waste and final decontamination of the pond (CxN, COGEMA).

B.3. Evaluation of results obtained with equipment for divers and for applied tools and procedures (all).

B.4. Generation of specific data on costs, radioactive job doses, working time and secondary waste arisings from the execution of items B.1., B.2. and B.3. (all).

C. PROGRESS OF WORK AND RESULTS OBTAINED

Summary of main issues

During this period, an agreement was obtained from COGEMA to relocate the pilot operation in Marcoule. Consequently, the end of the contract was postponed by one year to cope with the delay generated by the change of site.

Progress and results

Preparatory work (B.1)

1. Dose monitoring system (DMS) (B.1.1)

The development of the software programme was delayed due to the slow response time of the underwater sensors fixed on the diver body. This task is now completed, and the system tests in laboratory meet the initial requirements in terms of sensitivity and detection of the dose integrated by the diver.

In order to cope with an extension of the diver umbilical length, the original serial link between the multiplexing box positioned on the diver chest and the surface has been replaced by a more reliable system; this modification allows the transmission of the data over more than 60 meters (figure 1).

The integration of the different elements: multiplexing unit, interface board, computer, printer and power supply in a portable case is now almost completed.

The commissioning of the DMS is planned for February 1994 in Marcoule, and the tests in an inactive pool in Marseille during March. The final test in real conditions will be carried out in Marcoule during the pilot operation.

2. Comparative underwater cutting tests on flat plates (B.1.2)

The fabrication of the test bench is now completed.

Some modifications on the arc saw were performed during the last period, including the addition of a cooling system inside the saw body to avoid an excessive dilatation of the collector shaft provoked by the friction of the carbon brushes on the driving shaft.

The water motor driving the blade was replaced by a low speed high torque motor to reduce the wear of the carbon brushes.

The feed control of the saw, piloted by the arc voltage established between the edge of the blade and the plate to be severed, was tested and is working satisfactorily.

3. Cable sawing of metallic structures (B.1.3)

Additional cutting tests using a cable made of 80 beads per metre, were carried out on a mock-up, consisting of 5 stainless steel tubes (diameter 6 inches, wall thickness 8 mm).

In wrap-around mode, the cable jammed after 30 minutes of cut, on some sections of the cable the plastic spacers injected between the beads were stripped off; this leads to a rapid deterioration of the cable when the beads start sliding on the supporting cable.

From these series of tests can be concluded that the cable using beads even regularly crimped by a metal sleeve on the supporting cable, is not resistant enough when cutting hollow metallic structures in the wrap-around mode.

A prototype of the continuous diamond cable was tested in the Diamant Board facilities. The cable is made of a metallic ribbon on which the diamonds are embedded in a galvanic deposit, the ribbon is spirally wound around the supporting cable. Extremities of the cable are jointed by a crimped junction piece.

The cable was tried on a test-piece made of Inconel. The cutting efficiency of the cable is correct but the best was aborted after 30 minutes when the junction piece collapsed. To increase the reliability of this coupling is the next objective of Diamant Board before continuing the evaluation of this new concept.

4. Decontamination of pool walls (B.1.4)

The filtration unit associated to the decontamination rotating water jet is now completed.

The filtration system consists of two stainless steel vessels connected in serial and containing the filter cartridges; to avoid a risk of irradiation, the unit is standing on the pool floor during the operations.

The water containing the contaminated sludge removed from the walls is cleansed through the filters. Filtering is performed in two stages, compartment 1 is a prefilter, compartment 2 retains particles down to 0.5 μm in size.

A volumetric pump located in surface is incorporated into the loop in order to create the required flow of 20 m/h. The filtered water is then returned to the pool (figure 2).

A pressure gauge on the pump indicates when the filter elements are saturated.

A radiometric probe, in contact with the filter vessel, monitors the dose rate of the sludge gradually accumulating in the filter cartridges. To limit the dose integrated by the diver, the recovery of the filter elements and their transfer in a lead contained is performed from the surface by means of long handling rods. The filter and prefilter vessels are fitted with a hinged cover which opening is controlled from the surface by a pneumatic cylinder.

The filter element is a soft bag made of synthetic cloth which opening is fixed in a flange made of PVC.

The whole system was successfully function-tested under water by divers in inactive conditions in the pools of Comex. The system will be used for the pilot operation.

5. Preparation of documents for the pilot operation (B.1.5)

This operation is scheduled at Marcoule for mid-1994 and consists in the three underwater fuel decanning units. The decanning machines were part of the preparation chain of the fuel elements from the EdF GCR reactors. They are installed in three pools referred to as B1, B2 and C2.

Each pool is 3.5 m x 3.5 m square and 5.1 m deep. The walls and floor are covered by a 4 mm thick stainless steel liner. A part of the decanning operations were carried out in a shielded cell existing above one of the pools.

The floor of B1 is covered with a layer of sludge made of graphite powder and contains a large number of segmented cartridge tips. Some fuel elements remain in the pool.

In B2, the sludge covering the floor is made principally of magnesium oxide, and as for B1, several fuel elements remain on the floor or are wedged inside the chassis of the machines.

Quantity of waste

The quantity of metallic structures to be dismantled inside the pools is estimated at 18 m^3 and that of sludge at 12 m^3.

The scope of work includes: the dismantling and evacuation of all machines, their chassis and the top cell, the recovery and evacuation of the fuel elements remaining inside the pools, the removal of the sludge and the decontamination of the walls and floors.

The duration of the complete operation is estimated at 8 months.

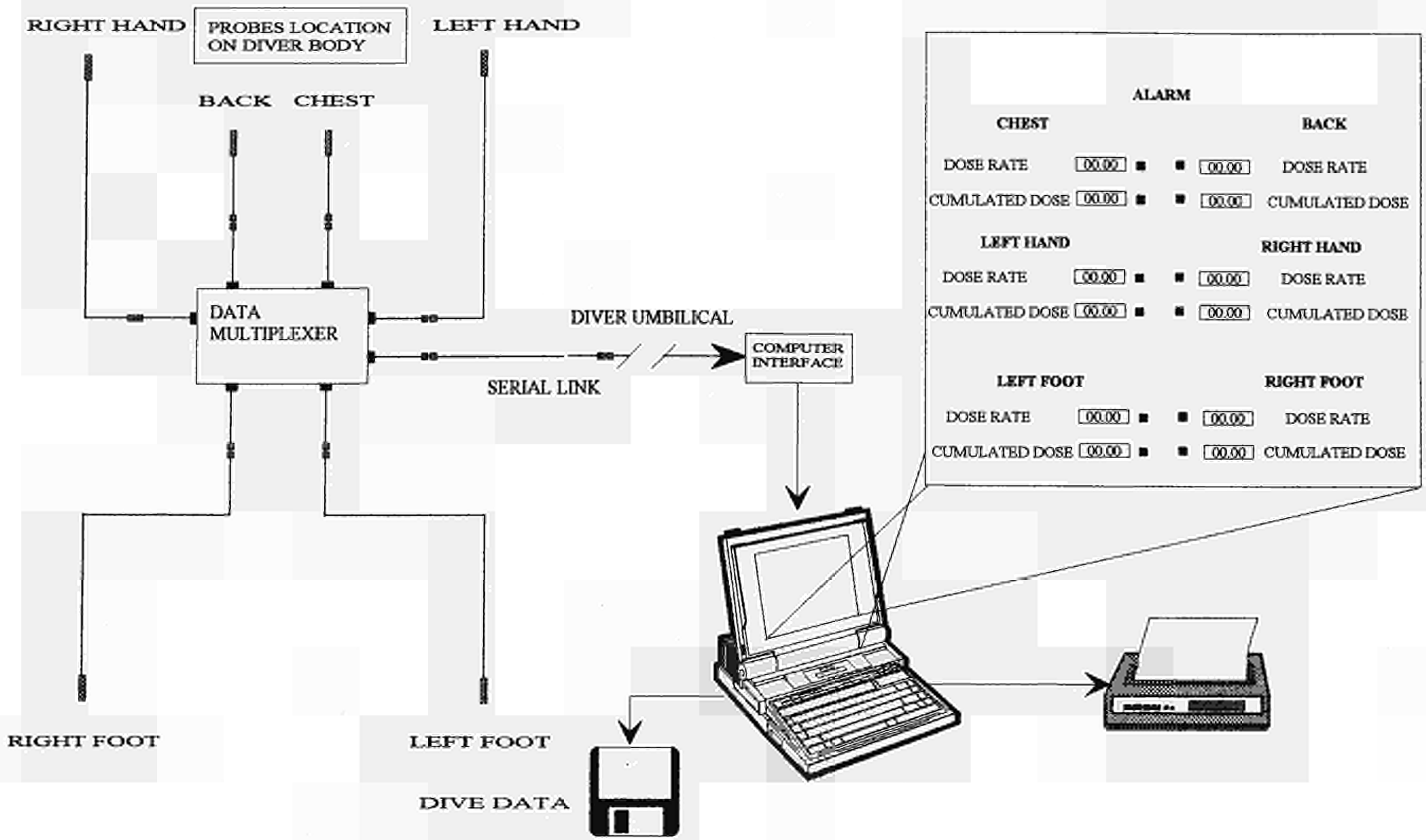


FIGURE 1 DOSE MONITORING SCHEMATIC DIAGRAM

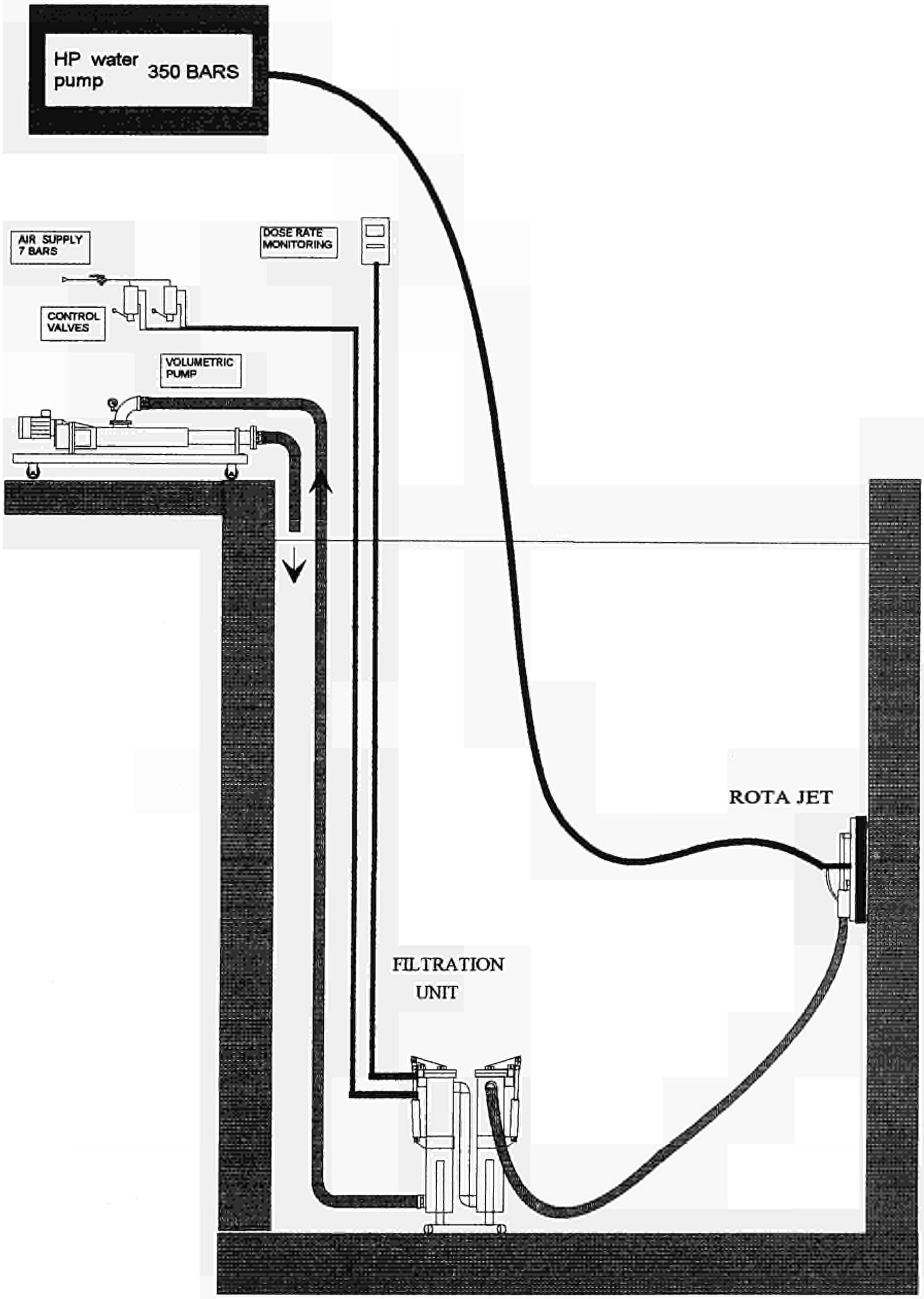


FIGURE 2 DECONTAMINATION SYSTEM

8.25. FURTHER DEVELOPMENT OF A DATA BASE ON CUTTING TOOLS AND ASSOCIATED FILTER SYSTEMS FOR DISMANTLING (Joint study)

Contractors:	Uni. Hannover, CEA-Valrhô, AEA Windscale
Contracts Nos.:	FI2D-0056, -0057, -0058
Work Period:	October 1991 - December 1994
Coordinators:	1) A RAHLFS-MARHOLD, Uni. Hannover
	2) J M DUFAUD, CEA-Valrhô
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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

Retaining the internal data base structure, the user interface application has been improved. It has a modular structure to facilitate supplements and changes. Using the Oracle tool SQL-menu it is now possible to combine data input, system functions and report possibilities via pull-down menus.

The data gathered from former and current EC R&D programmes and literature is being analysed and prepared for input.

To get reference data of institutions dealing with equipment etc. independent from special cutting tasks, a general reference sheet has been added.

The data collection will be continued to the end of 1994 to include the EC-contracts that are prolonged until end '94.

A public relation brochure has been prepared to give an overview of the EC-DB-TOOL possibilities.

Progress and results

1. Development of the data base structure (B.1)

The basic concept and the internal structure of the data base as the base of EC-DB-TOOL application development is complete. The user application has been improved. Supplements and modifications are simplified by means of a modular structure. This way the maintenance and adaptation in case of software updates can be executed favourably. Using the Oracle tool SQL-menu it is now possible to combine different Oracle tools for data input, report possibilities, storing and retrieval of data and system functions by means of pull-down menus. The data base system allows to assign special user permissions to ensure security aspects, e.g. to use the system administration facilities as adding or deleting a user it is necessary to have the authorization of these tasks.

The definition of the data sheet for general reference information in the data base application, independent from special decommissioning tasks, has been finished (Fig.3).

These changes of the application system are independent from the basic tables storing the data, and cause no loss of data collected.

2. Collection of new data (B.2.,B.3.)

The data collection is continuing. Data of technical reports from former and current EC R&D programmes and literature reviews are analysed. This includes a significant quantity of material from the USA.

The new data questionnaires were distributed (Fig.2) and a new general reference sheet is added (Fig.1). The upper part contains information on the institution e.g. name, address and media connections. The second part lists in detail the subjects associated with the possibility of naming a contact person for the special subjects. There is also the possibility of adding new subjects and to give comments concerning the institution. With this sheet information can be collected independently of a special dismantling task, so that e.g. information on project managers planning a new cutting task can be hold before the actual start of the dismantling.

A supplementary screen mask of EC-DB-TOOL makes this information available for planning tasks.

The new general reference sheet has been sent out to a number of project managers and experts and the returning-rate showed that the acceptance has increased.

To present the data base project to a wider public, a public relation brochure has been prepared and issued.

Its first part gives a general introduction of the background and the aims of EC-DB-TOOL. The reader is given a first impression on the possibilities and usefulness of the data base such as

- Structured Storage of Decommissioning Data
- Report Generation
    - standard
    - custom
- Technical References
- Up to Date Information on Decommissioning Technology
- Contacts

The other parts describe the sources of data, the data base structure, the user interface application, design strategies and the hardware requirements.

The data collection will be continued until the end of 1994 to include the EC-contracts that are prolonged until end '94. As it appears to be more difficult than expected to accomplish the data collection especially to retrieve filled in data sheets, the building of a data acquisition team was started to introduce the data sheets to investigators and experts.

3. Updating, treatment of collected data and incorporation into the data base (B.4)

The data sheets returned (16 tests with 51sheets in the german area) have been analysed and incorporated into the data base.

GENERAL REFERENCE

Institution:

Name: ....
....
Street, No.: ....
Postal code, town: ....
Country: ....
Phone: .... fax: .... telex: ....

Subject: **Contact person:**

Water Jet Cutting	[..]	....
Flame Cutting / Gouging	[..]	....
Laser Cutting	[..]	....
Plasma Arc Cutting	[..]	....
Consum. Elec. Water Jet Cutting	[..]	....
Mechanical Cutting	[..]	....
Electro Discharge Machining	[..]	....
Electrochemical Cutting	[..]	....
Dismantling Task	[..]	....
Air Filtration	[..]	....
Aerosol-Measurement	[..]	....
Water Filtration	[..]	....
Hydrosol-Measurement	[..]	....
Handling Devices	[..]	....
....	[..]	....
....	[..]	....
....	[..]	....
....	[..]	....

comment:

Fig.1: General reference data sheet

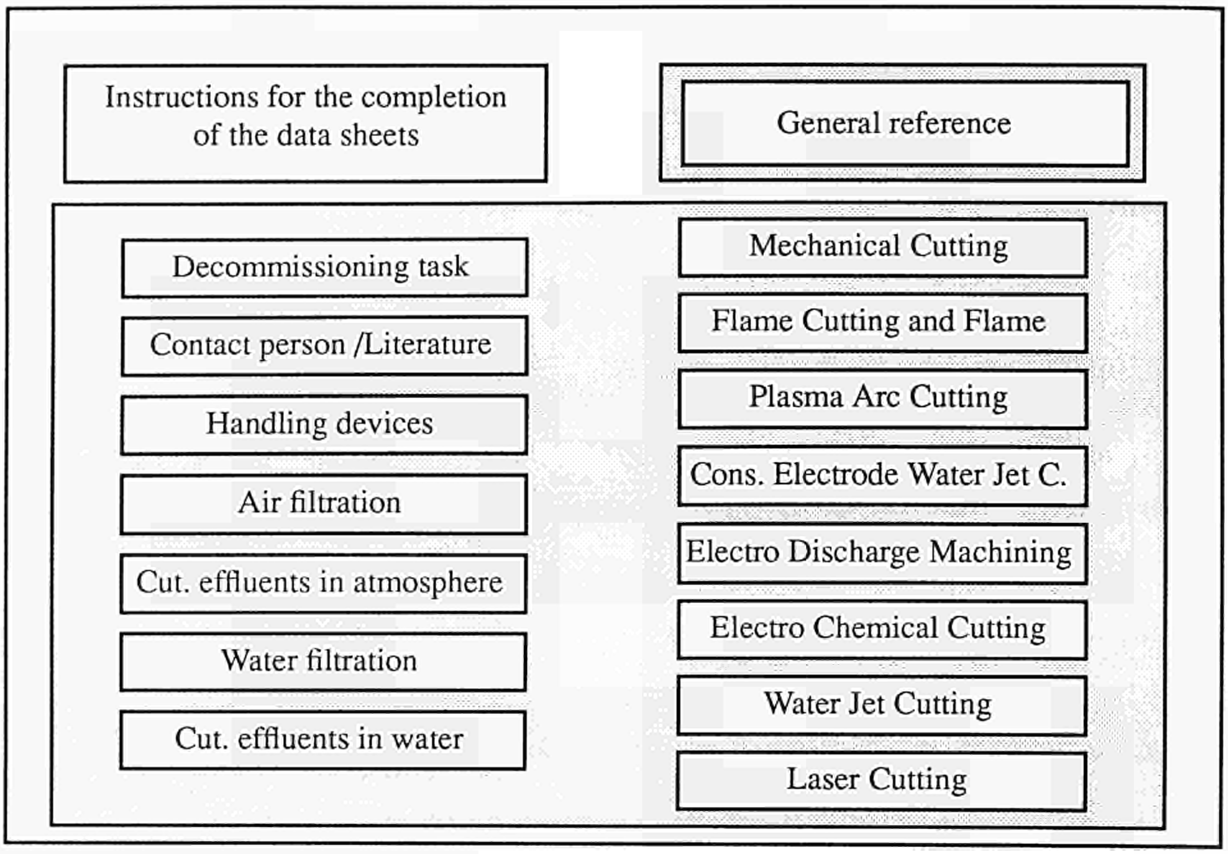

Fig.2: List of available data sheets

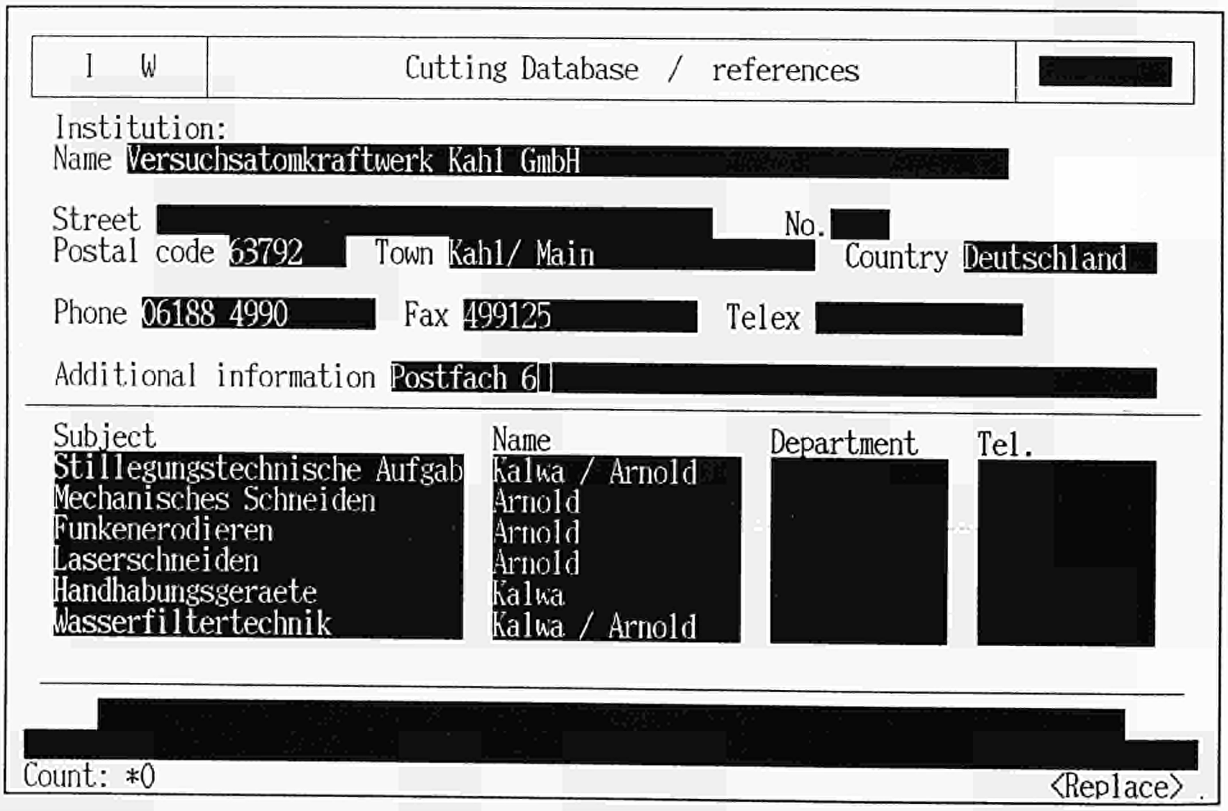

Fig.3: Screen mask for general references

8.26. COLLECTION AND TREATMENT OF SPECIFIC DATA GENERATED IN LARGE-SCALE DISMANTLING OPERATION WITH A VIEW TO THEIR USE FOR COST AND DOSE ESTIMATES (Joint Study)

Contractors:	NIS, CEA-Valrhô, BNF
Contracts Nos.:	FI2D-0059, -0060, -0061
Work Period:	October 1991 - June 1994
Coordinators:	1) P PETRASCH, NIS 2) J M DUFAUD, CEA-Valrhô 3) S WHITTY, 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 66 85
	3) 44/9467/74575 Fax: 44/9467/74070

A. OBJECTIVE AND SCOPE

Taking advantage of the orientation of the present R&D programme to large-scale dismantling operations in different types of installations and considering the need to dispose of data obtained under realistic conditions for cost calculations concerning large-scale dismantling operations, the CEC supports work concerned with the collection and treatment of specific data on costs for unit operations, on associated radioactive job doses, on working time and on waste arisings to be derived mainly from all contracts in Section C, where the generation of specific data is a mandatory task (particularly from pilot projects).
Work will take advantage of the methodology developed by NIS and CEA in former joint research contracts FI1D-0074/75 (EC programme 1984-88).

B. WORK PROGRAMME

B.1. Development of an EC-user-friendly data base package based on commercial software for data storage and processing (NIS assisted by the partners)
B.2. Collection and analysis of data on costs and working time for unit operations, on radioactive job doses and on waste arisings generated during former and in current EC contracts (mainly in Section C and Pilot projects) in Germany and elsewhere (NIS)
B.3. Collection and analysis of relevant data not considered under B.2. and generated in France (mainly data related to GCRs of UNGG-type and fuel cycle installations) and from Japan (CEA).
B.4. Collection of other relevant data not considered under B.2. and generated in the UK (mainly data related to AGRs, Magnox and fuel cycle installations) and from the USA and Canada (BNF).
B.5. Continuous updating; treatment and incorporation into the data base of all collected data (NIS) including identification of relevant cost indexes.
B.6. Definition of a self-supporting system for continuous and systematic updating of the data base after the end of the present contract (all).

C. Progress of Work and results obtained

Summary of main issues

The commission of the European Communities initiates and promotes research programmes on the decommissioning of nuclear installations since 1979.
The large number of research projects includes a host of experience and data usable on further decommissioning projects.

For saving this experience the EC commissioned the development of the database "EC DB-Cost" to collect and evaluate data relating to decommissioning cost and radiation exposure occurring during the decommissioning activities.
The main issues on the EC DB-Cost development in 1993 are:

- development of the EC DB-Cost using the ORACLE database system
- collection of decommissioning data from EC-Contractors
- collection of decommissioning data out of the EC research programme
- development of a user concept for updating and continuous enlargement

Progress and results

B.1. Development of an EC user-friendly data base package based on commercial software for data storage and processing.

The EC DB-Cost is developed and installed on a PC type 486 using the ORACLE data base system. The EC DB-Cost is ready to work. Today the outputs of the EC DB-Cost results are under development.

Also under development are an evaluation and a presentation of comparable decommissioning data. This kind of presentation allows a survey of data on working steps with comparable tasks in different countries or decommissioning projects.

Now the EC DB-Cost includes the following parts:

1. Working step information
   - description of working step
   - publication
   - working data (manpower, duration, dose, cost, waste management data)

2. General data on decommissioning
   - survey of decommissioning projects
   - national boundary conditions
   - national waste management concepts
   - economical data of the several countries

3. Specific data, related to a reference value (e.g. mass, volume, surface)
   - cost
   - manpower
   - occupational radiation exposure
   - waste management data

4. Thesaurus system and numbering code
   - organisation of EC DB-Cost
   - structure of decommissioning projects

5. Output and reports (not finished)
   - results of single working steps
   - results of groups of working steps

B.2. <u>Collection and analysis of data on cost and working time for unit operations, on radioactive job doses and on waste arisings generated during former and in current EC-contracts.</u>

The collection of data generated in the current EC research projects has not been very successful. In a first data collection campaign data form sheets were sent to the EC contractors in 1992. Only one data sheet from Germany and 12 data sheets from UK were returned.
In a second 1993 campaign some data sheets with previous data, generated on the half-year reports of the EC-Contractors were prepared. These contractors would asked to fill in the empty fields in the data form sheet. No data sheet has been coming back up to now.
Additionally a public relation brochure was prepared to resu..t the database and to inform interested users. This brochure was distributed to all EC-Contractors and other decommissioning projects in the EU.

Now in the third campaign decommissioning data coming from publications, theoretical studies and any other data sources is being collected.

B.3. <u>Collection and analysis of relevant data not considered under B.2. and generated in France and from Japan (CEA-UDIN).</u>

After the analysis of the different aspects of the data collection sheets, a study was launched on the setting up of sheets about the decommissioning at stage 2 of Fast Breeder Reactor RAPSODIE. This dismantling project is in process of completion, which allows to have a general view on dismantling work and on the total cost of operation. More, the personnel, who took part in the dismantling is still there and can answer technical enquiries.

The data collection sheets, are now being analyzed in order to put forward the difficulties that were met, and to possibly propose improvements.

The ORACLE software with the adequate material has been acquired by CEA-UDIN. The first function tests have started with hardware elements transmitted by Hannover University.

Moreover, a data collection has been launched on the dismantlings project of OKIS's cells at Saclay.

B.4. Collection of other relevant data not considered under B.2. and generated in the UK and from USA and Canada (BNF).

A new campaign to collect data was started. The public relation brochure (see B.2.) was distributed with a covering letter offering to visit projects to collect data. These have been sent to:

- All members of a UK Decommissioning Liaison Group, which comprises managers in the UK decommissioning community along with government and regulatory officials have been targeted in the hope that they will put pressure on their project managers to contribute to EC DB-COST.

- All UK EC Section C contractors were contacted. The reason for this is obvious.

- A copy was sent to a colleague of the BNFL USA subsidiary asking for their comments and advice on how to proceed in USA and Canada.

By distributing so widely, it is hoped to generate more interest than before.

B.5. Continuous updating, treatment and incorporation into the data base of all collected data including identification of relevant cost indexes.

Task B.5. is one of the main tasks of the EC DB-Cost system. The structure of the EC DB-Cost takes this aspect into consideration as follows:

- the EC DB-Cost includes no personnel cost factors because the personnel cost rates into the EU countries are very different with respect to the economical situation of each country. The manpower, the duration of a working step and the number of workers allow a better comparison of decommissioning data.

- other cost values, i.e. machine cost, tools, investment and consumables are stored in the EC DB-Cost, connected to the price indexes of the EU-Countries. So it is possible to show different cost values related to a base data and to update this cost data in intervals.
  The relevant cost indexes (and radiation exposure) will be performed by the specific data in the EC DB-Cost part 3. This data is directly usable for cost and exposure estimations for further decommissioning works.

B.6. Definition of a self-supporting system for continuous and systematic updating of the data base after the end of the present contract.

Different models for an users group are in discussion. NIS prefers a central concept supported by mobile PC units. This model does not require EC DB-Cost group members with knowledges in data base systems.

Usually the data transfer from EC DB-Cost to a user should be in a questionnaire - report form. If a bigger questionnaire problem is given, the work could be done by the customers.

Additionally a transfer of the EC DB-Cost to users is possible assuming that an ORACLE data base system exists.

The systematic and continuous updating of the EC DB-Cost will be performed by NIS supported by the users group. New data will continuously be collected, older data will be up-dated to the actual price basis.

8.27. SURVEY OF DECOMMISSIONING REQUIREMENTS FOR VVER REACTORS

Contractors: EWN, IND
Contracts Nos.: FI2D-0082
Work Period: December 1992 - February 1994
Coordinators: C FRANK, EWN
Phone: 49/38354/40 Fax: 49/38354/224 58

A. OBJECTIVE AND SCOPE

The present study should establish a sound basis to judge future R&D needs for the decommissioning of East European nuclear facilities.

Emphasis should therefore be put on VVER reactors due to large numbers of reactors of this type in Germany and several Eastern European countries, and the recognition that lack of certain safety features leads to the requirement to decommission the older type VVER 440 V 230 as soon as possible. Four of these reactors have already been closed down in Germany.

B. WORK PROGRAMME

B.1. Review of East-European nuclear facilities and selection of reference plant (IND)
B.2. Technical description of reference plant (EWN, IND)
B.3. Decommissioning of reference plant (Strategy, decontamination, dismantling procedures) (EWN, IND)
B.4. Comparison with West European experience (Strategy, decontamination, dismantling, spent fuel cycle) (IND, EWN)
B.5. Identification of R&D requirements (EWN, IND)

C. Progress of work and results obtained

Summary of main issues

A preliminary review of R & D requirements for the VVER reactor has been completed. In general, it is considered that the basic technology for complete dismantling of PWR's already exists and has been proved on present active dismantling operations on different pilot projects inside and outside of Europe. The identified R & D requirements relate to the scaling up or 'industrialisation' of this existing technology to suit the particular application to a VVER as well as requirements derived from the constructional and operational peculiarities of the VVER.

Progress and results

1. Selection and description of the reference plant

The VVER 440 design is one of the most common reactor designs within Eastern Europe (Table I).

The present CEC cooperation missions for the safety improvement of VVER's are mainly directed to the 440/213 and 1000/320 type. The VVER 440/230 is considered not to be reconstructable in an economically reasonable frame. Furthermore, a number of them will be close to their design lifetime by the turn of the century.

On the site in Greifswald (Germany) there are in total 8 reactor units of the Russian pressure water design VVER440, four of the earlier design V230 and four of the more recent V213 design. All V230 reactors have been in operation, and of the V213 one had reached start-up conditions. The more recent model V213 includes a number of changes to the V230 design generally related to improved safety requirements [1].

All units of these serial types have been constructed as double units with common reactor hall buildings in order to save certain equipments, notably in the fuel handling area. Each double unit also shares common buildings for the treatment and storage of liquid radioactive wastes, emergency power generation, ventilation system and chimney and cooling water supply. All turbines (two 220 MWe per reactor) are arranged in one turbine hall.

On site there is a central wet storage for fuel elements. This storage was erected in order to cope with changed conditions for returning the fuel elements to the former Soviet Union. There is also a central warm workshop with a small decontamination shop and the necessary mechanical equipment to perform all necessary repair work.

After the reunification of the German States, the feasibility to continue operation on the Greifswald site was investigated. Partly due to the deficiencies in certain technical systems but mainly due to decreasing electricity consumption and lack of political consensus, it was decided in 1991 to decommission and dismantle all units.

Thus, unique possibilities to study the different VVER designs, perform tests and material investigations and to finally perform dismantling operations under repeatable conditions on two different reactor designs on one site are given.

2. Decommissioning of the reference plant

2.1 General Strategy

The main project can be divided into three phases:

- post operation
- dismantling
- site restoration

During these three phases the main tasks can be summarized as follows:

Post operation

- planning activities, which include equipment design, QA plan and project controls manual

development, site characterization.
- operation of all systems relevant to the safe storage of fuel elements
- removal of fuel elements
- conditioning of operational waste
- dismantling of non-safety relevant systems and components
- decontamination (in situ)

Dismantling
- dismantling of all contaminated and activated systems
- handling and treatment of the dismantling waste, i. e. radiation measurement (free release), transport, conditioning, decontamination, melting, packing etc.

Site restoration
- dismantling of remaining systems (mainly utilities and handling equipment)
- building decontamination
- building demolition
- restore site or adapt to future use

The basic principles for the planning can be summarized as follows:
- The topical radiological situation in the plant is determining the detailed planning.
- The dismantling will be performed from systems or areas with lower contamination/radiation to higher contamination/radiation and finally the activated areas.
- The dismantling will in principle begin in unit 6 and end in unit 1. In this way the experience from inactive and low dose rate units can be taken into account by the work in the more contaminated and activated units.
- As far as possible market equipment and systems will be used.
- Whenever necessary, remote dismantling techniques will be applied.
- New, small and mainly mobile units for utilities are given preference for updating existing facilities.
- Obviously all works will be performed in such a way that the environment, the public and the workers are protected from unnecessary radiation.

2.2 Decontamination Strategy

Extensive use will be made of decontamination in order to reduce the level of radiation in working areas and hence also possible contamination, reduce personal dose committment, reduce remote handling requirements and reduce amount of radioactive wastes by clearance to free release or melting conditions.

In situ system- or component decontamination will be applied wherever possible, backed up by component decontamination in a special workshop.

On completion of all dismantling operations, final decontamination of the buildings (interior) will be carried out prior to demolition. Some investigations will have to be performed regarding the treatment of the contaminated concrete structures.

2.3 Dismantling Strategy

The plant will be divided into three dismantling areas, which in order of increasing radioactivity are as follows:

a) Monitored Area. This area contains mainly non-contaminated systems in the turbine hall. Work will progress from Unit 6 to Unit 1. Manual, commercial dismantling tools as well as conventional mechanical and thermal separation techniques will be adequate and applied. Dismantling and transport of components will be carried out using the existing lifting devices, handling tools and transport routes.

b) Controlled Area. This area contains the contaminated systems and components in the primary loop and all connected systems. Each dismantling task will be subject to detailed working step instructions. Manual dismantling is planned using mainly conventional mechanical separation techniques such as sawing and milling cutter. Thermal cutting may also be used but advantages will need to be considered against requirements for additional personal protection measures such as ventilation.
c) Active Area. This area consists of the reactor pressure vessel and internals, the biological protection structure and the storage for activated components. The activity inventory of Units 1-5 makes remote controlled dismantling procedures necessary. All dismantled components will require packaging and transport to the interim waste storage on site.

2.4 Waste Strategy

The general strategy for wastes is in accordance with German Atomic Law which states that radioactive materials and components must be recycled. Only if this is impossible or uneconomic disposal as radioactive waste is allowed. The two main activities to achieve a minimisation of radioactive waste are decontamination to free release or melting limits and melting to recycling limits.

For the waste from decommissioning, a first conceptual phase has been terminated. In this phase the masses were calculated from drawings and construction data sheets and activities were estimated from dose rate measurements and wipe tests in order to have a primary basis for the project planning. There are very significant amounts of different material coming together and it is mandatory to introduce methods and procedures for free release.

Operational type of wastes, i.e. solids, liquids and resins, will as far as possible be treated in existing facilities. Due to the design characteristics of these storages, development work is necessary for emptying and conditioning. Obviously the treatment aims at a reduction of the volume for disposal by e.g. evaporation, melting, high pressure compaction etc.

More detailed contamination measurements in all rooms and major systems are going on. In order to keep the radiological measurements at an acceptable level by the dismantling it is mandatory to determine the representative nuclide vectors.

It was not possible to remove the fuel elements to the central wet storage facility at the time of the reactor shut down. This means that in addition to the central storage, which is about half full, fuel elements are still in some reactor vessels and in all fuel pools of the reactors. A new dry storage is under construction on site and will accommodate the fuel elements in Castor casks.

3. Identification of R & D requirements

The preliminary identified requirements are briefly outlined below:
a) Remote dismantling of the pressure vessel and reactor internals - choice of dismantling techniques; inactive testing/training; sequence of operations; flow of wastes; packaging; storage.
b) Controlled area dismantling - optimise sequence of operations; detailed dismantling procedures; flow of wastes.
c) Development of a central data information system including the following three major parts:
- inventory, i.e. masses, materials and radiological condition
- evaluation, i.e. processing data from the dismantling and waste treatment in view of a standardisation and necessary project feedback
- project management

d) Waste management - automatic radioactivity measurement, monitoring and assessment; optimising decontamination; dismantling operations and waste packaging to minimise overall volumes for interim and final storage.
e) Further improvements and extensions in the field of application of melting and recycling of low radioactive metals, e.g.:
- application of separation melting to lead shielded cables,
- granulation of low radioactive austenitic steel and production of heavy concrete containers for storage, transport and final disposal of medium active waste.

Although these main areas are listed separately, it is to be recognised that they interact with each other and would benefit from an integrated R & D programme.

In addition to these general tasks , the following specific subjects for decommissioning VVER must be developed:
1. Development of technologies for emptying and treatment of solid and liquid operational wastes;
2. Development of procedures for the removal of larger amounts of contaminated supporting concrete structures;
3. Determination of system-related nuclide vectors;
4. Optimising free release strategies;
5. Automatic radioactivity measurement of e.g. building structures;
6. Improvement of Release Measurement Equipment in order to determine the radioactivity distribution in the measured materials and consequently optimise the free release procedure;
7. Development of an automatic scanner to determine surface contamination to achieve a free release of material;
8. Improvement of water jet decontamination technology of austenitic steel by adding suitable abrasives.
9. Development of specific treatment and handling procedures for the different dismantling materials, e.g. insulating material, electrical cables, insulating sheathing, non-ferrous material, electronic equipment, concrete etc.

<u>References</u>

[1] Dr. Heuser, F, Sicherheitsbeurteilung von WWER-Reaktoren, Energiewirtschaftliche Tagesfragen 42 (1992), pp834-843

Table I: Survey of the VVER-440

COUNTRY	SITE	MWe	MODEL	STATUS	START DATE
RUSSIA	NOVOVORONEZH 3	417	V-230	Operational	1971
	NOVOVORONEZH 4	417	V-230	Operational	1972
	KOLA 1	440	V-230	Operational	1973
	KOLA 2	440	V-230	Operational	1974
	KOLA 3	440	V-213	Operational	1981
	KOLA 4	440	V-213	Operational	1984
UKRAINE	ROVNO 1	440	V-213	Operational	1980
	ROVNO 2	440	V-213	Operational	1981
ARMENIA	ARMENIA 1	408	V-270	Shutdown ('89)	1977
	ARMENIA 2	408	V-270	Shutdown ('89)	1980
EAST GERMANY	GREIFSWALD 1	440	V-230	Shutdown	1974
	GREIFSWALD 2	440	V-230	Shutdown	1975
	GREIFSWALD 3	440	V-230	Shutdown	1978
	GREIFSWALD 4	440	V-230	Shutdown	1979
	GREIFSWALD 5	440	V-213	Shutdown	1991
	GREIFSWALD 6	440	V-213	Complete but not started	Suspended
	GREIFSWALD 7	440	V-213	Construction halted	Suspended
	GREIFSWALD 8	440	V-213	Construction halted	Suspended
BULGARIA	KOZLODUY 1	440	V-230	Operational	1974
	KOZLODUY 2	440	V-230	Operational	1975
	KOZLODUY 3	440	V-230	Operational	1981
	KOZLODUY 4	440	V-230	Operational	1982
CZECH REPUBLIC	DUKOVANY 1	440	V-213	Operational	1985
	DUKOVANY 2	440	V-213	Operational	1986
	DUKOVANY 3	440	V-213	Operational	1987
	DUKOVANY 4	440	V-213	Operational	1987
SLOVAK REPUBLIC	BOHUNICE V1-1	440	V-230	Operational	1981
	BOHUNICE V1-2	440	V-230	Operational	1981
	BOHUNICE V2-1	440	V-213	Operational	1985
	BOHUNICE V2-2	440	V-213	Operational	1985
	MOCHOVCE 1	440	V-213	Commissioning	
	MOCHOVCE 2	440	V-213	Under construction	
	MOCHOVCE 3	440	V-213	Under construction	
	MOCHOVCE 4	440	V-213	Under construction	
HUNGARY	PAKS 1	440	V-213	Operational	1983
	PAKS 2	440	V-213	Operational	1984
	PAKS 3	440	V-213	Operational	1986
	PAKS 4	440	V-213	Operational	1987
FINLAND	LOVIISA1	440	V-213	Operational	1977
	LOVIISA2	440	V-213	Operational	1981

ANNEX I

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

A. Annual Progress Reports

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

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

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

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

B. 1984 European Conference

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

C. Final Contract Reports

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

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

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

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

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

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

Verry, P, Lecoffre, Y (1984). Décontamination de surfaces par érosion de cavitation. EUR 8956.

Allibert, M, Delabbaye, F (1984). Extraction du cobalt des aciers inoxydables. EUR 8966.

Ebeling, W, et al. (1984). Dekontamination von Betonoberflächen durch Flammstrahlen. EUR 8969.

Boulitrop, D, Rouet, D (1984). Etude de la décontamination au moyen de supports gélifiés. EUR 9102.

Peselli, M (1984). Individuazione quantitativa delle impurezze del contenitore a pressione del reattore del Garigliano. EUR 9167.

Avanzini, P G, et al. (1984). Valutazione delle caratteristiche di progetto che facilitano lo smantellamento delle centrali nucleari PWR. EUR 9191.

Regan, J D, et al. (1984). Design features facilitating the decommissioning of Advanced Gas-Cooled Reactors. EUR 9207.

May, S, Piccot, D (1984). Détermination analytique d'éléments traces dans des échantillons de bétons utilisés dans les réacteurs nucléaires de la Communauté européenne. EUR 9208.

White, I F, et al. (1984). Assessment of management modes for graphite from reactor decommissioning. EUR 9232.

Goddard, A J H, et al. (1984). Trace element assessment of low-alloy and stainless steels with reference to gamma activity. EUR 9264.

Bregani, F, et al. (1984). Chemical decontamination for decommissioning purposes. EUR 9303.

Larcombe, M H E, Halsall, D R (1984). Robotics in nuclear engineering. Graham & Trotman Ltd., London. EUR 9312.

Glock, H -J, et al. (1984). Dokumentationssystem für den Abbau von Kernkraftwerken. EUR 9343.

Ahlfänger, W (1984). Zusammensetzung von Kontaminationsschichten und Wirksamkeit der Dekontamination. EUR 9352.

Brambilla, G, et al. (1984). Vernici per la fissazione della contaminazione superficiale dei materiali. EUR 9358.

Paton, A A, et al. (1984). Civil engineering design for decommissioning of nuclear installations. Graham & Trotman Ltd, London. EUR 9399.

Bittner, A, et al. (1985). Konzepte zur Minimierung der Aktivierung des biologischen Schilds. EUR 9442.

Brambilla, G, Beaulardi, L (1985). Rivestimenti rimovibili per la protezione di superfici in calcestruzzo dalla contaminazione. EUR 9463.

Arndt, K -D, et al. (1984). Thermisches Trennen von plattierten Komponenten des Primärkreises von Kernkraftwerken. EUR 9479.

Rawlings, G W (1985). Development of large diamond-tipped saws and their application to cutting large radioactive reinforced concrete structures. EUR 9499.

Barody, I I, et al. (1985). Treatment of active concrete waste arising from dismantling of nuclear facilities. EUR 9568.

Bernard, A, Gerland, H (1985). Recherche et caractérisation de revêtements pour la protection des structures en béton. EUR 9595.

De Tassigny, C (1985). Mise au point et essais d'une méthode pour le revêtement de déchets métalliques contaminés, par des résines thermodurcissables. EUR 9666.

Migliorati, B, et al. (1985). Smantellamento di componenti metallici e di strutture in calcestruzzo mediante raggio laser. EUR 9715.

Fleischer, C C (1985). A study of explosive demolition techniques for heavy reinforced and prestressed concrete structures. EUR 9862.

Wieling, N, Hofmann, P J (1985). Erosionskorrosionsversuche mit kobaltfreien Werkstoffen. EUR 9865.

Antoine, P, et al. (1985). Intégrité à long terme des bâtiments et des systèmes. EUR 9928.

Lewis, G NH (1985). Degradation of building materials over a lifespan of 30-100 years. EUR 10020.

Hasselhoff, H, Seidler, M (1985). Anlage zum Einschmelzen von radioaktiven metallischen Abfällen aus der Stillegung. EUR 10021.

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LIST OF PUBLICATIONS RELATING TO THE RESULTS OF THE 1984-88 PROGRAMME ON THE DECOMMISSIONING OF NUCLEAR INSTALLATIONS

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"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - First Annual Progress Report (year 1985)", EUR 10740, 1986.

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