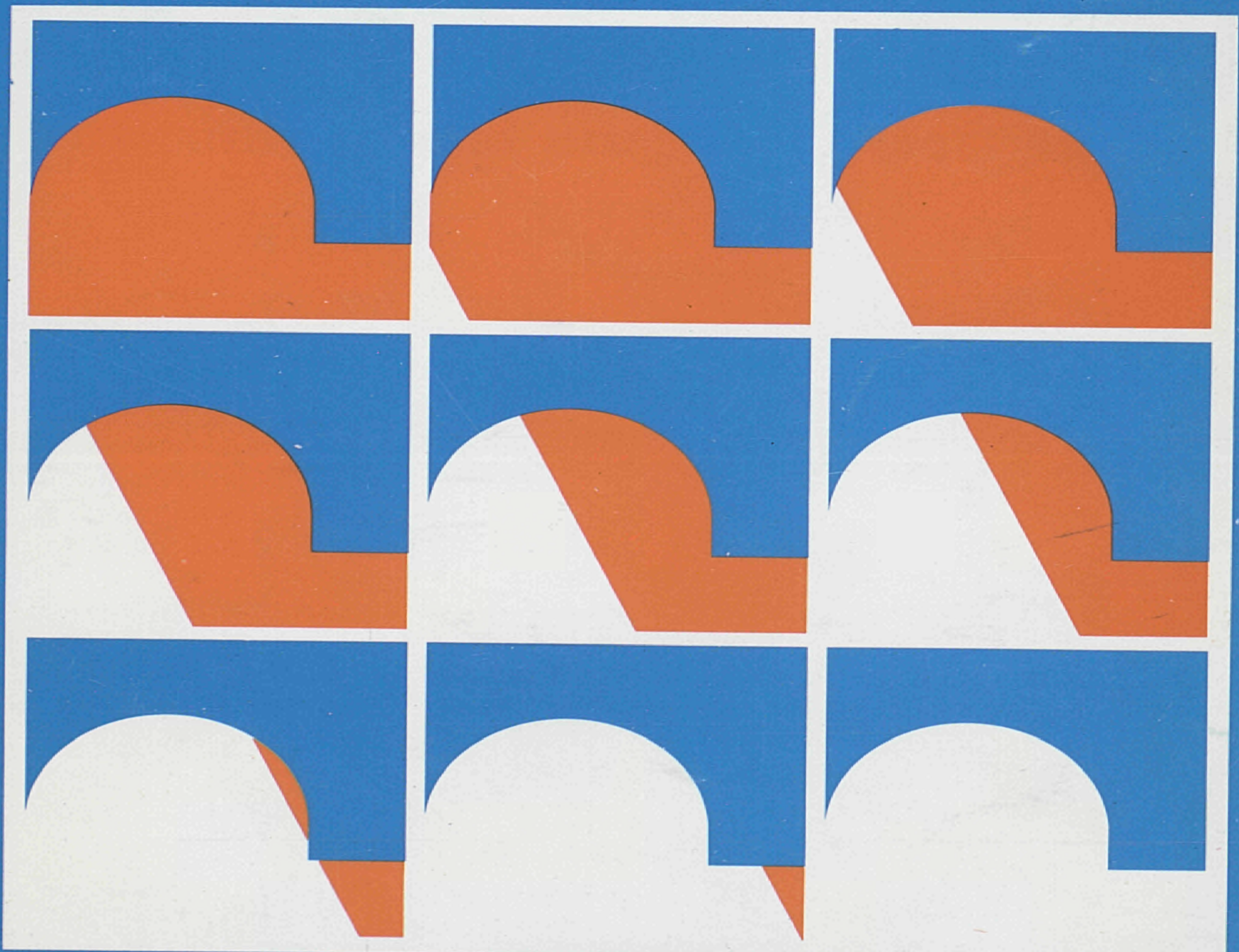
Commission of the European Communities

nuclear science and technology

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

Fourth annual progress report 1988



Report

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Directorate-General
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FOREWORD

This is the fourth and last Annual Progress Report of the European Community's 1984-88 programme of research on the decommissioning of nuclear installations. It covers the year 1988 and follows the 1985, 1986 and 1987 Reports /1,2,3/.

The Council of the European Communities adopted the programme in January 1984 /4/, considering: "Certain parts of nuclear installations inevitably become radioactive during operation; it is therefore essential to find effective solutions which are capable of ensuring the safety and protection of both mankind and the environment against the potential hazards involved in the decommissioning of these installations".

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

The 1984-88 programme has the following contents:

A. Research and development projects concerning the following subjects:
   - Project N° 1: Long-term integrity of building and systems;
   - Project N° 2: Decontamination for decommissioning purposes;
   - Project N° 3: Dismantling techniques;
   - Project N° 4: Treatment of specific waste materials: steel, concrete and graphite;
   - Project N° 5: Large containers for radioactive waste produced in the dismantling of nuclear installations;
   - Project N° 6: Estimation of the quantities of radioactive wastes arising from the decommissioning of nuclear installations in the Community;
   - Project N° 7: Influence of installation design features on decommissioning.

B. Identification of guiding principles, namely:
   - certain guiding principles in the design and operation of nuclear installations with a view to simplifying their subsequent decommissioning,
   - guiding principles in the decommissioning of nuclear installations which could form the initial elements of a Community policy in this field.

C. Testing of new techniques under real conditions, within the framework of large-scale decommissioning operations undertaken in Member States.

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

The Commission is responsible for managing the programme and is assisted in this task by the Management and Coordination Advisory Committee "Nuclear fission energy - Fuel cycle/processing and storage of waste" (see Annex III).

The subject of this report is formed by 72 research contracts, including three new contracts concluded in 1988 as well as 42 contracts of which the execution has been completed in 1986, 1987 and 1988.

The present report describes the objectives, scope and work programme of each research contract concluded, as well as the progress of work achieved and the results obtained in 1988.

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

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

B. Huber
Head of the Programme

References

/1/ "The Community's research and development programme on decommissioning of nuclear installations. First annual progress report (year 1985)". EUR 10740, 1986.

/2/ "The Community's research and development programme on decommissioning of nuclear installations. Second annual progress report (year 1986)". EUR 11112, 1987.

/3/ "The Community's research and development programme on decommissioning of nuclear installations. Third annual progress report (year 1987)". EUR 11715, 1988.

/4/ Council Decision of 31 January 1984 adopting a research programme concerning the decommissioning of nuclear installations. OJ N° L 36, 8.2.1984, p. 23.

CONTENTS

Page

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

A. Objective

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

B. Research performed under the 1979-83 programme

The work performed under the previous programme relates mainly to the following aspects:
- mode and pace of degradation of various materials as they exist in nuclear power plants;
- measures for maintaining plants in a safe condition and for keeping the necessary ancillary systems operable;
- monitoring and inspection procedures;
- radiological consequences and costs of maintaining the plants.

C. 1984-88 programme

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

D. Programme implementation

Three research contracts relating to Project N°1 were executed in 1988, of which two were completed.

1.1. Deterioration Assessment of Nuclear Power Station Buildings

Contractor: Taylor Woodrow Construction Ltd, Southall, UK
Contract N°: FI1D-0030
Working Period: April 1986 - December 1988
Project Leader: D.C. Pocock

A. Objectives and Scope

The objective of this research is to study the long-term performance of structures comprising nuclear power plants. The time period of interest for this study is 140 years (this figure is based on maximum periods of 40 years for operation and 100 years of storage). Particular attention will be given to those parts of the plant for which leak tightness and structural integrity are required, both during operation and for long periods after final shutdown.

This research will be executed in close co-operation with Zerna, Schnellenbach und Partner GmbH (see Par. 1.2.).

The specific aim of this research is to predict future deterioration of nuclear power station buildings, due to corrosion of reinforcement and prestressing steel. The state and rate of degradation of existing buildings will be assessed to provide qualitative data and to improve existing knowledge of the factors controlling the ageing process of nuclear plant buildings. Relevant plant materials will be identified and proposals made for monitoring procedures, preventive measures and recommendations for future designs.

Buildings to be investigated will be typical of power stations of the United Kingdom. However, the results will be applicable also to plants sited in other European Community countries due to the nature of the specific problems posed. The survey of station buildings will be carried out on a range of nuclear power sites in the United Kingdom, selected to provide a range of exposure conditions, various degrees of deterioration and a range of concrete types. This survey will include the shut-down Gas-cooled Reactor stations of Chinon-A1 and Marcoule-G2 in France.

B. Work Programme

B.1. Selection of sites and concrete types.

B.2. Literature survey including the assessment of the in-situ state of the concrete and steel and the determination of the causes for concrete deterioration and corrosion of steel.

B.3. In-situ testing of the materials, using non-destructive techniques, including the measurements of ultrasonic pulse velocity, rebar potential, concrete resistivity, etc.

B.4. Laboratory tests on samples removed from safe areas (concrete strength, depth of carbonation, water permeability, oxygen diffusion, etc.).

B.5. Use of the test results, to develop a computer program predicting rate of deterioration, onset and rate of corrosion, extent of cracking and spalling, damages, service life of the structure, etc.

B.6. Recommendations for damage prediction and for reducing corrosion rates.

C. Progress of Work and Obtained Results

Summary

Three suitable sites for survey work were agreed with the Central Electricity Generating Board (CEGB), namely Bradwell, Hinkley Point "A" and Wylfa Power Stations. The UK Atomic Energy Authority (UKAEA) site at Windscale was also made available. Preliminary surveys were conducted at each of the four sites, for planning and familiarisation purposes. These were followed by full surveys, including visual inspections, environmental monitoring, non-destructive testing and sample retrieval for subsequent laboratory analysis. The analysis of data and samples is substantially complete and information gathered is being used to predict the estimated timescales of reinforcement corrosion. Reporting is underway.

Progress and Results

1. Basis of Nuclear Site Selection (B.1.)

The research programme involves surveying existing nuclear power plants to provide quantitative data on the state and rate of degradation of buildings. Constraints within the programme dictated that it was not practicable to include all UK nuclear power stations. Therefore to meet the objectives of the programme, stations for survey work were selected to cover a range of geographical locations. To improve the calibration process for the deterioration prediction package, reference was also made to previous work by the Central Electricity Generating Board under the CEC's phase 1 research programme on Decommissioning of Nuclear Power Plants (1).

2. Survey Planning (B.3., B.4., B.5.)

At stations available for the work, survey planning began with preliminary visits for discussions with staff, selection of specific work locations and to facilitate preparation of survey specifications. Since Exposure and performance of the concrete were expected to differ significantly in external and internal environments, areas of both externally and internally exposed concrete were selected.

3. Survey Activities (B.3., B.4., B.5.)

Following preliminary visits, the main surveys were performed including the following four main categories of work.

a) visual inspection
b) environmental monitoring
c) non-destructive testing (N.D.T.)
d) sampling

Specific survey activities were each closely related to either environmental considerations, corrosion activation processes, or corrosion propagation processes.

4. Environment (B.3, B.4., B.5.)

Overall, the environmental considerations may be summarised as follows:

a) The external concrete is presently exposed to coastal, saline environments conducive to chloride-induced corrosion;

b) Internal environments are conducive to carbonation of the concrete, but corrosion of reinforcement is at present unlikely in general to proceed at significant rates, except when supported by regular localised wetting;

c) The potential exists for indoor concrete, carbonated during the station's operating life, to support increased corrosion rates later in the overall lifespan;

5. Literature Survey and Deterioration Prediction (B.2., B.6.)

In freshly placed concrete, reinforcement steel is protected by the high alkalinity of the surrounding cement paste, which causes the steel to be passivated. With time, however, the alkanity of the surrounding cement paste may be reduced by the neutralising effects of carbon dioxide diffusion into the concrete from the atmosphere (carbonation). Furthermore, the ingress of corrosive salts, primarily chlorides, can destroy passivation resulting in corrosion even under the highly alkaline environment of steel in concrete.

The rate of progression of the de-passivation front through the reinforcement cover is slow. The time taken for the depth of penetration of chlorides or carbonation to become equal to the depth of cover to the reinforcement is known as the initiation time (T_0).

Once corrosion has been identified, the rate at which the corrosion progresses is determined by both the rate of the anodic and cathodic reactions and the resistivity of concrete and the availability of moisture and oxygen. Assuming an adequate supply of oxygen and water, corrosion proceeds for a second period (T_1) until the amount of corrosion products formed is sufficient to cause cracking and eventually spalling of the concrete.

The programme concentrates on the period leading to the initiation of corrosion, which is the key to the life prediction process. The two main mechanisms, mentioned above, by which corrosion is initiated are being pursued, with the aim of producing a more rigorous model for estimating the time to the onset of corrosion, based on the following factors:

- rate of progress of the carbonation front

- rate of chloride diffusion into the concrete and to reach a threshold level at the steel location

- depths and variations in depth of concrete cover to the steel

6. Preliminary Conclusions (B.3., B.4., B.5.)

From surveys undertaken and initial analysis work, the following preliminary results have emerged.

a) Whilst there was little if any reinforcement corrosion underway in internally exposed structures they displayed various forms of cracking and spalling, either where cover was low, or in very severe local environments.

b) Chloride concentrations affecting reinforcement were generally low in internal concretes as expected. Some external concretes, however, contained appreciable concentrations of chloride; this data will be used to predict the onset of chloride-induced corrosion.

c) Depths of carbonation were negligible in external concrete, but up to 50 mm in internal concrete, which was in excess of some reinforcement cover depth measured. This indicates that carbonation is likely to have penetrated to reinforcement in a percentage of the concrete and to have caused corrosion initiation. This extent of initiation is likely to increase considerably over the structures' total lifespan as a result of continuing carbonation penetration.

d) The consequent rates of corrosion propagation are likely to be high in external concretes, based on damage observations and N.D.T. results so far. Internal concrete may support corrosion rates of long-term significance if moisture penetration into the indoor environment is permitted to increase, which could be expected to occur with progressive ageing of the various building components. This should be taken into account in ongoing maintenance and planning of decommissioning operations.

References

/1/ LEWIS, G.H., Degradation of Building Materials Over a Life-span of 30 - 100 years. CEC Report EUR 10020. 1986.

1.2. Long-term Stability and Leak Tightness of Reactor Containments

Contractor: Zerna, Schnellenbach und Partner GmbH, Bochum, Germany.
Contract N°: FI1D-0031
Working Period: April 1986 - December 1988
Project Leader: R. Oberpichler

A. Objectives and Scope

The objective of this research is to study the long-term performance of structures comprising nuclear power plants. The time period of interest for this study is 140 years (this figure is based on maximum periods of 40 years for operation and 100 years of storage). Particular attention will be given to those parts of the plant for which leak tightness and structural integrity are required, both during operation and for long periods after final shutdown.

This research will be executed in close co-operation with Taylor Woodrow Construction Ltd (see Par. 1.1.).

The specific aim of this research is to investigate the behaviour of complex composite structures, taking as a basis the long-term behaviour of materials. The possible susceptibility to long-term damage will also be assessed, and the areas most prone to such damage will be identified. Further consideration will be given to the possible interaction between sealing steel components (steel containments, steel liners) and load bearing concrete structures.

This building survey will be carried out on structural elements of actual PWR stations (e.g. Emsland-Lingen) and BWR stations (e.g. Gundremmingen B and C). Consideration will be given to the validity of the investigations for relevant structures of other commercial nuclear power plants in the European Community. This investigation will include the shut-down BWR station of Garigliano in Italy.

B. Work Programme

B.1. Investigation on reinforced concrete and prestressed concrete structures.

B.1.1. Selection of structural elements considered important with regard to the integrity of long-term containment.

B.1.2. Literature study on material behaviour covering long-term properties.

B.1.3. Analysis of the long-term behaviour of the selected structural elements.

B.2. Investigation of steel containments

B.2.1. Selection of elements susceptible to damage, in particular plastic sealings with concrete and steel.

B.2.2. Assessment of damage (state of material, types of corrosion, formation of condensed moisture, permeability of the concrete, etc.)

B.2.3. Optimisation of ultrasonic testing techniques (angular sound, weakening, creep wave, etc.) and application of the selected techniques to decommissioned Niederaichbach and Gundremmingen I nuclear power plants.

B.3. Recommendations for monitoring and enhancing long-term integrity of reinforced and prestressed concrete and for assessment of in-situ corrosion of steel elements.

C. Progress of Work and Obtained Results

Summary

During 1988 the work was continued concerning the following parts of the research programme:

- long-term performance of selected structural elements of the shut-down BWR station Garigliano in Italy (B.1.3. and B.2.),
- classification and assessment of damage caused by corrosion (B.2.2.),
- long-term behaviour of plastic seals (B.2.1.),
- non-destructive tests (B.2.3.).

The first topic deals with the long-term behaviour of the main structural elements of the shut-down BWR station Garigliano. The following three chapters deal with the steel containments and the steel liner. Concerning the corrosion mechanism literature studies were continued. The long-term behaviour of plastic seals was drawn up by a test programme. Ultrasonic tests were carried out in order to define decisive criteria for the detectability.

Progress and Results

1. Long-Term Performance of Selected Structural Elements of the Shut-Down BWR Station Garigliano in Italy (B.1.3. and B.2.)

The nuclear power plant Garigliano is a boiled water reactor-station, designed for the production of an electric power of 160 MWe. Figure 1 shows a section of the reactor building. The 2 m thick concrete foundation plate is supported by 750 concrete piles. The concrete calotte having a shape of a 1 m thick spherical shell section is lying on the foundation plate. The steel containment with an outer diameter of about 49 m and a thickness of 25.4 millimeters is embedded in this concrete calotte shell. The inner structures are very bulky concrete structures (e.g. the biological shield has a wall thickness of at least 1.85 m).

The requested long-term performance of the reactor building is mainly dependent on two structural elements: the supporting structures, consisting of the concrete piles, foundation plate and calotte shell, which will be treated here, and the steel containment.

Presumed the used concrete meets according to DIN 1045 with the requirements of a B25 at least and the designed load considers not only dead load but also some extreme plant states, effecting the utilization factor being far below 1,0 during the past-operational state, it can be concluded - applying the formerly discussed general results -, that the bulky concrete structures of the Garigliano power plant are not going to deteriorate in consequence of normal environmental conditions within the regarded period of time. Nevertheless, it should be recommended to monitor the supporting structures periodically. After all it can be stated, that the concrete structures are not the "weak point", as far as the long-term performance is concerned. More relevant for the life-time of the Garigliano NPP is the leak-tightness of the steel containment. There are two critical areas which should be monitored carefully. Firstly, the transition area of the steel containment in the concrete calotte may lead to corrosion because cracks in the grouting material cannot be excluded. Secondly, the penetrations of the containment are sensitive points with regard to corrosion, too. Monitoring and maintenance procedures should be taken into consideration.

2. Classification and Assessment of Damage Caused by Corrosion (B.2.2.)

Corrosion protection of the containments of pressurized-water reactors (PWRs) and of the steel liners of boiling-water reactors (BWRs) is generally guaranteed by special measures such as protective paint coatings and seals. However, despite these measures, the occurrence of corrosion is to be expected in critical design areas, for example

- in the encased bottom portion of containments of pressurized-water reactors,
- at platforms of adjoining areas and penetrations of pressurized-water reactors and
- at steel liners of boiling-water reactors.

The results obtained for the given boundary conditions indicate that the occurrence of uniform corrosion, shallow pit corrosion and pitting corrosion cannot be ruled out although shallow pit and pitting corrosion have not yet been observed in detected damage cases. For the materials and medium conditions used here, values of between 0.09 mm/a and 0.3 mm/a were found in literature for mass loss rates respectively erosion rates resulting from uniform corrosion.

Data on the corrosion velocity of shallow pit corrosion and pitting corrosion are generally not very reliable since they frequently depend on non-definable local conditions such as partly disturbed surface areas e.g. in gaps, porosity of anticorrosive coatings etc. and cannot be described by specific laws. At local points this means that metal can be eroded quickly and that the corrosion velocity can be many times the corrosion velocity of uniform corrosion.

Consequently, as determined within the scope of our investigations taking account of operating experience with older plants in particular, a long-term corrosion risk for the containments of PWR plants and steel liners of BWR plants can only be ruled out if design, operational, monitoring and repair measures as well as drying measures are taken.

3. Long-Term Behaviour of Plastic Seals (B.2.1.)

During 1988, thermal aging tests were performed at different temperatures (40, 60, 80 and 120 °C) on 200 prepared specimens of the silicon material RTV-1 manufactured by Wacker Chemie Burghausen (FRG) which was the subject of the examination. For this purpose, the silicon specimens were compressed by 25 % in special facilities. Then after each of the 9 different aging periods, 5 silicon specimens were removed from the heating furnace and the compressive deformation set values were established.

Instead of continuing the tests for the planned aging period of 10000 hours they had to be terminated after 9000 hours because of schedule reasons.

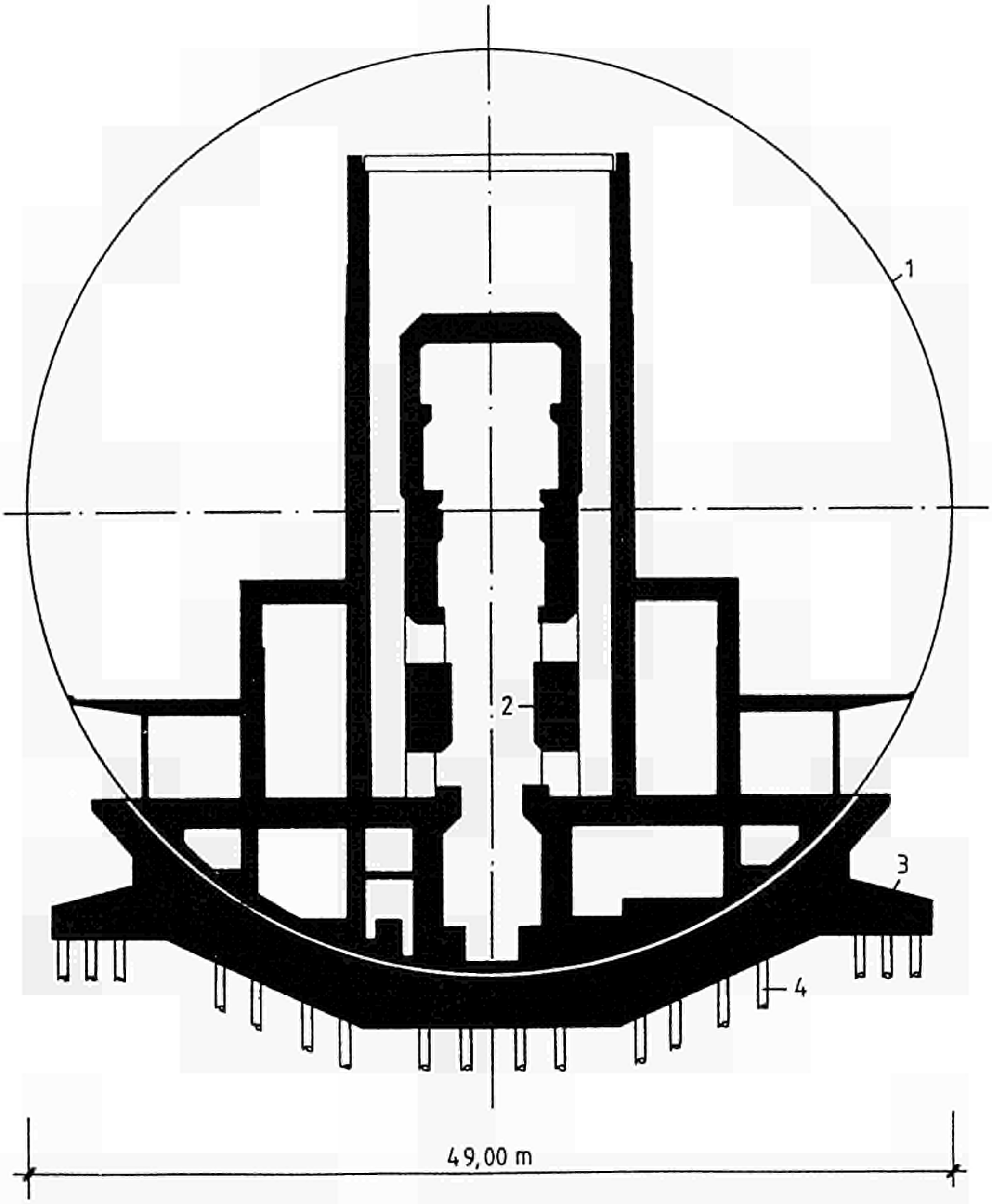
After the thermal aging tests at the different aging temperatures other material tests such as Shore-A-Hardness measurement, density determination and tensile tests were performed

The test results were presented in tables. The average values of the test results were shown graphically, evaluated using valid aging laws and, wherever possible, extrapolated to a long application period.

The results of the material tests were interpretated on the basis of additional literature studies and an attempt was made to evaluate the long-term suitability of the silicon material for use as sealing material in safety-related areas of nuclear power plant components.

4. Non-Destructive Tests (B.2.3.)

By means of calibration blocks with corroded surfaces it was verified that with 45° angle beam search units and a test frequency of 2 MHz corrosion can be clearly detected within a distance of up to 150 mm from the upper edge of the encased bottom portion of the containment. Thereby, the decisive criterion for detectability is not the depth of the corrosion point but solely the type of pitting. The signal/noise ratio should amount to approximately 12 dB for detection of the indications.



1 steel containment
2 biological shield
3 foundation plate
4 concrete piles

FIGURE 1 : Section of the BWR-station reactor building, Garigliano

1.3. Consequences of Suppression of Negative Pressure in the KW-Lingen Containment

Contractor: Kernkraftwerk Lingen GmbH, Lingen, Germany
Contract N°: FI1D-0032
Working Period: February 1987 - January 1989
Project Leader: W. Harbecke

A. Objectives and Scope

It is common practice to maintain a negative pressure in the containment of shut-down nuclear reactors in order to avoid a transfer of the remaining radioactivity to the outside.

The objective of the present contract is to assess, from the standpoint of radiation protection of the environment, the acceptability of suppressing the provided ventilation and, consequently, the negative pressure in the containment of the shut-down Lingen Boiling Water Reactor Plant. The use of the ventilation system would then be limited to casual drying and air-conditioning purposes.

The work is aimed at demonstrating that safe enclosure without negative containment pressure might be acceptable.

B. Work Programme

B.1. Inventory of the relevant plant characteristics and of the key issues to be considered.

B.2. Estimation of the activity release to be expected with shut-down ventilation system.

B.3. Design of an instrumentation system and a measurement programme for the control of the containment atmosphere.

B.4. Implementation of the measurement programme.

B.5. Evaluation of the obtained results.

C. Progress of Work and Obtained Results

Summary

During this period, the measurement programme for the control of the containment atmosphere was pursued. Several data were collected.

Progress and Results

1. Implementation of the measurement programme (B.4.)

The calculated air exchange with the environment is shown in figure 1. The data are based on the measured distribution of wind velocity and the leakage-rate of the buildings as a function of pressure-difference (measurements carried out with and without negative pressure).

The activity releases are shown in figure 2. The calculations are done twice: with negative pressure inside the buildings (the average value, forced by an off-air rate of 600 m^3/h was 10 pa) and without negative pressure. The aerosol activity inside the area of safe enclosure was measured. The total beta/gamma activity was found to be in the range of 1.5^{6} to 2.6^{6} Bq/m^3 condensate. That means nearly 10 Bq/m^3 of air according to a water content of 6 g/m^3 of air.

Figure 3 shows the relative humidity inside the reactor building at 3 different floors and the outside values. The figures are average values for one week.

The humidity inside the area of the safe enclosure is kept low by a drying system. The condensate is collected in tanks. The amount of collected water is given in figure 4.

Figure 5 shows some typical temperatures inside and outside the plant. The figures are average values for one week.

2. Evaluation of the obtained results (B.5.)

Due to the low specific aerosol activity and the good tightness of the plant, the air exchange with the environment and the output of activity is very small. The outdoor temperatures were unusual mild during the winter period.

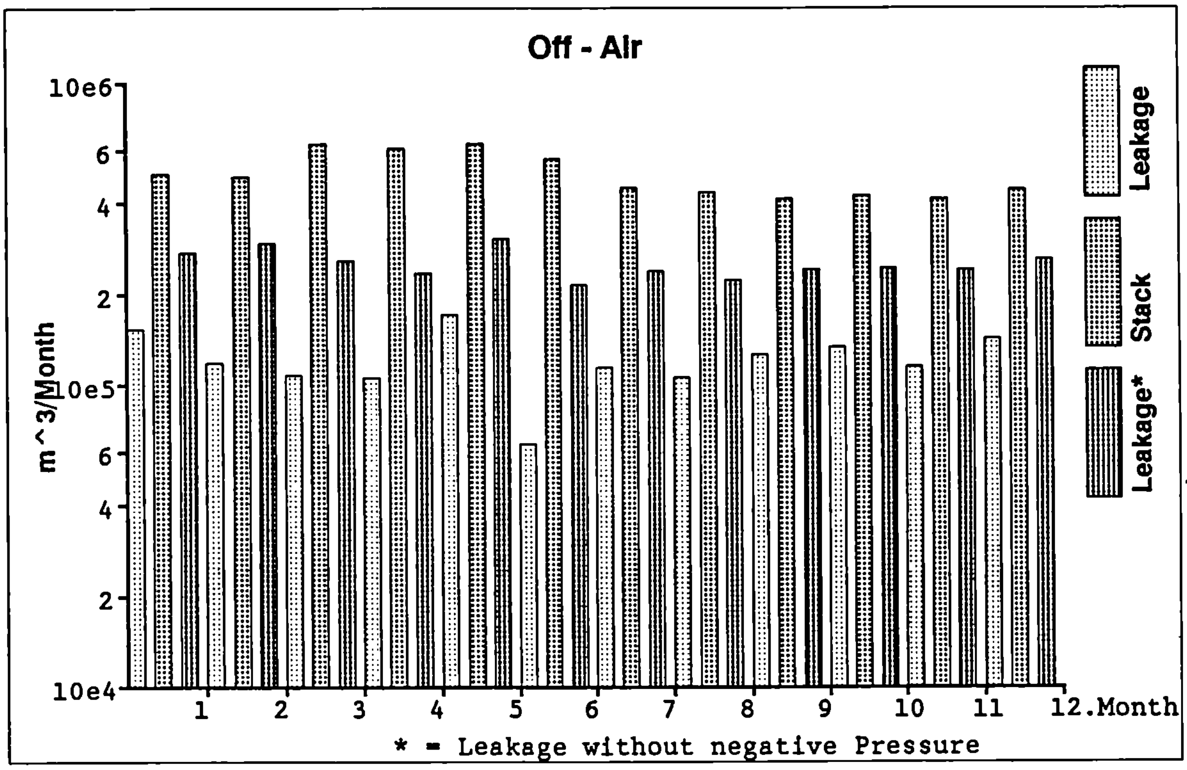


Figure 1 Air exchange with the environment with and without negative Pressure

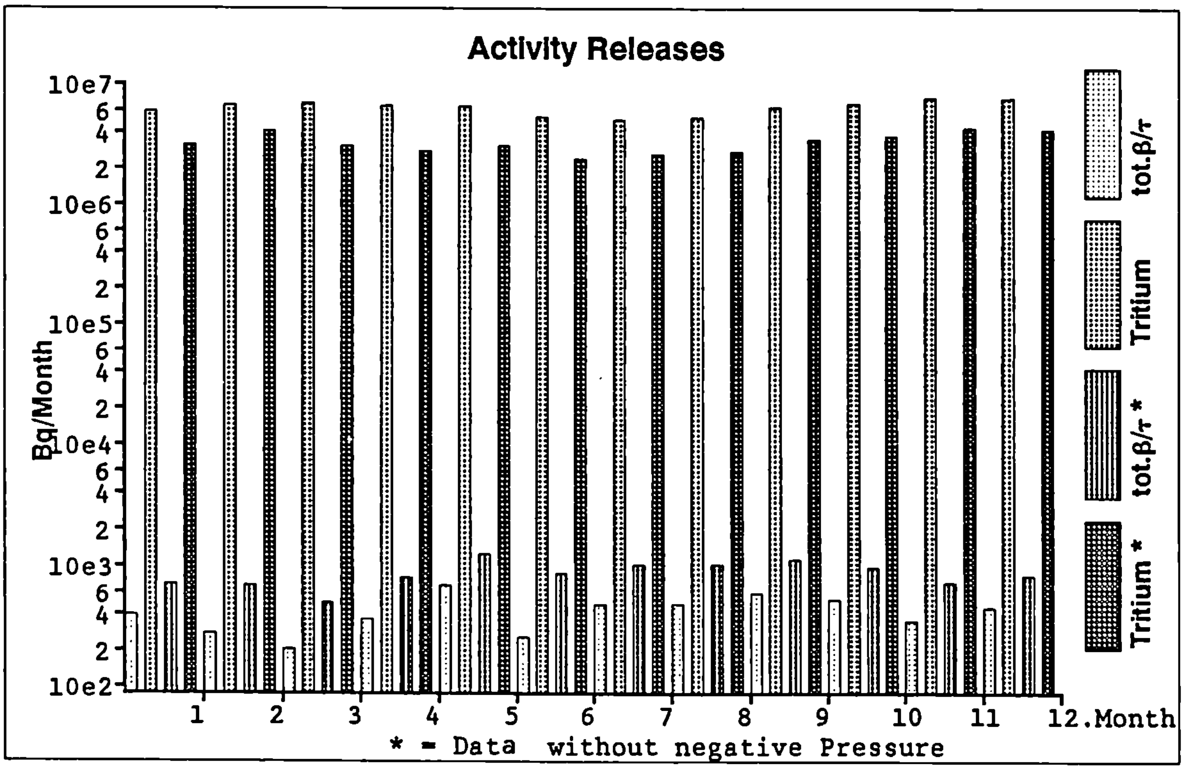


Figure 2 Activity releases with and without negative Pressure

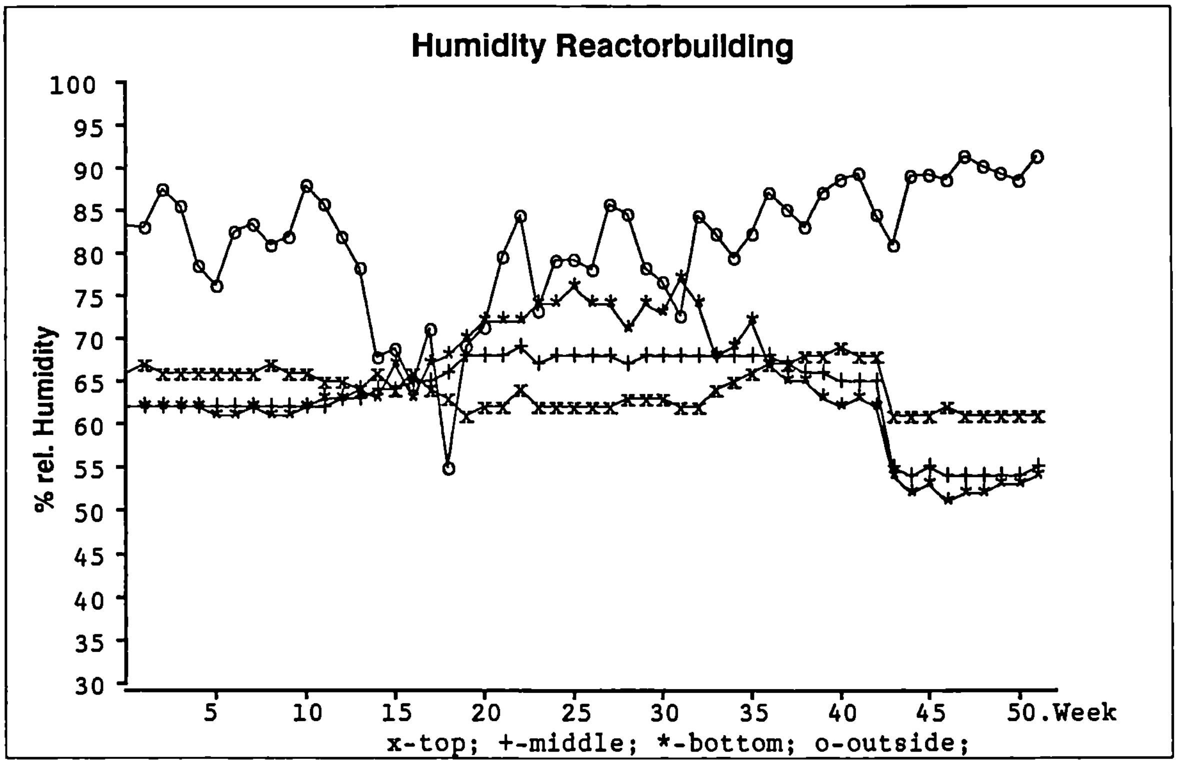


Figure 3 Relative Humidity at different floors inside the reactorbuilding and outside the plant

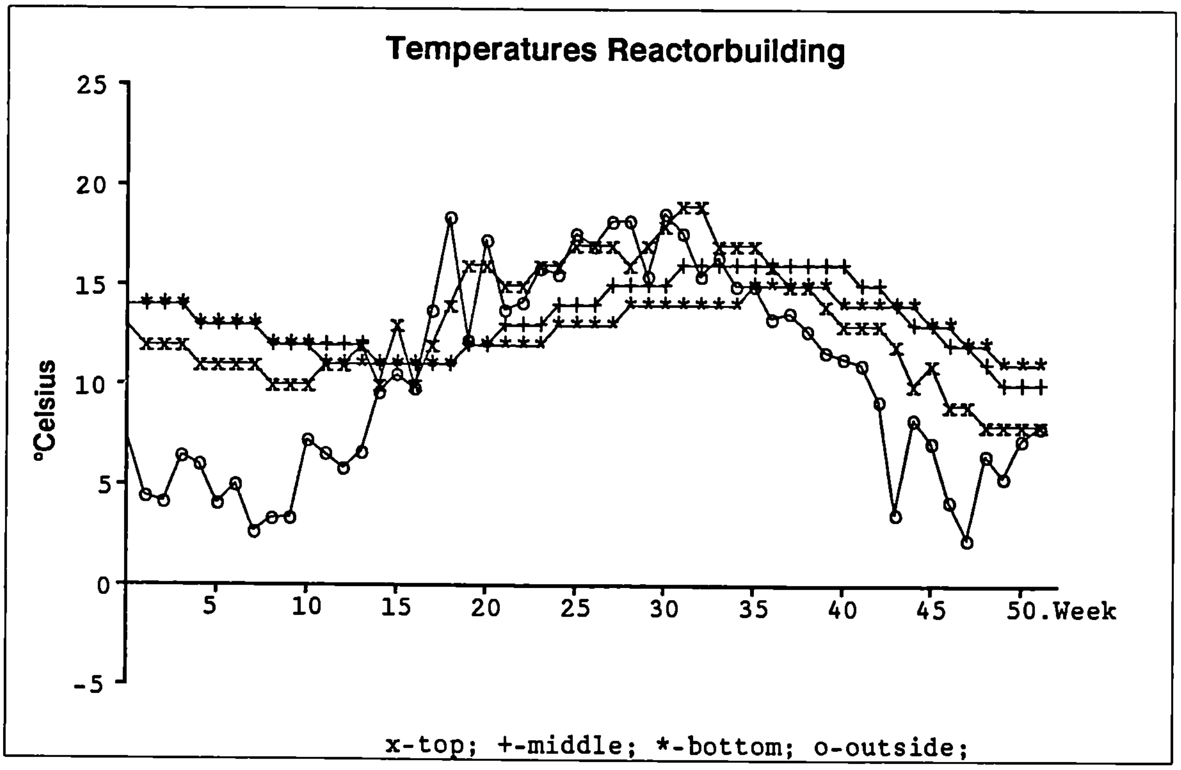


Figure 4 Temperatures at different floors inside the reactorbuilding and outside the plant

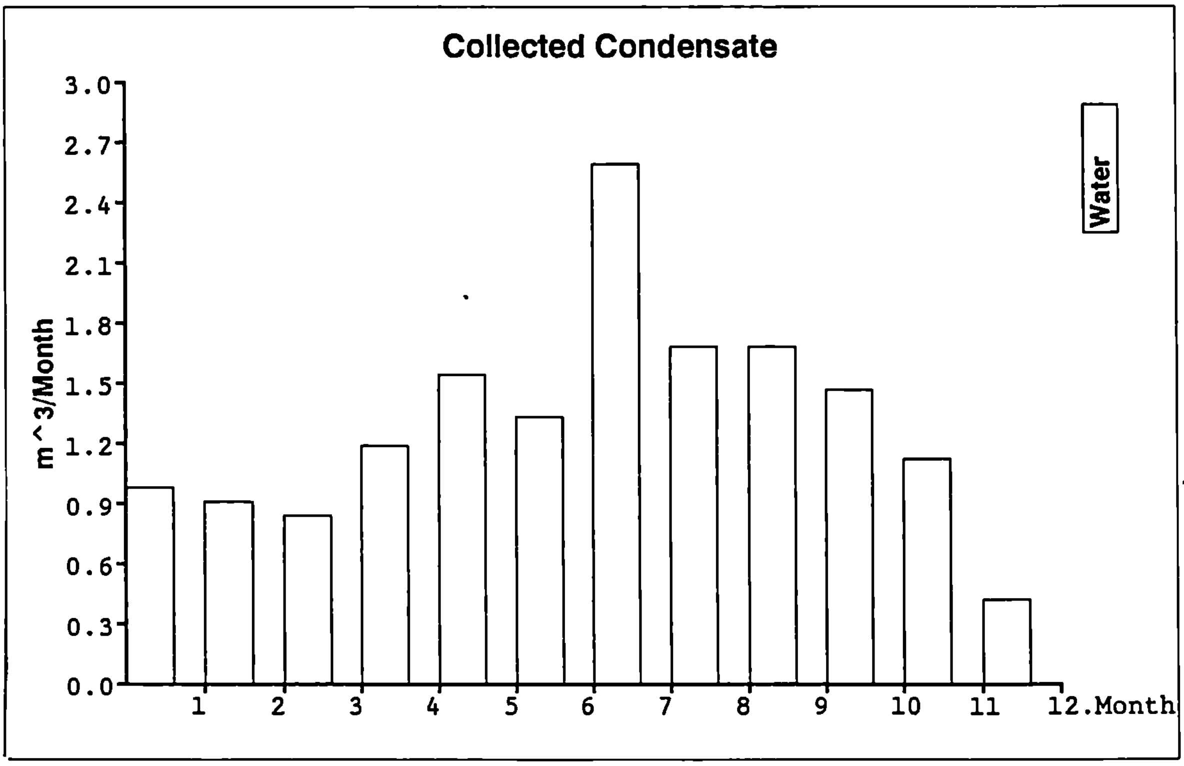


Figure 5 Condensate, monthly collected by the air-drying system

2. PROJECT N°2: DECONTAMINATION FOR DECOMMISSIONING PURPOSES

A. Objective

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

B. Research performed under the 1979-83 programme

The following decontamination techniques have been developed and assessed:

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

Other activities were:

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

C. 1984-88 programme

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

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

D. Programme implementation

Seven research contracts relating to Project N°2 were executed in 1988, of which six were completed.

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

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

A. Objectives and Scope

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

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

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

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

B. Work Programme

B.1. Manufacturing of the decontamination rig comprising the sample steam pipe and all needed components for decontamination.

B.2. Preliminary laboratory decontamination tests of representative samples including determination of the composition and activity level of the contaminated layer.

B.3. Main test programme using the decontamination rig.

B.4. Assessment on optimal treatment of the generated radioactive secondary waste.

B.5. Evaluation of experimental results with respect to man-dose, quantities of secondary waste and cost analyses, with extrapolation to a 1200 MWe BWR.

C. Progress of Work and Obtained Results

The work has been completed and the final report is now available as EUR report No. 11435.

2.2. Aggressive Chemical Decontamination Tests on Valves from the Garigliano BWR

Contractor: Ente Nazionale per l'Energia Elettrica, Roma, Italy
Contract N°: FI1D-0002
Working Period: January 1985 - December 1988
Project Leader: F. Bregani

A. Objectives and Scope

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

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

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

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

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

Through a supplementary agreement concluded in 1987, the initial work programme is extended by items B.4. to B.6.

B. Work Programme

B.1. Aggressive chemical decontamination tests on valves (2-3 inches) of the primary cooling system of the Garigliano BWR in DECO loop.

B.2. Identification and qualification of a simple procedure to condition the spent decontaminant.

B.3. Neutralisation and flocculation tests in order to select and evaluate the best neutralising agent and specific chemical agents, such as chitosan, as supporter in flocculation.

B.4. Development of the decontamination process by using ultrasounds together with aggressive chemicals.

B.5. Decontamination tests with this method on contaminated samples of about 10 cm^2 surface (stainless and carbon steel). If the results of the previous tests are satisfactory, the decontamination process will be applied to a little valve (1-2 inch).

B.6. Radiochemical measurements on selected samples before and after decontamination, in particular for the following elements: Fe-55, Ni-59 and Ni-63.

B.7. Cost evaluation of the process and assessment of the possibility of reprocessing and reutilizing of specific agents.

C. Progress of Work and Results Obtained

Summary

After the literature review on the effect of ultrasonic waves on decontamination and the first experiments, more than 100 lab tests in a beaker configuration were carried out in order to investigate and to optimize the decon process using ultrasounds in aggressive chemical baths. The effect of the different process parameters such as ultrasonic frequency and power, fluid viscosity and surface tension, pressure and temperature of the test solution, test duration, kind and concentration of chemicals and test materials was investigated on contaminated AISI 304 specimens from the Garigliano BWR.

As a conclusion of the experimental activities two in-scale tests were carried out at Garigliano BWR Power Station, using a special US machine, on two contaminated 2" valves similar to the valves used in the aggressive chemical decon tests performed on the DECO loop in the first phase of the research.

Progress and Results

1. Introduction

The literature review showed that ultrasound is widely used in the nuclear industry as a decontamination technique, whilst it is never checked for decommissioning. It is generally used by immersing the piece to be decontaminated in water or water with detergents. No review has been performed on the simultaneous use of ultrasound in aggressive chemical solutions.

The process parameters that affect the scrubbing factor or the decontamination effectiveness were found. A general relationship can be written as:

$$\Delta P \text{ (or DF)} = \Delta P\,(f,\ w,\ p,\ \sigma,\ \nu,\ T,\ t,\ c)$$

where f is the ultrasound frequency, w the ultrasonic power, p the pressure of the fluid, σ the surface tension of the fluid, ν the fluid viscosity, T the process temperature, t the test duration and c the type and concentration of the chemicals.

Using ultrasound in a tank the pressure is atmospheric and it does not vary; moreover for aggressive chemical solutions at 3-10% of concentration, the surface tension and the fluid viscosity were found not to vary greatly so the previous relationship becomes:

$$\Delta P \text{ (or DF)} = \Delta P\,(f,\ w,\ T,\ t,\ c)$$

2. Laboratory testing (B.4, B.5)

In the first stage of the experimental activities, described in the last progress report, some laboratory tests were carried out with a frequency around 20 kHz on two typical HCl and HF/HNO_3 aggressive solutions, to evaluate test temperature and test duration.

The results were very scattered and difficult to analyze.

For this reason it was decided to perform more laboratory experiments for a more precise investigation of test temperature and test duration, and to evaluate the effect of other parameters such as ultrasonic frequency, ultrasonic power, test materials, concentration of the solution and presence of dissolved metals in the decontaminating solution. A list of the lab tests is presented in Tab. I.

The tests carried out on metallic specimens from the Garigliano BWR on the decon process with ultrasound in aggressive chemicals allow us to understand the effect of the different process parameters, although the definition of a mathematical relationship did not appear to be possible.

With reference to the formula mentioned above we can remark that:

- the pressure of the liquid, the surface tension of the fluid and the solution viscosity do not play any substantial role in decontamination;
- when decontaminating materials with very adherent oxide layers the following test conditions have been found to be the best in terms of decontamination factors:
  - . ultrasonic frequency, f : 22 kHz (see Fig. 1)
  - . ultrasonic power, w : > 25 W;
  - . test temperature, T : 60°C (see Fig. 2)
  - . test duration, t : 30-60 min;
  - . test solution, c : 1% vol. HF/5% vol. HNO_3;
- when decontaminating materials with not so adherent oxide layers, some of the test conditions mentioned above can be varied so as not to damage the base material; in particular the ultrasonic frequency can be raised to 40 kHz, the test temperature can be lowered to 40°C and the 4% vol. HCl solution can also be used as chemical;
- the chemical solution should be renewed when the concentration of dissolved metals reaches to 5 g/l.

3. Decon tests on valves (B.5)

After the lab testing phase, the decon process was tested by in-scale experiments and carried out at Garigliano BWR Power Station in order to evaluate the real decontamination effectiveness. Two contaminated valves from the Garigliano BWR were used in particular.

The experiments were carried out at Garigliano BWR Power Station, in September 1988, using a special US machine, which was designed and acquired to test a similar decon process on pin tubes arising from a preheater (CEC-ENEL Contract FI-1D-0023). The tested 2" valves were cut from a draining line of a SSG and they were contaminated similarly to the valves used in the decon tests on DECO loop without US.

They were tested in similar conditions (I phase: US in water; II phase: US in HF/HNO_3 solution); in the second valve the packing was removed before the test (Tab. II).

The main comments which can be made on these experiments are the following:

- the valves are not completely decontaminated; some radioactivity remains in particular zones of the valve such as crevices and areas not wetted by the solution;

- the part that retains most of the radioactivity is definitely the obturator (plug) at the end of the stem, in fact the obturator is fixed to the stem by means of a weld leaving a deep crevice where the radioactivity tends to concentrate and to remain;
- in terms of total removed radioactivity the valve tested after the packing removal shows a better decontamination (97% of removed radioactivity as opposed to 95%) and the parts near the packing are better decontaminated than in the valve tested with the packing installed;
- some external parts of the valves, the handwheel in particular, were slowly decontaminated during the tests. This was due to the diffusion of radioactivity in the process solution and to the redeposition which mainly affected the dirty surfaces.

Comparing these results with the results obtained in the decon tests on valves in the DECO loop, the following observations can be made:

- the profiles are similar, therefore the decontamination processes with in flow aggressive chemical (DECO loop) or with US appear to behave in a similar way;
- in terms of decon effectiveness, in the complex areas in particular, the action of the flow, in the tested valves, appears to be greater than the action of US. It should however be remembered that in the in-scale US testing much of the chemicals were exhausted by the external surface of the valve;
- looking at the US data in water it appears that the use of a US machine with a high power gives a better decontamination than the usual machine used to preclean the valves tested in the DECO loop.

4. Cost evaluation (B.7)

The tested new decon process appears still to be in a preliminary stage of development and a precise cost evaluation was not possible.

Table I - Decontamination process with US in aggressive chemicals. Schematization summary of laboratory testing on AISI 304 contaminated specimens from the Garigliano BWR.

Lab. Test series	Investigated parameters (*) (and ranges)	Test material (**) (and other constant parameters)	Number of tests
1	- T, 25-80°C - t, up to 60 min - US-f, static, 21-22 and 40 kHz - US-p, 0-14-25-27 W/l	GP 4% HCl or 1.5% HF/5% HNO_3	68
2	- T, 25-80°C - t, up to 60 min - US-f, static, 22 and 40 kHz - US-p, 0-25-27 W/l	GS 4% HCl or 1.5% HF/5% HNO_3	30
3	- % vol. HF, up to 3 - % vol. HNO_3, up to 10 - t, up to 60 min	GP 60°C 40 kHz	20
4	- dissolved metals, up to 10 g/l - US-f, static and 40 kHz - US-p, 0-27 W/l - t, up to 60 min	GP 60°C 1.5% HF + 5% HNO_3	10

(*) T = test temperature; t = test duration, US-f = ultrasonic frequency; US-p = ultrasonic power; (**) GP = 24" primary pipe; GS = divider plate of SSG.

Table II - Operating conditions in the in-scale testing on US/aggressive chemical decon process.

Test	Phase	Step	Chemical solution (% vol.)	Duration (min)	Temperature (°C) Initial	Temperature (°C) Final	Position of the valve in the tank
First	I		Water	30	60	60	Handwheel top
	II	1	3 HF/5 HNO_3	30	60	80	Handwheel top
		2	3 HF/5 HNO_3	30	40	53	Handwheel top
		3	3 HF/5 HNO_3	30	40	55	Handwheel top
Second (*)	I		Water	30	60	60	Handwhell top
	II	1	3 HF/5 HNO_3	30	40	52	Handwheel top
		2	3 HF/5 HNO_3	30	40	55	Handwheel bottom

(*) Valve disassembled, packing removed and reassembled.

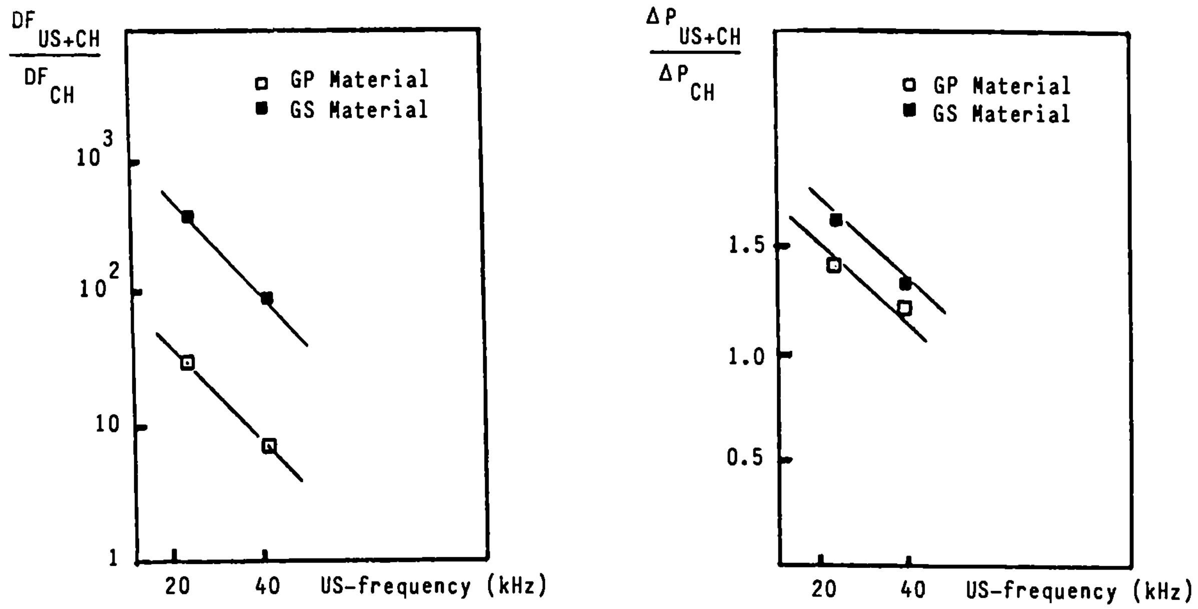


Figure 1 – Decontamination by ultrasound in aggressive chemical solution of Garigliano BWR contaminated samples: effect of ultrasonic frequency in 1.5% HF + 5% HNO_3 solution ((*) see Tab. I).

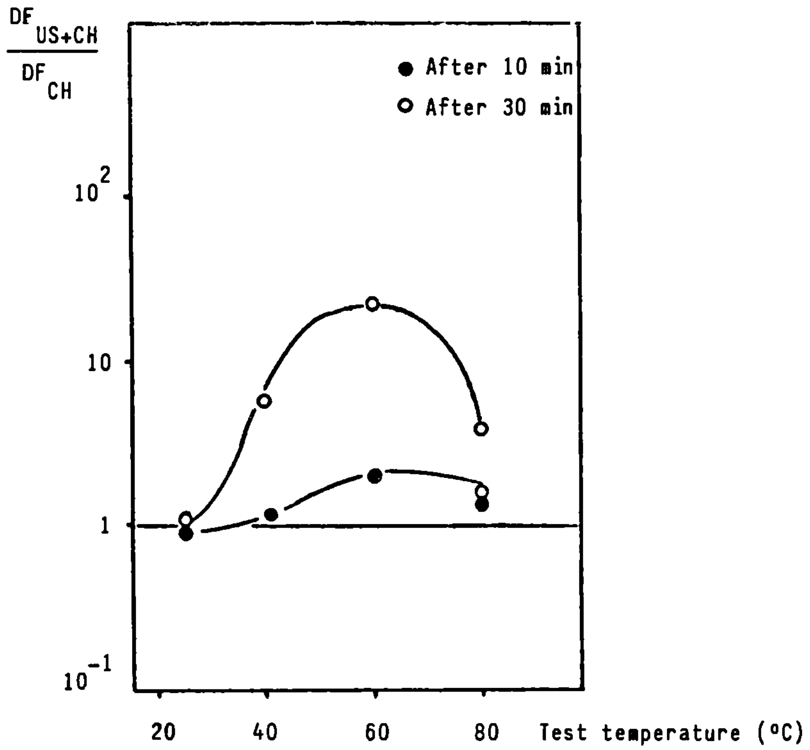


Figure 2 – Decontamination by ultrasound in aggressive chemical solution of Garigliano BWR contaminated samples: ratio between decontamination factor with US and without US, versus test temperature on GP materials in the 1.5% HF + 5% HNO_3 solution, at 22 kHz.

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

Contractor: Commissariat à l'Energie Atomique, CEN-Cadarache, France
Contract N°: FI1D-0003
Working Period: January 1985 - April 1988
Project Leader: G. Brunel

A. Objectives and Scope

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

The aim of this research is to study the following new decontamination techniques with a view to their application in the dismantling of nuclear installations:
- spraying of gels,
- electrolytical swab and jet,
- abrasive water blasting.

These techniques are expected to usefully complement the established methods (immersion in chemical bath, electrolytical bath, high-pressure jet) developed in a previous study (Ref.: EUR 10043).

B. Work Programme

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

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

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

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

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

C. Progress of Work and Obtained Results

The work has been completed and the final report is now under publication.

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

Contractor: Kraftanlagen Aktiengesellschaft, Heidelberg, Germany
Contract N°: FI1D-0004
Working Period: November 1984 - December 1987
Project Leader: A. Steringer

A. Objectives and Scope

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

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

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

B. Work Programme

B.1. Literature survey for identification of the available information on already existing experience.

B.2. Selection of electrolytes other than phosphoric acid, promising easier conditioning and waste disposal.

B.3. Test series on contaminated and non-contaminated samples in order to optimise the electrolytes with regard to decontamination efficiency (effect of chemical additives, of modifying process parameters,...).

B.4. Optimisation of the process to minimise the final waste volume.

B.5. Development of procedures to extend the lifetime of electrolytes, in particular by continuous filtration.

B.6. Processing of selected electrolytes (sediment elimination, salt precipitation, solidification of sludges, volume reduction of the residual liquid, solidification of electrolyte residues).

B.7. Investigations about "on-the-job-safety": chemical aggressiveness, formation of toxic products, explosion hazards, etc.

C. Progress of Work and Obtained Results

The work has been completed and the final report is now under publication.

2.5. Optimisation of Filtering Systems for Various Concrete Decontamination Techniques

Contractor: Salzgitter AG, Salzgitter, Germany
Contract N°: FI1D-0005
Working Period: January 1985 - June 1987
Project Leader: W. Ebeling

A. Objectives and Scope

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

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

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

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

B. Work Programme

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

C. Progress of Work and Obtained Results

The work has been completed, the final report is available as EUR report No. 11995.

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

Contractor: Gesellschaft für Nuklear-Service mbH, Essen, Germany
Contract N°: FI1D-0029
Working Period: January 1985 - June 1986
Project Leader: K. G. Janberg

A. Objectives and Scope

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

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

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

This study takes into account the nuclear installations in Germany.

B. Work Programme

B.1. Review studies

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

B.1.2. Review of existing decontamination methods.

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

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

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

C. Progress of Work and Obtained Results

The work has been completed, the final report is available as EUR report N° 11149.

2.7. Remote Electrochemical Decontamination for Hot Cell Applications

Contractor: United Kingdom Atomic Energy Authority, Harwell Laboratory, United Kingdom
Contract N°: FI1D-0033
Working Period: April 1986 - July 1989
Project Leader: A. D. Turner

A. Objectives and Scope

The primary aim of the programme is to develop and evaluate remote liquid-based decontamination systems for metal surfaces. The bulk of the waste volume should be reduced to a reuse or low-level waste disposal category, while concentrating most of the activity in a small volume suitable for immobilisation. The goal of the development programme is to test these techniques in both alpha-active and alpha-beta-gamma hot cells in order to ascertain their usefulness as a component of an overall decommissioning strategy. As a result of the radiological environment, particular emphasis will be placed on remote operation in order to reduce occupational radiation exposure.

Two types of techniques based on the electrochemical dissolution of thin surface layers of the substrate will be investigated: immersion of small items in tanks for electroetching, and in-situ electropolishing. In both cases, reagents will be chosen with their subsequent disposal in mind.

B. Work Programme

B.1. Investigation of immersion electroetching.
B.1.1. Optimisation tests on the synthetic and genuine waste samples.
B.1.2. Design and construction of a full size unit with ancillary electrolyte management system.
B.1.3. Testing of this unit inactively, and in hot cell facility.
B.2. Investigation of in-situ electropolishing.
B.2.1. Optimisation tests on inactive, synthetic and genuine waste samples.
B.2.2. Design and construction of an automatically controlled unit for use with a remote handling system.
B.2.3. Testing of the unit inactively and in the high alpha-beta-gamma active handling facility.

C. Progress of Work and Obtained Results

Summary

The procurement, fabrication and assembly of a 0.3 m^3 immersion tank decontamination facility has been finished, and inactive commissioning is virtually complete. A dynamic reference electrode based on the evolution of oxygen from platinum under an intermittently applied constant current has been selected as the best system to optimize decontamination effectiveness on the basis of the independence of its response to acid concentration, temperature and the presence of dissolved iron - as well as a negligible maintenance requirement. Zr together with a Pb(II) additive to the electrolyte has been identified as the optimum cathode system to minimize NO_x to < 3 $\mu\ell$/C and completely suppress H_2 from 1-5M HNO_3 electrolyte. Active tests with this equipment will be carried out for the remainder of the contract.

Inactive trials on the mobile hot spot electropolishing decontamination unit have concluded with the demonstration of a range of improvements identified previously. In particular, the use of only 6M acid has been shown to be satisfactory - even on 1 mm thick stainless steel. Silicone front face seals have been shown to be stable for prolonged periods in this medium. The system has also been modified for active use by providing an in-cell effluent tank. This will be demonstrated on active samples until the end of the programme. Robotic handling trials have also been carried out with a PUMA 560, using inductive proxistors to assist in head placement and a miniature multistage air ejector as a source of vacuum to retain the decontamination head in position and to suck out any residual electrolyte after the completion of treatment.

Progress and Results

1. Construction and commissioning of full size immersion electroetching unit (B.1. B.1.2, B.1.3)

The fabrication of the 0.3 m^3 immersion electrolytic decontamination tanks and ancillary equipment was completed and delivered to Harwell in June 1988. These units together with pumps, valves and sensors have now been assembled into an integrated plant and inactive commissioning is virtually complete. The unit comprises three main components:- the decontamination tank itself, an interim electrolyte storage tank and a mobile service trolley (Ref. /1/). The latter is operated from an inactive area and carries sealed stock acid tanks, the power supply and a microprocessor controller. The latter provides a way of automating the system - permitting safe operation without the need for constant operator supervision.

As part of the decontamination system, a reference electrode is required to optimize the effectiveness of the process through adjustment of the cell voltage. From a range of five potential systems identified previously, a dynamic electrode based on the evolution of oxygen from platinum under a controlled pulsed current has been identified as the most suitable for active use. Not only is it completely maintenance free, but its response is essentially independent of acid concentration, temperature and the presence of dissolved iron.

Work on the selection of cathodes has now also been completed with the identification of zirconium plus a 3 mM Pb(II) additive as the optimum system for the suppression of NO_x from a 5M nitric acid electrolyte (< 3 $\mu\ell$/C). Hydrogen evolution was undetectable. The reason for the minimization of any off-gas is that 50% of the cathodic current yield results in the reduction of NO_3^- to NH_4^+. Performance of this system in 1M HNO_3 is similarly impressive. As a result, no special ventilation system has had to be incorporated into the decontamination tank, thus further simplifying its construction. Not only is the evolution of insoluble gases minimized, thus suppressing the formation of any active aerosols, but the production of toxic or flammable gases is also avoided.

Due to delays in the receipt of several items of equipment, this programme has now been extended until March 1989. Routine decontamination trials (B.1.3) will be carried out until this date.

2. Construction and inactive testing of a mobile electropolishing hot-spot decontamination unit (B.2, B.2.2, B.2.3)

Further work on the prototype mobile hot-spot electropolishing decontamination unit has been undertaken in the light of inactive commissioning trials carried out previously (Ref. /1/). In order to minimize the volume of secondary waste arising and to improve the stability of the elastomeric front face seals in the nitric acid electrolyte, the effect of further reductions in acid strength were investigated. As a result, stainless steel sheet of down to 1 mm thickness can be satisfactorily electropolished in only 6M HNO_3 (Figure 1). The chemical stability of a range of potential seal materials was subsequently investigated by long term exposure in 6, 8 and 10M acid to identify the optimum system. For the continuation of prototype development, unloaded silicone tube filled with a soft silicone foam was confirmed as being the most suitable - both from the point of view of its chemical and radiation stability, but also the ease with which it could be fabricated into soft O-ring seals. Viton and Nordel were identified as polymers more suited to injection-moulded fabrication on a commercial scale.

A range of modifications have been successfully made to the decontamination head (Ref. /2/) to make it sufficiently reliable for active trials. These included improving the means of front face seal retention, reducing feed pulsation arising from the use of a diaphragm pump and the evaluation of inductive proxistors in place of the simple mechanical switch as a surface placement sensor. In addition, the unit has been modified for hot-cell use by providing an in-cell effluent tank and a miniature multistage air ejector and aerosol trap to suck out any residual electrolyte after decontamination. This will be used shortly to demonstrate the complete system on active waste samples for the remainder of the programme.

Robotic handling trials have also been carried out using a PUMA 560 in the Harwell Robotic Demonstration Facility (Figure 2). The decontamination head was successfully placed and sealed to both horizontal and vertical surface and water subsequently passed through the system to simulate the electrolyte. Inductive proxistors were satisfactorily demonstrated as a non-mechanical method of sensing when the head was in position on the surface to be treated. In addition, a miniature multistage air ejector proved effective as an alternative source of vacuum - both to retain the decontamination head in position, and also to suck out residual electrolyte. Before the end of the programme, a PUMA 762 (10 Kg payload) robot will be used to demonstrate an electropolishing decontamiantion head with an integral activity monitor (G-M tube) and proximity detector for automated robotic placement.

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/2/ TURNER, A.D., JUNKISON, A.R., POTTINGER, J.S. and DAWSON, R.K. "Electrolytic treatment device". UK Patent GB 2203756A.

/3/ TURNER, A.D., JUNKISON, A.R. "Electrolytic treatment", GB 2205584A.

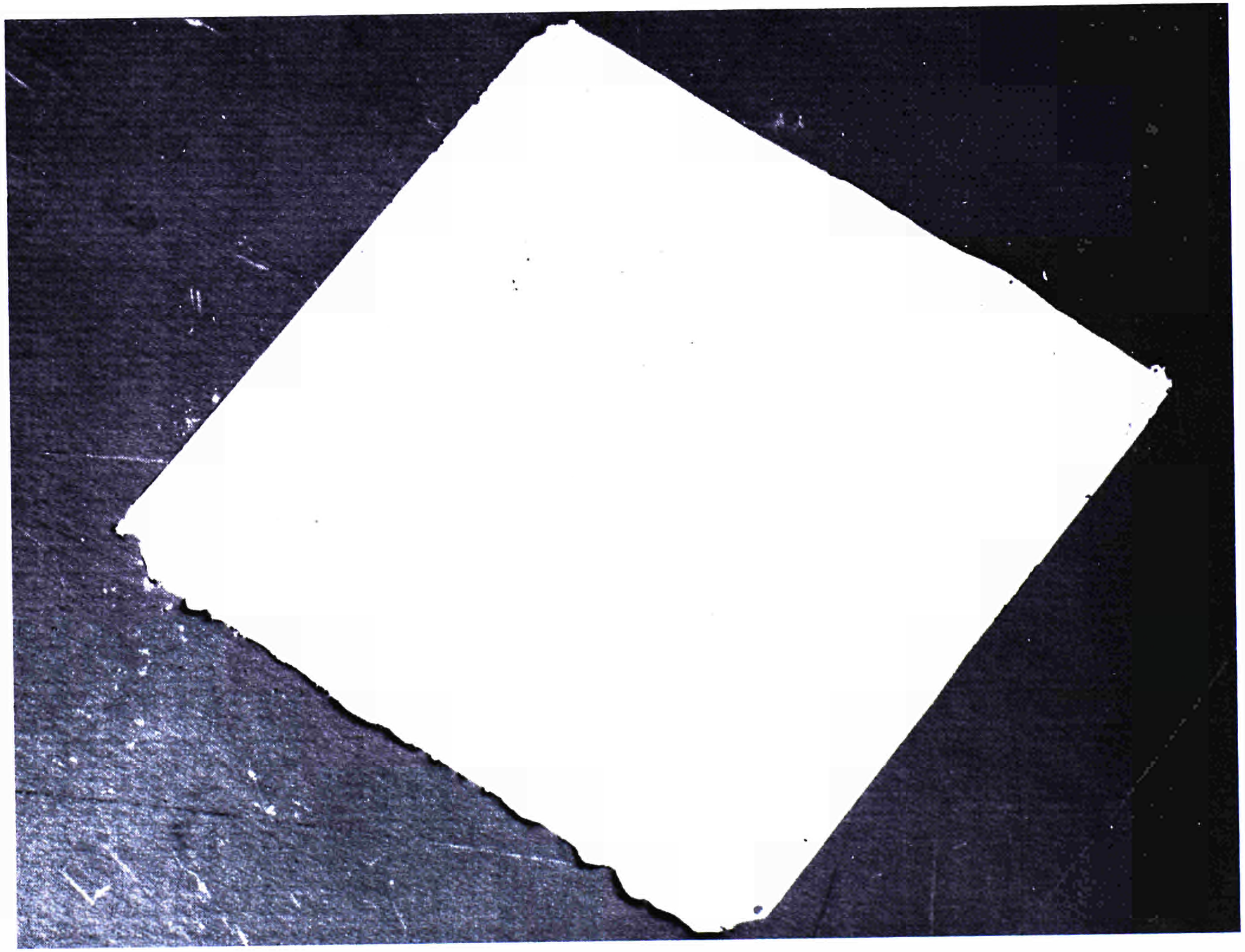

Figure 1: High rate electropolishing on 1.5 mm thick 2B mill rolled 321 stainless steel sheet obtained in 15 seconds with the Harwell decontamination head.

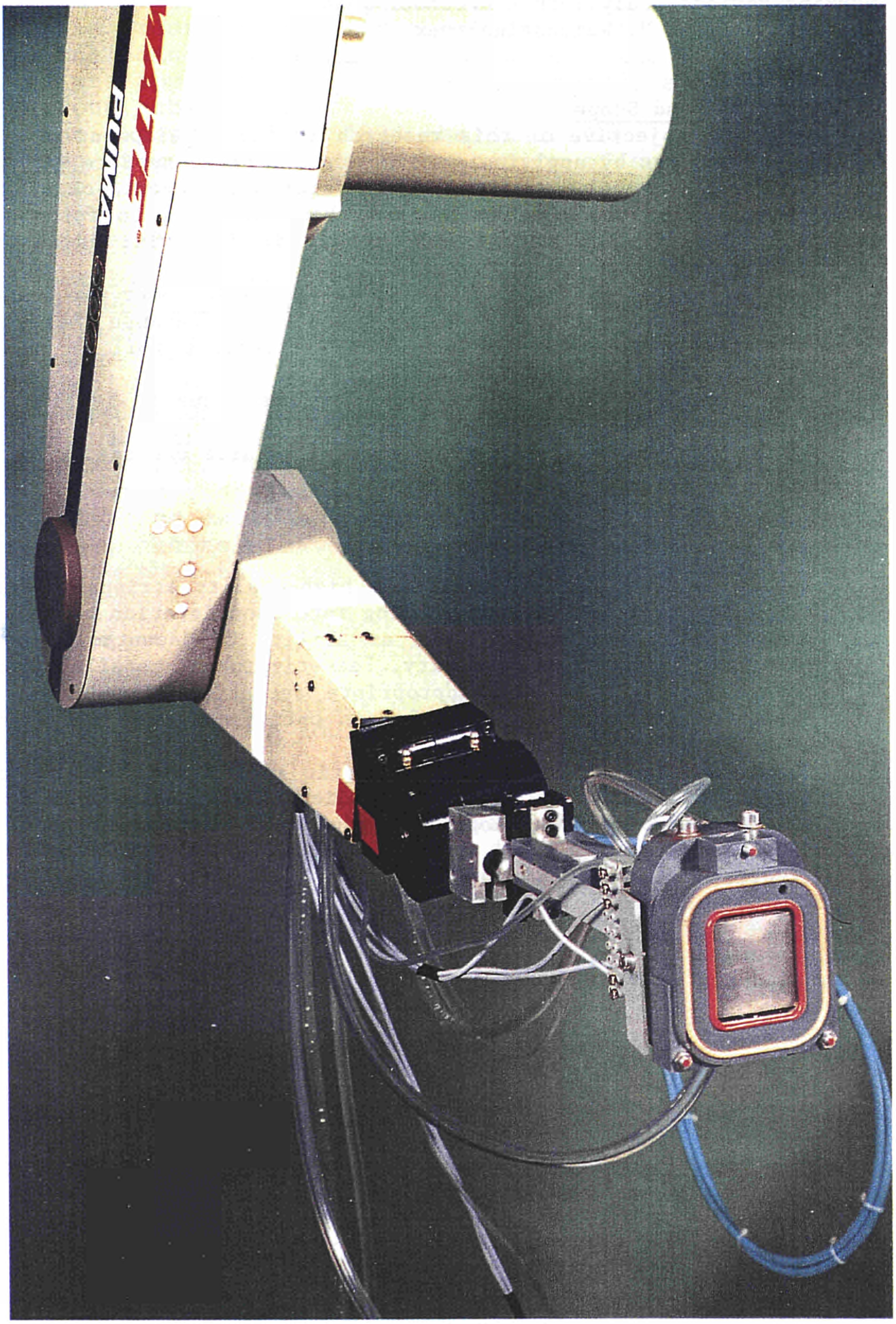

Figure 2: Prototype hot-spot decontamination head being manipulated by a PUMA 560 robot.

2.8. Decontamination with Pasty Pickling Agents Forming a Strippable Foil

Contractor: Max Morant Chemische Fabrik, Aschau, Germany
Contract N°: FI1D-0034
Working Period: July 1986 - December 1988
Project Leader: H. Weichselgartner

A. Objectives and Scope

The main objective of this research is the development of a decontamination procedure by applying onto the contaminated surface (in a one-step or multi-step process) pasty, chemically aggressive agents causing dilution and absorption of the contaminant and then hardening to forma strippable foil. The use of such a foil will result in following advantages, with respect to usual techniques:

- sensibly shorter operating duration resulting in lower personnel doses;
- reduction of the arising secondary waste volume because there is no need for washing; the volume of the spent strippable foil is much smaller than currently used water volumes;
- optimal conditioning of the radioactive waste due to its fixation in a solid (foil);
- an accidental contamination in a controlled area can easily be fixed and covered avoiding its propagation.

B. Work Programme

B.1. Development and optimisation of a high-quality strippable foil, ready for industrial application, taking into consideration various physical and chemical properties, such as ability to decontamination, strong adhesion, appropriate viscosity, leak-tightness, tensile strength.

B.2. Development of the most appropriate operation procedures, including different deposition methods and an optimisation of the layer thickness.

B.3. Commissioning testing under realistic conditions, including various types of surfaces and tests with radioactive conditions in hot cells.

B.4. Development of a technology for industrial application, including the preparation of a users' manual, taking account of gained practical experience.

C. Progress of Work and Obtained Results

Summary

Hitherto, most of the decontamination tests have been performed with a few standard mixtures in a single-step mode. This means that, in order to simplify the tests, all components of the foils, including hardening resins, were mixed togehter and the mixture was applied in a single-step procedure by brushing it onto various substrates.

During the second half of 1988 the basic idea of using a multiple-step procedure for decontamination of surfaces, equipment, walls and the like was further investigated.

Progress and Results

1. Recapitulation and evaluation of former investigations (B.1.).

Results of earlier investigations show that polyvinyl alcohols and polyvinyl acetals are very suitable as matrix material for strippable foils. It was also found that those foils in which polyvinyl alcohols are used as matrix material are formed mainly by evaporation of the solvents applied during preparation of the foil material mixture.

In accordance with current regulations and recommendations, all those solvents containing halogens such as chlorine or fluorine have to be excluded as ingredients.

In the past, alcohols have therefore been used, care being taken to ensure that the risk of fire or explosion is strictly excluded in preparing, storing and applying the foil material.

These two facts, a) formation of the foil depending on evaporation of the solvents,
b) risk of fire,
constitute a major disadvantage of the prototype foils used up to now. But there are two additional drawbacks with the single-step prototype foil applied in the past:

- the active time interval, e.g. the decontamination procedure itself, is limited by the evaporation time of the solvents used during preparation of the mixture,
- the amount of etching agents which can be added to the basic foil mixture is limited by the properties of the solvent. Desirable high concentration of etching agents often result in precipitation, thus yielding non-uniform foils which cannot be wholly removed from the substrate.

2. Investigations of the production of multi-step foils (B.1.)

In the first step a mixture of one or more etching agents and softeners, e.g. polyethylene glycols (PEG), was brushed onto metallic substrates.

In the second step polyvinyl alcohols dissolved in a mixture of water, ethanol and/or butanol were applied. Different time intervals between these two applications were selected in order to achieve the optimum decontamination effect.

Aluminium, copper, brass and mild steel were chosen as substrates. When phosphorous acid was selected as etching agent, the surface of the steel substrates was etched very intensively, so that the foils formed after application of the polyviol mixture could not be wholly removed.

Additionally, sodium citrate and mixtures of formic acid with sodium citrate were applied. The latter mixture again showed strong pickling effects.

At present it is difficult to evaluate the results obtained with the multi-step foil, because only a few tests were performed. In Table I an assessment of some tests is presented.

3. Decontamination of tritiated substrates (B.3.)

These tests were conducted at the tritium laboratory L2 at Max-Planck-Institut für Plasmaphysik, D-8046 Garching.

For this purpose small dishes made of teflon, aluminium and copper were contaminated in the following manner:
an aqueous tritium standard (BAKER-INSTAND 3H-W) with known activity (e.g. 107000 dpm on June 1, 1983) and calibrated volume was distributed over a series of dishes (20). After drying, the activities were measured with a wiping test assembly (FAG Kugelfischer). The following value (averaged from several measurements) was obtained: A(o) = 11,900 dpm.
The surface thus contaminated was then coated with different mixtures of the first part of the multiple-stage pickling foil and left for a selected time interval before the second part was applied. After final formation of the foil itself (curing time of the multiple-stage foil) the foil was stripped, mostly without any residue being left and the dishes were measured again.

At first glance, this seems to be a desirable result. But, on the other hand, it is not possible to get further information from these decontamination tests with aqueous tritium standards. It is therefore necessary to conduct decontamination tests with other radionuclides which are more difficult to remove.

4. Methods of application (B.2.)

Simple application techniques such as brushing, painting, rolling or priming are acceptable in laboratory tests but are ruled out if glove-boxes, hot cells and more comples facilities have to be decontaminated.

In the past the single-stage foil has been modified in several cases to try to achieve sprayability by one or more spraying techniques. So far no optimum results have been obtained by the special technique of airless spraying at high pressures. But it was possible to obtain uniform, closed, transparent films with compressed air at 3 to 5 bar.

No spraying tests have been done so far with the multiple-stage strippable foil. It is assumed that it will be readily possible to apply spraying with compressed air.

Nevertheless, airless spraying at high pressures will be the ultimate technique to be applied in commercial application of the multiple-stage strippable foil.

Table I Assessment of multi-step pickling foil

Assessment criteria	Aluminium	Cu/brass	C-steel
Pickling agent		H_3PO_4/Na citrate	H_3PO_4
film formation	good	moderate	moderate
strippability	good	good	moderate
tearing strength		not recorded	
elasticity		soft	
adhesion	good	acceptable	too firm
residues	none	slightly wet	slightly/dry
pickling effect	good	good	very good

2.9. Rack-torch Unit for Remote Decontamination of Concrete

Contractor: Société des Techniques en Milieu Ionisant, Trappes, France
Contract N°: FI1D-0035
Working Period: November 1986 - August 1987
Project Leader: J.F. Routier

A. Objectives and Scope

The decontamination of concrete, in the framework of the decommissioning of nuclear installations, poses a particular problem, due to the migration of contaminants into concrete to a depth of 1 to 5 cm.

The technique of fissuration by rack-torch and rapid cooling is expected to suit application in a hostile environment and to involve notably lower radiation exposure of the personnel than the methods used nowadays such as hammering.

The objective of the present research is the setting and perfecting of the fissuration technique by rack-torch with a view to its secure and optimised use in decontaminating concrete structures.

B. Work Programme

B.1. Design and manufacturing of the rack-torch prototype.

B.2. Optimisation of the concrete fissuration technique including the study of motion of the prototype (manual or automatic), the piezo-electric ignition device, the system for aspiration, sedimentation and filtration of aerosols and concrete particles, various types of rack-torch for the scraping of different surfaces.

B.3. Scraping tests on various types of concrete (inactive and contaminated samples).

B.4. Design and manufacturing of the industrial prototype.

B.5. Application of the fissuration technique to a nuclear installation including recommendations for the best use of this technique.

C. Progress of Work and Obtained Results

The work has been completed; for results and conclusions, refer to Annual Progress Report 1987 (EUR report No. 11715).

2.10 Feasibility of Concrete Decontamination Using a Plasma-augmented Burner

Contractor: Société Bertin et Cie, Plaisir, France
Contract N°: FI1D-0063
Working Period: January 1987 - May 1988
Project Leader: C. Morillon

A. Objectives and Scope

The contamination of concrete in nuclear installations is mainly located in the vicinity of the exposed surface, to a depth generally estimated at a few millimeters. Therefore, during the decommissioning of these installations, techniques are preferred which are capable of removing concrete by successive thin layers so as to minimise the quantity of radioactive waste generated. On the other hand, the minimisation of aerosols emission, dangerous to workers, constitutes a second criterion of choice for decommissioning techniques.

Contrary to traditional mechanical techniques, electrocombustion is likely to respond simultaneously to the two previous criteria, causing superficial melting and weakening of the concrete by very high temperatures.

The aim of the present research is to determine by inactive experiments on the existing test bench:

- the efficiency of the plasma burner as regards the shallow destruction of concrete at a temperature which may exceed 3000°C;
- the approximate levels of aerosol emission and NO_x involved in this operation.

B. Work Programme

B.1. Optimisation of the plasma-augmented burner under non-radioactive conditions as a means for removing concrete layers, including the following parameters: concrete structure, gas composition, temperature and velocity of exit gas.

B.2. Application of the plasma burner to two types of concrete, one of them impregnated with non-radioactive cesium chloride for simulation of the contamination.

B.3. Continuous measurements of aerosol and NO_x gas quantities produced and analysis of aerosol concentration and particle size.

B.4. Conclusive assessment of obtained results and elaboration of recommendations for the best application to concrete decontamination.

C. Progress of Work and Obtained Results

Summary

Experimental destruction of concrete samples with the plasma-augmented burner was carried out. It was demonstrated that this technique could remove concrete by superficial erosion and weakening. Furthermore, simulation with non-radioactive cesium chloride showed that the process could decontaminate concrete walls in which Cs-137 is present.

Progress and Results

1. Experimental results (B.2. and B.3.)

The existing test facility was adapted for qualifying the electro-combustion technique. It consists of:

- a plasma burner,
- a test cabin where concrete samples were installed,
- exhaust ducts with measuring points for aerosol emission, NOx levels and temperature of the exhaust gases,
- a scraping cabin for removing the debris after thermal exposure.

Two types of concrete were tested:

- A: standard concrete with rock and sand aggregates,
- B: heavy concrete with baryte aggregates.

In order to simulate the contamination, some samples of concrete A were impregnated with cesium chloride.

Two main types of plasma were selected for the experiments:

- air plasma,
- C_3H_8 plasma + air (plasma + combustion).

High heat fluxes (1.6-1.7 MW/m^2) were achieved by impingement on the exposed concrete surface of a high-temperature high-velocity jet with following characteristics:

- electric power: 50 kW (with 380 V alternating current)
- plasma gas mass flow: 3 g/s,
- exhaust gas temperature: 4300 - 4500 K,
- gas velocity: 135-150 m/s.

Several samples of concrete A and B were treated by the plasma-augmented burner.

The degraded concrete particles are removed from the surface and are falling into the containment without having reached the fusion point.

The main results of aerosol analysis are reported in table 1.

The mean mass aerodynamic diameter of aerosols was 1 µm during thermal stripping, and 9 µm during scrapping for removing the concrete debris.

2. Conclusive assessment (B.4.)

The experimental survey showed that the process is capable of decontaminating walls impregnated with Cs-137. Furthermore, the C_3H_8 plasma + air configuration seems to be more promising than the air plasma configuration, for the following reasons:

- low-level of NOx: 20 to 40 ppm for the first configuration - 1500 to 1700 ppm for the second one,
- higher thermal efficiency of the impinging jet.

In all the experiments, no molten phases occurred and after a thermal exposure of 30 s, an erosion depth of 3 to 5 mm could be obtained. With a larger exposure time, the erosion depth is not proportionally greater.

The feasibility of this technique was demonstrated, nevertheless and in order to characterize fully the technique, the following step of development would be the design and construction of a pilot unit to be tested under more representative conditions (larger samples and continuous sweeping).

The work has been completed and the final report is now under publication.

TABLE 1 - Aerosol emissions - Thermal (T) stripping with mechanical (M) scraping

Samples references	Concentration mg/m³		Slag mass / Mass of removed concrete in %		Aerosol mass / Mass of removed concrete in %		Mean aerodynamic diameter in µm		
	T	M	T	M	T	M	T	M	
A 372 - 10	13,8	594	17,6	100	0,13	0,41		12,1	Concrete A
A 376 - 11	55	686	9,36	91,8	0,56	0,56	1		
A 370 - 12	29	269	21,3	95,8	0,29	0,70		9,8	
A 371 - 24	16	212	35,6	98,1	0,07	0,45	1,38	4,6	
A 374 - 25	6	337	9,0	94,7	0,03	0,88			
A 378 - 26	36	308	11,2	95,3	0,32	0,62	0,64	7,2	
A 411 - 13	38	307	28,3	94,4	0,46	0,52	0,96	8,8	Concrete B
A 420 - 14	26	722	30,4	94,5	0,28	0,84		9,1	
A 421 - 31	48	904	40,4	97	0,49	0,62	0,47	10,1	
A 418 - 32	26	581	7,75	92,4	0,29	0,44		-	
D28TM - 33	39	325	15,8	98,2	0,15	1,05	1,45	7,7	Concrete A impregnated with CsCl
DOTM - 34	47	296	50	93,6	0,09	1,16		2,1	

2.11 Closed Electropolishing System for Decontamination of Underwater Surfaces

Contractor: Equipos Nucleares S.A., Madrid, Spain
Contract N°: FI1D-0065
Working Period: January 1987 - December 1988
Project Leader: E. Benavides

A. Objectives and Scope

The objective of this research is to develop an electropolishing technique for the decontamination of large stainless steel surfaces in flooded systems without loss of electrolyte.

The scope of the programme involves essentially:

- the implementation of a closed loop system able to electropolish large stainless steel surfaces without loss of electrolyte;
- the development of various electropolishing cathodes with electrolyte rinsage system for underwater use;
- the underwater application of this technique to contaminated surfaces.

This research is directed mainly at reactor cavities and fuel storage pools of LWRs, which need decontamination in maintenance as well as in decommissioning. Electropolishing is able to decontaminate completely the stainless steel surface enabling it to be treated afterwards as conventional scrap, reducing thus the amount of waste generated during the decommissioning work.

B. Work Programme

B.1. Development of the closed electropolishing system for underwater remote operations including the study of various types of cathode.

B.2. Construction of a selected system.

B.3. Underwater testing on non-radioactive surfaces (stainless steel AISI 304 and 316 and welding joints) with different roughness, at a water depth up to 20 m.

B.4. Underwater application to contaminated surfaces of nuclear installations including determination of the decontamination factors.

B.5. Assessment of the arisings and the treatment of secondary waste, and evaluation of the cost and radiological consequences of the decontamination.

C. Progress of Work and Obtained Results

Summary

An automated system was implemented to the electropolishing installation to control all the process. Also an electropolishing head covering an area of 2 m^2 in one step was developed and fabricated. Laboratory tests were realised to investigate treatment and/or purification of the radioactive waste.

Progress and Results

1. Development of an automated system to control all the underwater electropolishing process (B.2. and B.3.)

A new installation was erected with stainless steel tanks (three of 700 l each), three microfilters (10 µm), three Ph-meters, one vacuum tank with two diagram pumps, stainless steel pipes, vitrified valves pneumatically controlled from a control panel, which is automatised with a computer to operate all the electropolishing process. In this panel, a synoptic chart of the installation, as well as the instrumentation indicators are shown, with the possibility also to operate manually all the valves of the installation from this panel (figure 1).

2. Development of an electropolishing head covering an area of 2 m^2 (B.2. and B.3.)

As needed to cover a large surface in pool liners, the electropolishing head has to run along a great distance.

The first electropolishing heads developed were fabricated thinking about the limitation of the electrode surface. That means, with a rectifier of 3000 A, a surface not bigger than 1000 cm^2 (0.1 m^2) was needed to reach the appropriate current density for good results.

With these heads, the continuous movements of the underwater head are discarded, along the pool liner wall, because no washing can be applied. The way chosen is a batch process in which the area electropolished is washed before the head goes to a new position.

Disadvantage with this batch process using electrodes of 500 x 200 mm is that the contractor needs a lot of interruptions in an electropolishing process for a 10 m high pool liner.

Also, the mechanical system to move the head along the wall and to procure all the movements needed is very complicated and implied room enough inside the pool.

A new electropolishing head was developed (figure 2), taking into consideration all problems detected in the first stage of the investigation; this new head has the following characteristics:

- size: 2000 mm high, 1000 mm wide, 150 mm thick;
- it creates a chamber of 2000 x 1000 x 50 mm which is filled with electrolytes;
- in this chamber, an electrode with a working surface of 1000 x 100 mm runs up and down the chamber, electropolishing an area of 2 m^2 in one step, so that less interruptions are needed in the process along large surfaces of 10 m height;
- the frame of the head is made from an isolated material to avoid current bypass;
- to get an electrolyte chamber that procures a large work surface but a little volume of electrolyte, the thickness of the chamber was designed with 50 mm. This space is enough to allow the electrode displacement. Therefore, the back of the head is a stainless steel reinforcement plate which deflection is not bigger than 1 mm under water pressure;

- the electrode flowing inside the chamber is distributed by means of a grid located at the entrance of the head chamber;
- the electrode with a working area of 1000 x 100 mm is a stainless steel plate with cooling fins at the back to avoid high temperatures in the electrode;
- a sophisticated mechanism is not needed to move this head because of its big size. No precision is required (as it was for the smaller ones) for the movement to a new electropolishing area. So, with a simple crane, this new head can be moved to new position;
- with a thickness of 150 mm, this head can be installed in a lot of narrow places which could not be reached otherwise;
- during the electropolishing process, the head is filled with electrolytes across its lower side, and the extraction is done at the upper side. So, the gases produced during electropolishing are easily extracted.

To empty the head, the circuit is inverted and the liquid (electrolyte or washing water) is extracted by the lower side of the head, to facilitate its way out.

3. Assessment of the arisings, treatment of secondary waste, and determination of the decontamination factor (B.4. and B.5.)

Till now, all the electropolishing heads for electropolishing of pool liners were not too large, and these liners have mainly large flat areas. So, with the research and development of an electropolishing head covering an area of 2 m^2 in one step, a great advance was reached in the electropolishing of large flat areas, not only under water, but without water. This head having a thickness of 150 mm, only a small space is needed, and the head itself is easy to handle. An electropolishing rugosity of 0.5 µm (average) was obtained under water (figure 3), with all equipment working properly.

By the electropolishing process, layers of atoms are removed from the surface of the object (0.05 mm) with resultant removal of contamination and smoothing of the surface (0.5 µm Ra). The decontamination factors which can be reached with the electropolishing process under water, are those usually obtained in electropolishing decontaminations, which can be situated between DF 100 to 1000. Result of all this process is that the electrolyte removes the contamination by a metal surface dissolution process. Laboratory tests were directed to precipitate heavy metals (iron, cobalt, etc.) with basic solutions, so that the radioactive waste volume can be reduced after filtering the precipitates, they can be treated for storage as radioactive waste.

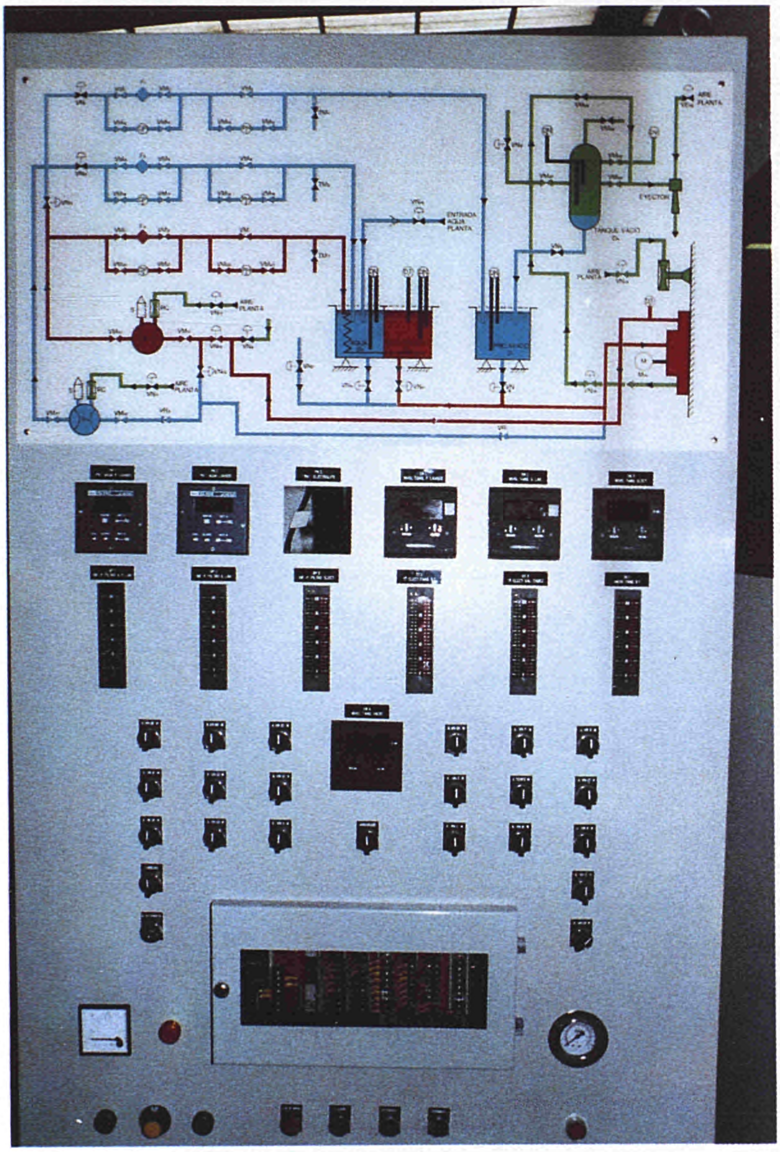

Figure 1: Control panel and synoptic display of the installation

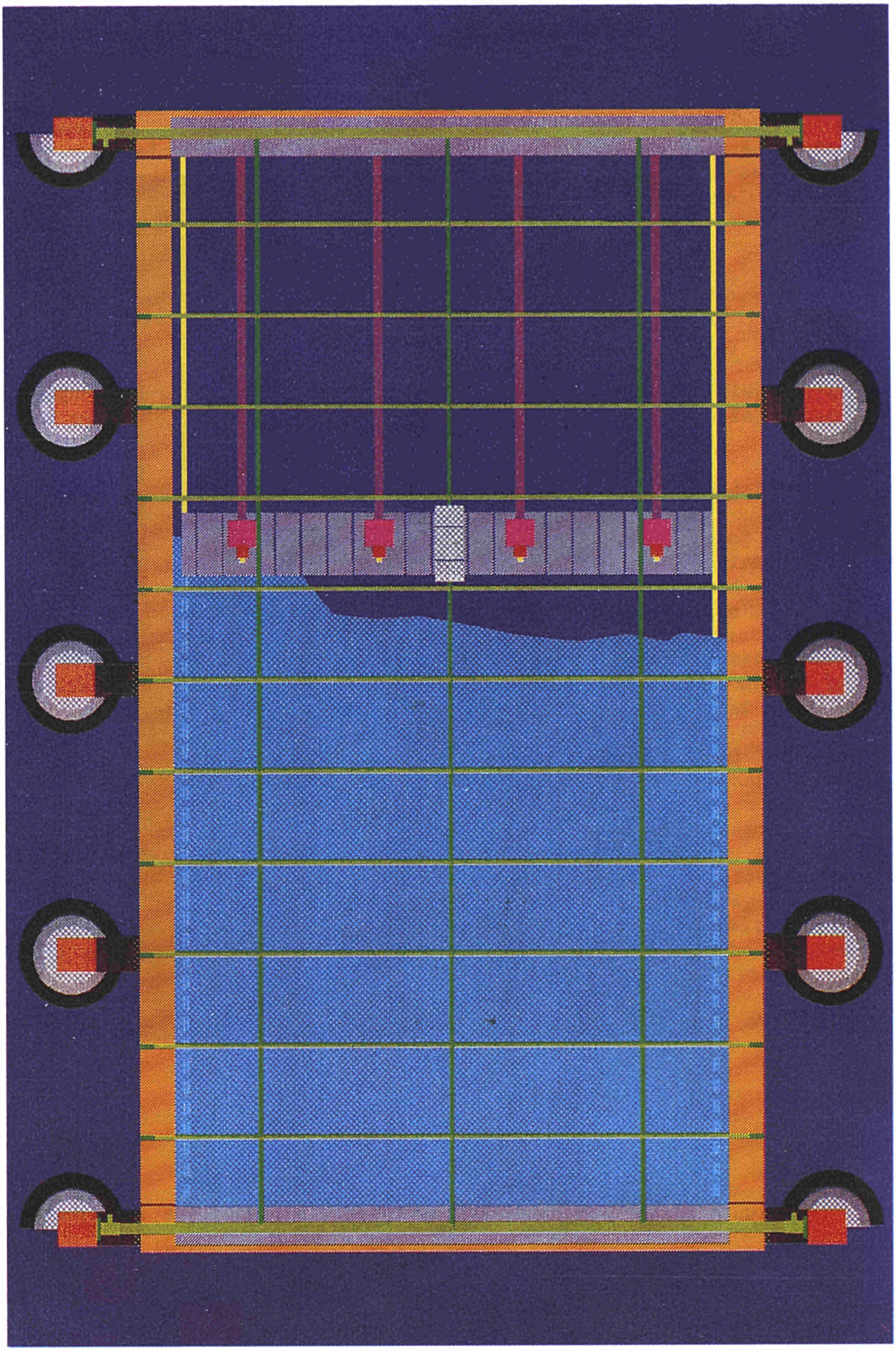

Figure 2: Electropolishing head

Figure 3: Electropolished surface

2.12 Development of Vibratory Decontamination with Abrasives

Contractor: Equipos Nucleares S.A., Madrid, Spain
Contract N°: FI1D-0066
Working Period: January 1987 - December 1988
Project Leader: E. Benavides

A. Objectives and Scope

The objective of the research is to develop a technique for the decontamination of nuclear components using vibratory equipment with self-cleaning abrasives generating a minimum quantity of waste.

The development is aimed at the following goals:

- decontamination of metal and non-metal items,
- decontamination without personnel attendance,
- high decontamination factors with minimum waste generation,
- self-decontaminating equipment which lowers the occupational radiation exposure and can be installed as mobile units.

The research will seek for the best abrasives which do not retain contamination but generate a high decontamination factor and a minimum volume of waste, as well as for the equipment required to handle nuclear components during the process. The researched process will not require operation personnel except for loading and unloading the components.

B. Work Programme

B.1. Optimisation of the vibratory technique including the study of various frequencies and abrasives.

B.2. Construction of a small prototype of the defined vibratory decontamination system.

B.3. Testing of the prototype on non-radioactive metal (stainless steel AISI 304 and 316) and non-metal (glass, plastics, etc.) samples.

B.4. Decontamination tests on contaminated metal and non-metal samples including determination of the decontamination factors.

B.5. Assessment of the arisings and the treatment of secondary waste, and evaluation of the cost and radiological consequences of decontamination.

C. Progress of Work and Obtained Results

Summary

More tests were realised with the first prototype (vibratory machine) developed, with the obtained results new vibratory equipment was provided, to realise new tests in a closed circuit with a decontamination unit working with solvent.

Progress and Results

1. Development of a new vibratory machine (B.1., B.2. and B.3.)

New tests were realised with the existing vibratory machine, with more pieces as samples having same process results as all the tests realised before. But some mechanical problems were detected in the vibratory machine after hours working. The power transmitted by excentric weights made gaps and the vibratory frequencies changed continuously.

Then a new vibratory machine (figure 1) was provided to solve thé problems described above.

This machine has a torous shape similar to the first machine, where the abrasives and pieces are introduced. An electric motor is suspended vertically from the torous. This motor transmits by means of an axis, on the lower side, a circular movement to an off-center weight system. This system produces a vibratory movement; amplitude can be regulated with the off-center weights.

The inside of the torous is coated with polyurethane to resist the solvent action.

2. Tests of the new vibratory machine in a circuit with solvent (B.3. and B.5.)

The new vibratory machine was tested in a closed circuit with solvent and abrasive media.

A new stainless steel cover was fabricated to tighten the torous and avoid solvent leaks. This cover has a sluice to facilitate loading and unloading of pieces, and several spray nozzles to wash with the abrasive media with solvents.

The abrasive media are the same as used before, as well as the pieces to be tested. The freon unit distillation is the same as used before, because it has been working satisfactorily till now.

The circuit mounted is shown in figure 2, in which the solvent is distilled continuously, and the sludge is collected as waste.

The abrasive process gives the same results as in the first machine, but this machine can work more hours without falling out of adjustment.

3. Assessment of the arisings, decontamination factors, and treatment of secondary waste (B.4. and B.5.)

Decontamination factors which can be reached with this decontamination equipment are similar to those reached with other experimental abrasive methods (brush abrasion, dry vibratory finishing, abrasive blasting, etc.) plus the action of decontamination reached with a solvent media. So DF with this developed technique can be situated between 5 and 50.

Vibratory finishing has demonstrated an excellent decontamination technique for tools and large quantities of small items, but larger components require extensive disassembly or sectioning. The maximum size of items that can be processed is about 20-30 cm of wrenches, hammers, screwdrivers and other miscellaneous tools, which can be decontaminated for reuse within an hour, with minimal operator attention.

The solvent action causes removal of oil greases, and the products of abrasion, producing contaminated waste. This waste is treated and reduced in volume and in cleaning the contaminants, the solvent is removed by filtration and evaporation.

Relatively low-boiling point (47.6 °C) of the solvent facilitates its separation from virtually all dissolved organic compounds, and from any water that might also be removed in normal decontamination operations. When processed in this manner, the solvent can be recycled for use, and wastes are confined to the filter elements and the solvent evaporation vessel.

With two techniques used separately in industrial processes, it has been demonstrated that it is possible to transform and join these techniques to develop a new machine useful for nuclear decontamination purposes. So little pieces can be decontaminated by means of an abrasive action, and the waste volume produced can be reduced, be transported by an easy-to-handle solvent, and distilled to concentrate and reduce its volume.

Figure 1: New vibratory machine

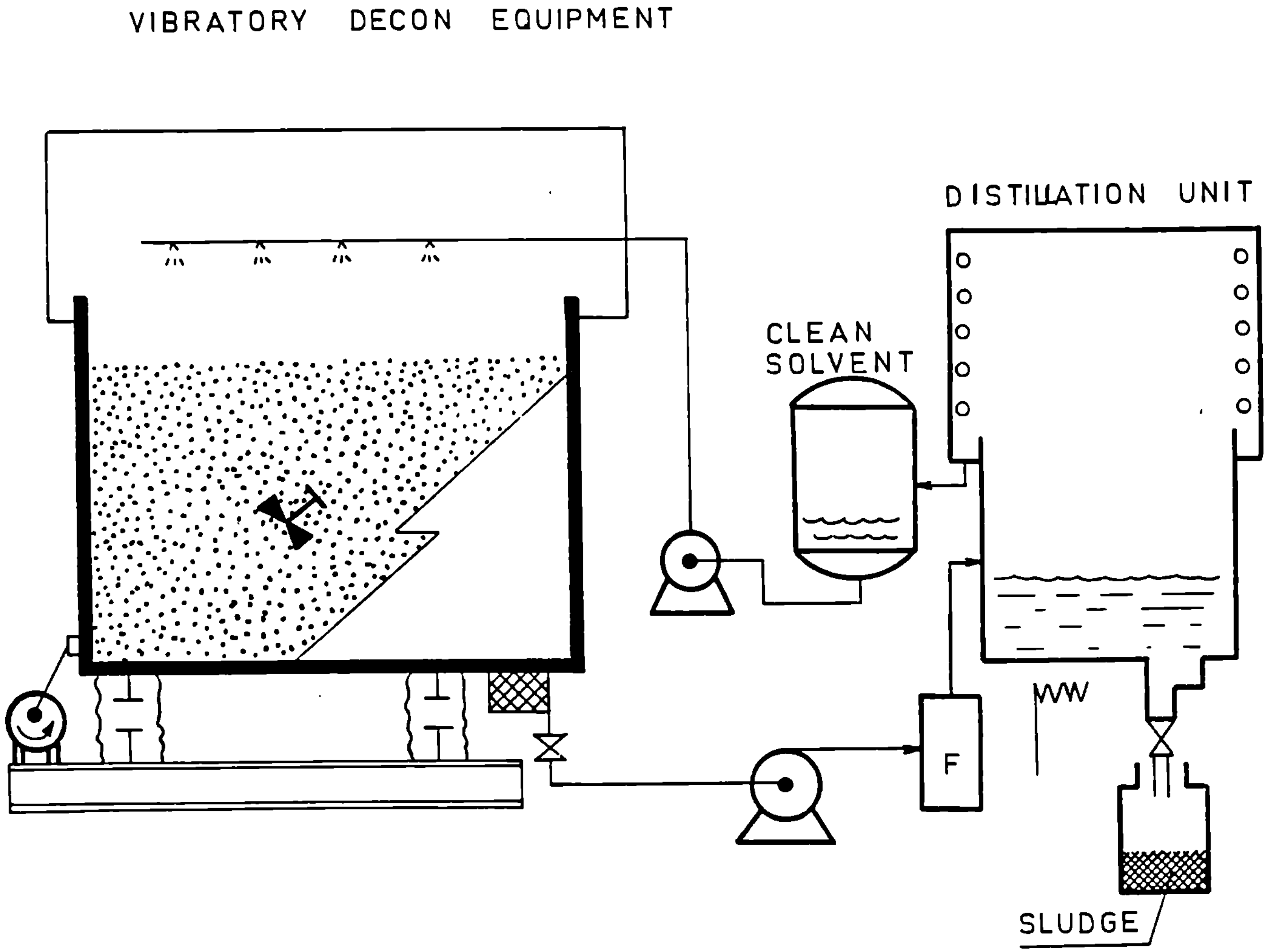

Figure 2 : Vibratory decontamination system

3. PROJECT N°3: DISMANTLING TECHNIQUES

A. Objective

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

B. Research performed under the 1979-83 programme

The following techniques have been tested and developed:
- thermal techniques such as plasma-arc and oxygen cutting and cutting by laser beam;
- mechanical techniques such as sawing;
- explosive techniques for the dismantling of concrete structures.

C. 1984-88 programme

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

The necessary equipment for the remote operation of dismantling and other decommissioning techniques will be an important new aspect for investigation under the 1984-88 programme.

D. Programme implementation

Eighteen research contracts relating to Project N°3 were executed in 1988, including two new contracts concluded that year. The execution of twelve contracts was completed in 1988.

3.1. Ventilation and Filtration Techniques for Thermal Cutting Operations

Contractor: United Kingdom Atomic Energy Authority, Windscale Nuclear Laboratories, United Kingdom
Contract N°: FI1D-0006
Working Period: October 1984 - June 1988
Project Leader: J.R. Wakefield

A. Objectives and Scope

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

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

Co-operation with CEA Saclay (contract N°FI1D-0007) will take place over the work period and will take the form of information exchange and the interchange of apparatus and personnel.

B. Work Programme

B.1. Literature survey for identification of former work and of alternative techniques.

B.2. General investigation into aerosol behaviour for various cutting techniques.

B.3. Construction of a filtering rig and detailed study of various filtration systems.

B.4. Assessment of various tested filter systems for their appropriateness in decommissioning applications with active aerosols.

B.5. Execution of full-size ventilation trials including aerosol deposition in ductings and plate out on the decommissioning machine.

C. Progress of Work and Obtained Results

Summary

This report details the work during the first half of 1988 until July when the contract was completed.

Trials are described on a High Gradient Magnetic Separation (HGMS) prefilter in the laser fume generation facility. The efficiency of the unit was low, typically 20%. This was later increased to 35% after cleaning of the filter medium. The work was terminated following a decision not to use the HGMS as a prefilter.

Studies have continued into how to remove the radioactive burden from an electrostatic precipitator (ESP) prefilter. The 'boxing-out' design described in the 1987 Annual Progress Report was modified to fit non-standard ESPs. At the same time, the opportunity was taken to simplify the design to ease manufacture and operation of the modified ESPs.

The measurements from an earlier collaborative experiment with CEN Saclay were used as a measure of the amount of aerosols reaching the filter. The results are presented here.

Progress and Results

1. Construction of a filtering rig and detailed study of various filtration systems (B.3)

A 400 watt carbon dioxide laser has been installed in the laboratory cutting facility at the request of British Nuclear Fuels plc. The 1987 Annual Progress Report detailed the results of experiments on the ESP. The ESP was then disconnected and an HGMS prefilter installed (see Figure 1). The HGMS operated at 520 amps to generate a magnetic field of 0.3 Tesla. The cutting fume was sampled upstream and downstream of the HGMS using filter paper to measure the efficiency (by mass). The results of the tests carried out on the HGMS are kindly supplied by British Nuclear Fuels plc and are shown in Table I.

Tests 1-5 indicated some problem with the HGMS resulting in a low efficiency. Tests were carried out on fume generated by cutting mild steel plate for comparison. Tests 6 and 7 showed that the efficiency was still low. To try to correct the problem the steel wool used to capture the particles of fume was cleaned. It was hoped that this would remove the particles of dioctylphthalate (DOP) present on the steel wool during commissioning trials. The steel wool was replaced and further tests carried out. The results of Tests 8 and 9 show an improvement although they are still below the expected value of approximately 90% efficiency.

The work was terminated shortly after this following a decision not to use the HGMS as a prefilter. The results of the tests are being analysed to identify possible reasons for the unit's poor performance.

2. Assessment of various tested filter systems for their appropriateness in decommissioning applications with active aerosols (B.4)

Earlier tests on prefilters indicated that the ESP was suitable for further evaluation /1/. Work was undertaken to modify a standard ESP to ensure the safe removal of the active dust and contaminated plates by 'boxing-out' the components (see Figure 2). A preliminary design was produced and was described in the 1987 Annual Progress Report. However, it was later decided to construct the ESPs that were specified for the Windscale Advanced Gas-cooled Reactor (WAGR) decommissioning project using non-standard gas-tight construction. Because of this the 'boxing-out' design proposal was modified to fit the WAGR ESPs. At the same time, the opportunity was taken to simplify the design to ease manufacture and operation of the modified ESPs. The shutter system was made simpler and is bolted onto the ESP side. A tight-fitting seal is made by clamping the cover plate to the shutter. Also, the ESP cells are contained within a light cage to enable them to be withdrawn or replaced and remain in the correct alignment.

At present, the receptacle box has not been designed in detail. The designs have concentrated on the modifications to the ESP to enable them to be completed for commissioning of the WAGR ventilation system. These designs have been completed. Further work to develop the design for the receptacle to hold the cells for decontamination will continue using alternative funding.

3. Full-sized ventilation trials including aerosol deposition in ductings and plate out on the decommissioning machine (B.5)

The measurements from an earlier collaborative experiment with CEN Saclay were used as a measure of the amount of aerosols reaching the filters in the HERO ventilation and filtration facility (see Figure 3) /1/. Earlier work had found that 0.3% of the material cut (kerf) formed sub-micron aerosol with 40% of the aerosol produced being below 0.5 µm. Assuming that 1% of the kerf formed as aerosol, the amount of aerosol reaching the filter in HERO was calculated and is shown in Table II.

As the table shows the fraction of aerosol generated that reaches the filters is low, typically <10%. These results have a direct bearing on the WAGR ventilation circuit and can be used for assessment of the aerosols generated for calculating the atmopsheric emission of the plant.

References

/1/ Wilson K, Bishop A, Le Garreres I, Pilot G, Vendel J. Joint Report on the French Pre-Filter Tests in the HERO Development Facility, WNPDL.

Table I: Mass efficiency of the HGMS

Test No.	Material	Efficiency %	Flow rate $m^3 sec^{-1}$
1	Stainless steel	25.0	1.2
2	Stainless steel	10.7	1.2
3	Test aborted	-	-
4	Test aborted	-	-
5	Stainless steel	20.6	1.2
6	Mild steel	10.3	1.2
7	Mild steel	29.4	1.2
8	Mild steel	35.0	1.0
9	Mild steel	35.0	1.0

Table II: Fraction of aerosol reaching filters in the HERO facility

Cutting method	Material thickness (mm)	Fraction of aerosol generated reaching filters %
Flame only - mild steel	80	2.6
Powder injection - mild steel	80	6.4
Stainless steel	25	11.2

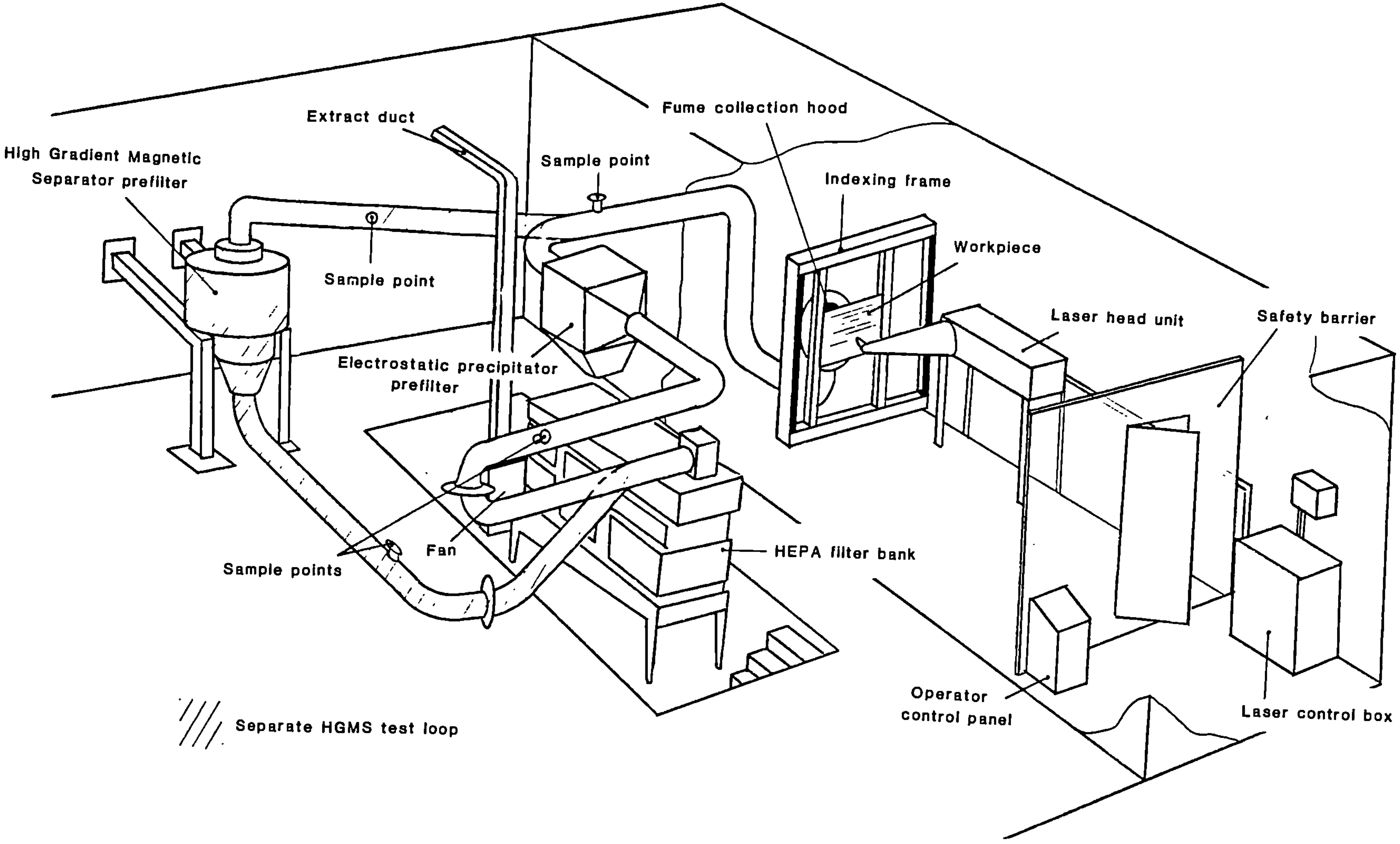

Figure 1 Laser cutting and fume generation facility

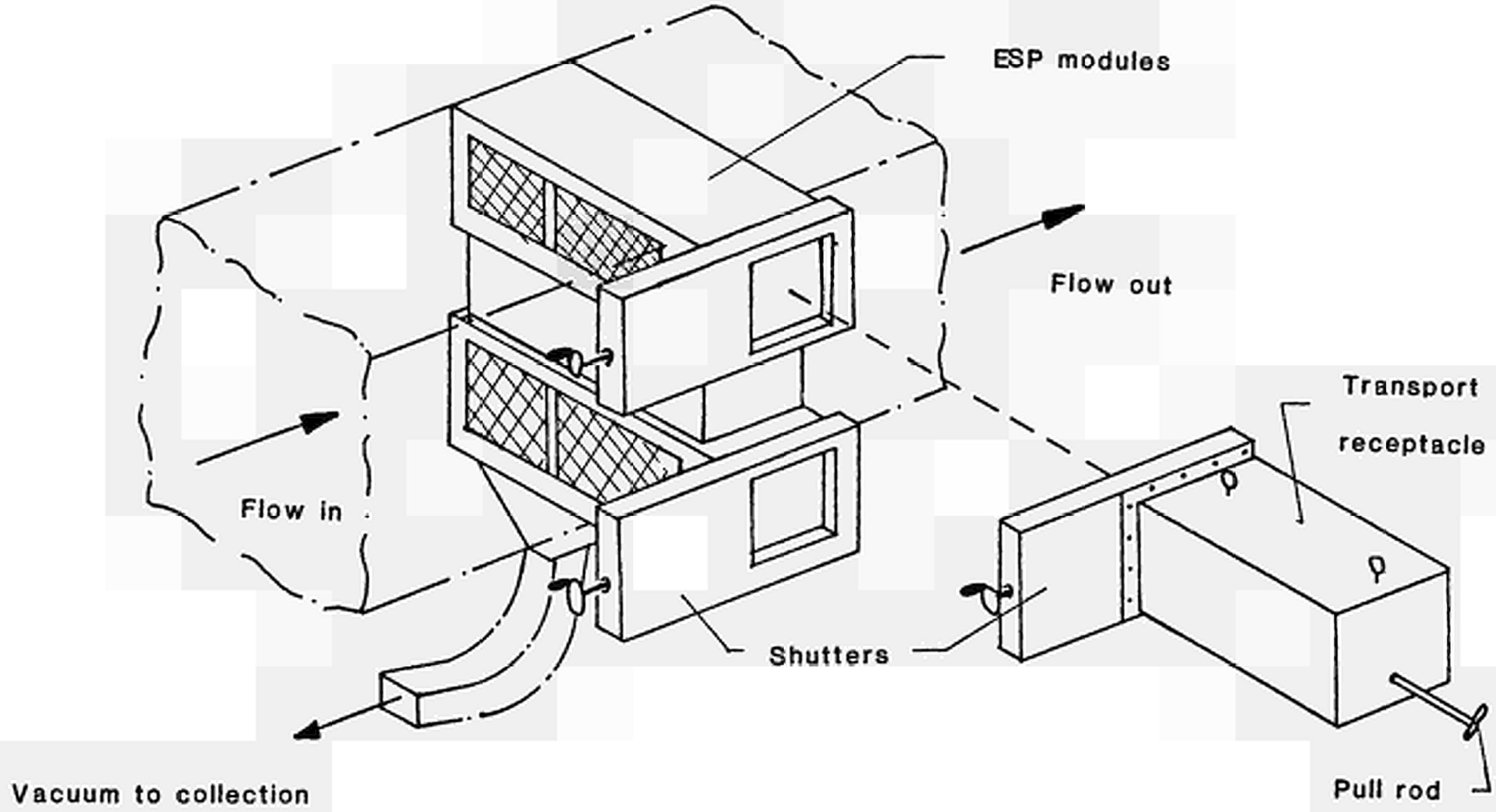

Figure 2 Schematic of an ESP modified to remove contaminated plates

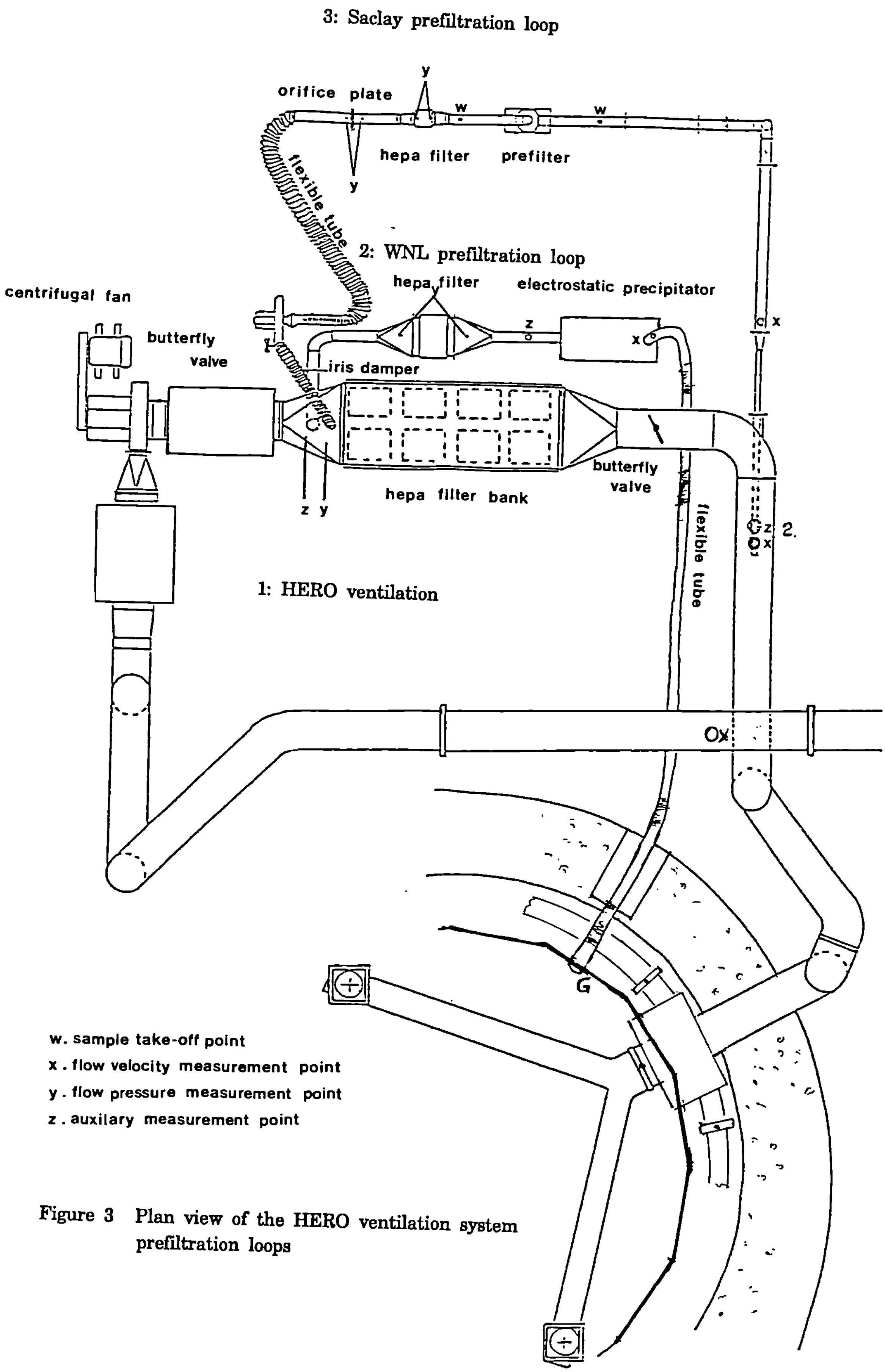

Figure 3 Plan view of the HERO ventilation system prefiltration loops

3.2. Prefiltering Devices for Gaseous Effluents from Dismantling Operations

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FI1D-0007
Working Period: January 1985 - December 1988
Project Leader: M. Pourprix

A. Objectives and Scope

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

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

This survey will be performed in co-operation with UKAEA-Windscale (see Par. 3.1.).

A supplementary agreement concluded in 1987 provides a co-operation with Heriot Watt University, Edinburgh (see Par. 3.3.) by execution of on-site aerosol measurements (B.6.).

B. Work Programme

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

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

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

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

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

B.6. Execution and evaluation of measurements on aerosol arisings from underwater plasma arc cutting tests of steel samples at Heriot Watt University, including chemical analysis of the aerosols.

C. Progress of Work and Obtained Results

Summary

This contract has been executed in cooperation with three other contractors:

- UKAEA Windscale (contract FI1D-0006),
- Heriot-Watt University of Edinburgh (contract FI1D-0008),
- CEA Cadarache (contract FI1D-0037).

All experimental work has been finished in 1987 and most of the results were presented in the previous annual progress reports (1985, 1986, 1987).

In 1988, some complementary chemical analyses have been made in order to interpret the data obtained more thoroughly and also to finalize the common reports about underwater plasma torch cutting. One of these reports deals with the influence of water depth on secondary emissions and the other with the cutting of radioactive materials coming from actual dismantling of nuclear facilities.

Progress and Results

1. Underwater plasma torch cutting

1.1. Experiments in cooperation with Heriot-Watt University of Edinburgh (B6)

Most of the experimental results have been presented in the third annual progress report (year 1987). Some complementary chemical analyses have been made in 1988 in order to study the data thoroughly. Here are the main results:

Stainless steel (304) was cut in deionised water at four different depths (0.08 m, 1.06 m, 3.55 m and 9.55 m) using Ar/N_2 plasma gas.

By the balance of solid secondary emissions, we found that:

- sedimented drosses and aerosols decrease with water depth
- attached slags and suspended particles in water increase with water depth.

Release of aerosols decreased by a factor of over 5 on increasing the water depth from 0.08 to 9.55 metres. Most of the decrease occurred in the first 3.5 metres.

The mass mean diameter, determined with an impaction/diffusion battery instrument, was around 0.2 µm and there was no obvious trend with water depth.

Furthermore, there is an enrichment in chromium and especially in manganese in aerosols due to their volatility being greater than other metal elements in the steel.

H_2 and NO were the main gaseous by-products in the effluent gas streams, reaching 20% and 1.5% volume respectively before dilution. NO concentration decreases with water depth whereas H_2 concentration was steady (after inquiries we noted this H_2 increase for 9.55 metres was due to the stronger current used).

The PH of water was significantly decreasing due probably to absorption. This acidity led to a partial dissolution of suspended particles.

1.2. Experiments in cooperation with CEA/CEN Cadarache (B5)

These experiments presented in the previous annual report (1987) have been described in detail in a common report /1/.

Several radioactive plates coming from the dismantling of two nuclear facilities have been cut by underwater plasma torch with selected captation and prefiltration devices for the aerosol secondary emissions.

With the results (indicated in the previous annual report) for the cutting of Rapsodie plates (1900 Bq/g) we have made a synthesis in order to evaluate the distribution of the activity in the different parts of the facility where the cutting took place.

Figure 1 gives the distribution of the ^{137}Cs activity, that accounts for 95% of the total activity, in the experimental facility. The released activity comes mainly from a decontamination effect by immersion and also from the kerf volume.

We can remark that:

- the decontamination factor by immersion is about 1.6,
- from the total of the lost activity (immersion and kerf), 0.8% of the activity is in the ventilation duct (aerosols), 0.3% in sedimented drosses and almost 99% in the water.

The fact that in the present case of a plate with a ^{137}Cs contamination mainly not fixed, most of the activity is going into the water necessitates a water treatment, besides filtration, if the contamination is in a soluble phase.

Figure 2 shows the volumic ^{137}Cs activity along the ventilation and filtration circuit for the cutting of a Rapsodie plate. Before the dilution point the values indicated are measured, after this they are calculated.

References

/1/ PILOT, G. (CEA/SACLAY), LEAUTIER, R. (CEA/CADARACHE)
Découpage sous eau à l'arc plasma de matériels radioactifs. Evaluation des émissions secondaires en gaz et aérosols. 15ème congrès ATSR, Grenoble 5-6-7 octobre 1988.

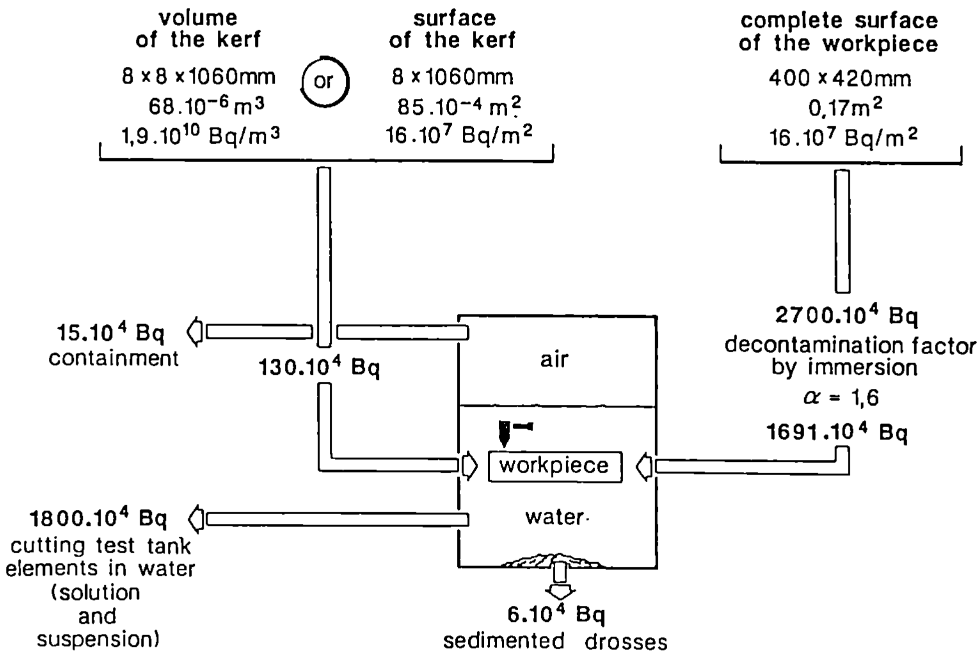

Figure 1: Cutting by underwater plasma torch of a radioactive steel plate. Global distribution of the ^{137}Cs activity.

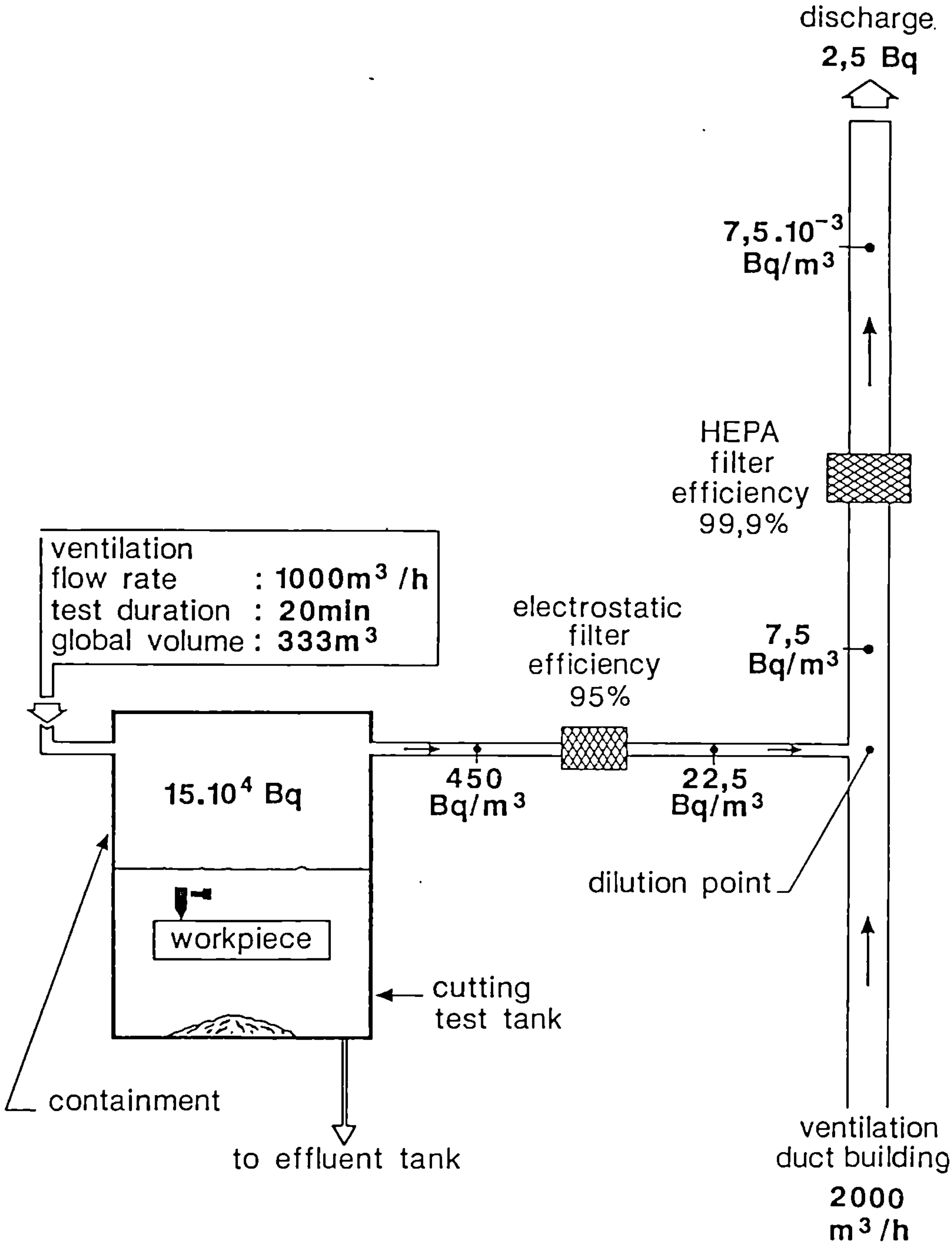

Figure 2: Cutting by underwater plasma torch of a radioactive steel plate. Volumic ^{137}Cs activity along the ventilation circuit.

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

Contractor: Heriot-Watt University, Edinburgh, United Kingdom
Contract N°: FI1D-0008
Working Period: January 1985 - December 1988
Project Leader: B. Waldie

A. Objectives and Scope

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

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

In a supplementary agreement concluded in 1988, the initial work programme was extended by the inclusion of a set of 8 supplementary standard cutting tests with plasma arc torch (underwater on stainless steel samples), undertaken together with the Commissariat à l'Energie Atomique and Universität Hannover (see ref. 3.10. and 3.11.)

B. Work Programme

B.1. Updated literature review and analysis of data on secondary waste (dross, ultrafine fume particles) generated during underwater plasma-arc cutting of steel.
B.2. Design and construction of a dross collection system, appropriate for underwater cutting.
B.3. Design and construction of a collection device for ultrafine fume particles appropriate to underwater cutting.
B.4. Development of TV and/or photographic techniques for underwater monitoring of the behaviour of cutting waste.
B.5. Tests on cutting of non-active stainless steel samples in hyperbaric flooded test chambers, with monitoring of dross and ultrafine fume characteristics under various cutting parameters (cutting vertically upwards).
B.6. Idem B.5., with cutting vertically downwards.
B.7. Idem B.5., with cutting in horizontal position.
B.8. Design and construction of a test vessel providing a 10m water depth and monitoring/sampling devices for cutting waste.
B.9. Cutting tests in the facility developed in B.8. with cutting parameters selected in B.5. to B.7.
B.10 Analysis of the surface layer material behaviour by trace and compound work-piece techniques.
B.11 Conclusive assessment of obtained results.
B.12 Execution of supplementary underwater cutting tests with working parameters common to other contractors, with subsequent comparative analysis of results.

C. Progress of Work and Obtained Results

Summary

1. The joint study with the SPIN Group of CEA Saclay on the effect of water depth on secondary emissions has been completed and the final report submitted. New data are reported on the distribution of solid emissions, compositions of these emissions including enhancement of aerosol particles by more volatile species in the workpiece, aerosol size distributions, gas evolution and changes in water quality.

2. Experimental work on the original programme has been completed and a final report is in preparation.

3. Further data on filtration rate characteristics of fine water suspended particles have been obtained. An overall correlation of rate in terms of concentration has been produced.

4. A series of standard underwater plasma cuts and associated emission measurements has been done as part of an interlaboratory comparison with CEA Saclay, University of Hannover and CEA/CEN Cadarache.

Progress and Results

1. Joint Study in Characterization of Secondary Emissions from Cutting of Stainless Steel with Underwater Plasma (B.9, B.11)

Preliminary results from this joint study with the SPIN Research Group of CEA Saclay to establish the effects of water depth on secondary emissions were reported in the last annual report /1/. The project has been completed and the final joint report submitted./2/. The main findings were summarised as follows:-

Stainless steel, grade 304 of thickness 37 mm, was cut in deionised water at depths up to 9.55 metres using Ar/N_2 plasma gas. Yields of attached slag, sedimented dross, fine suspended particles and aerosol particles were measured for four water depths. Release of aerosol particles decreased by a factor of over 5 on increasing the water depth from 0.08 to 9.55 metres. Most of the decrease occurred in the first 3.5 metres.

Size distribution measurements of aerosol particles, done mainly with an impaction/diffusion battery instrument, indicated a mass mean size around 0.2 microns, independent of depth.

Detailed metal element analysis of fine particles suspended in the water and of aerosol particles showed that in both cases the mass of particles approached twice their total metal content. Oxidation or other reactions may partially account for this increase. Aerosols were substantially enriched in manganese due to its volatility being greater than other metal elements in the steel. There was no obvious variation in composition with depth or particle size.

H_2 and NO were the main gaseous by-products in the effluent gas streams, reaching 20% and 1.5% volume respectively before dilution. NO concentration decreases with depth, presumably due to oxidation and absorption. Acids resulting from absorption would explain the significant decrease in the pH of the water. There is some evidence that suspended particles could be dissolved with time, presumably due to the acidity.

Data on mass loss from the stainless steel workpieces for four water depths and the distributions of collected solid emissions are summarised in Table I. The pronounced enhancement of manganese concentration in the aerosol particles is quantified in Table II. Concentrations are an order of magnitude higher than in the 304 stainless steel workpiece. This effect is attributed to the significant higher volatility of manganese compared to iron or nickel. This finding complements results from earlier work in which the faces of plasma cut workpieces were found to be deficient in manganese. The apparent enhancement in copper (Table II) cannot be positively attributed to the same effect due to the possibility of copper loss from the plasma torch nozzle. The extent of enhancement of other more critical species would be worth examining in future studies.

Similar to the fine water suspended particles /1/, the total mass of aerosol particles is higher by a factor of about two than their elemental metal content (Table III). Again this suggests formation of oxides or other compounds from the initial metal vapours or particles.

Yields of hydrogen gas resulting in concentrations up to 20% in effluent gas before dilution were not inconsistent with previous yields /4/ in shallow water when compared on a basis of electrical power input. Water depth therefore only affects hydrogen evolution rate if the current and/or voltage increase with depth.

2. Further Cutting Experiments (B.5, B.9, B.10)

Cutting studies under the original main contract have been completed. Vertical and horizontal (strictly the conventional gravity position) cutting in the deep water vessel have been compared and a study made of the distribution from a tracer layer mounted on the front of the workpiece.

Cutting with the workpiece horizontal produced somewhat more suspended particles at both depths, 0.08 and 3.55 m (Table IV). Yield of aerosol particles however was less for the horizontal orientation in shallow water (0.08 m). Presumably this is due to better contact between gas bubbles, aerosol particles and water in the region beneath the workpiece in the horizontal orientation. In deep water (3.55 m) contacting in the additional water column reduces the aerosol yield significantly and essentially equalizes the yields from horizontal and vertical orientations. As in earlier runs /1/, total solids collected exceeded the mass loss of the workpiece, presumably due to oxidation and other reactions.

To help assess the destination of surface layers in a workpiece, tests were done on stainless steel with a 0.6 mm thick layer of silver tracer in the front surface. Silver was chosen to ease subsequent chemical analysis. Analysis of fine particles suspended in the water and aerosol particles showed much higher concentrations of silver in the aerosol. This distribution would be promoted by the relatively high volatility of silver - about the same as manganese.

3. Filtration Characteristics of Particles Suspended in Water (B.2, B.9)

Additional data which extends the previous concentration range /1/ has been obtained on filtration rates of dilute suspensions of fine particles remaining in the water after a cut. These come from cuts in the deep water vessel with workpieces on the horizontal and vertical orientations. No systematic trend with cut conditions is evident. The main factor influencing the filter cake resistance parameter is the initial particle concentration, see results on fig. 1.

4. Standard Cutting Tests and Secondary Emissions (B.12)

Under a supplementary agreement a series of standard cutting runs and associated emission measurements have been made as part of a joint programme with the Institut für Werkstoffkunde, Universität Hannover, SPIN group of CEA Saclay and CEA/CEN Cadarache. Aerosol and gas emission measurements and chemical analysis of particles were done by SPIN/CEA. The overall aim is to compare results on emissions from the three laboratories and establish reasons for any significant differences. Underwater plasma cuts were made in 20 and 40 mm thick grade 304 stainless steel, workpieces being supplied by Universität Hannover from a common source. To achieve a longer cut length, 0.9 m, the traversing system in the base of the deep water test vessel was modified to give motion in X and Y directions. Cutting speed and gas flow rate (60% A, 40% N) were standardized at 60 mm/min and 50 l/min respectively. Cuts were done in 0.5 and 1.0 m of deionized water. A total of eight cuts was carried out in this laboratory and good reproducibility achieved.

Results are currently being assessed and a joint report is in preparation.

References

/1/ The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Second Annual Progress Report (year 1986), EUR 11112, pp. 82-87.

/2/ WALDIE, B., PILOT, G., LOYER, H. and HARRIS, W.K. Heriot-Watt Report C.P.E. PT/01/B, 1988.

/3/ ANTOINE, P., Le GARRERES, J., PILOT, G. and POURPRIX, M. Proc. Spectrum '86 Niagara falls (N.Y.) Sept. 14-18, 1986.

/4/ RYAN, E.J. AND WALDIE, B. Proc. 5th Int. Symp. Plasma Chemistry, 1, 242, 1981.

TABLE I: WORKPIECE MASS LOSS AND DISTRIBUTION OF COLLECTED SOLID EMISSIONS

Water depth m.	Workpiece Mass loss: As cut $g.m^{-1}$	Workpiece Mass loss: Attached slag removed $g.m^{-1}$	Attached slag (1) $g.m^{-1}$	Attached slag (1) %*	Sedimented dross (2) $g.m^{-1}$	Sedimented dross (2) %	Suspended particles (3) $g.m^{-1}$	Suspended particles (3) %	Aerosol (4) particles $g.m^{-1}$	Aerosol (4) particles %	Total collected $g.m^{-1}$
0.08	1,559	1,606	47	2.81	1,614	96.53	10.6	0.634	0.413	0.025	1,672
1.06	1,512	1,561	49	3.07	1,537	96.24	11.0	0.689	0.269	0.017	1,597
3.55	1,451	1,601	150	8.91	1,519	90.20	14.7	0.873	0.180	0.011	1,684
9.55	1,684	1,804	120	6.01	1,850	92.64	27.2	1.36	0.079	0.004	1,997

* As percentage of total (1 + 2 + 3 + 4) collected

TABLE II: RATIOS OF ELEMENTAL METAL CONTENTS OF AEROSOL PARTICLES ON FILTERS

	Filter diameter mm	Air Cut	Water cut depth, m: 0.08	1.06	3.55	9.55	304 stainless steel
Cr/Fe	130	0.307	0.247	0.354	0.368	0.299	0.252
	47	0.319	0.358	0.383	0.350	0.250	
Ni/Fe	130	0.085	0.107	0.170	0.159	0.148	0.125
	47	0.085	0.126	0.133	0.115	0.092	
Mn/Fe		0.365	0.287	0.338	0.263	0.219	0.025
Cu/Fe		0.029	0.189	0.127	0.080	0.082	0.006

TABLE III: COMPARISON OF TOTAL MASS AND ELEMENTAL METAL CONTENT OF AEROSOL FILTER DEPOSIT

Cut water depth, m	130 mm ϕ filter Total Mass / Fe + Cr + Ni + Mn
(Air)	2.21
0.08	1.73
1.06	1.97
3.55	1.96
4.55	1.86
Mean of underwater cuts	1.88

TABLE IV: COMPARISON OF CUTTING WITH WORKPIECES HORIZONTAL AND VERTICAL AT WATER DEPTHS OF 0.08 AND 3.55 m

Water Depth m	Workpiece Orientation	Material Thickness mm	Mass loss of Workpiece		Attached slag		Sedimented dross		Suspended particles		Aerosol particles		Total collected g/m
			As Cut	Attached Slag Removed	g/m	%	g/m	%	g/m	%	g/m	%	
0.08	Vertical	37	1523	1568	45	2.8	1529	96.6	7.876	0.498	0.464	2.93×10^{-2}	1582
0.08	Horizontal	37	1559	1606	47	2.81	1614	96.53	10.6	0.634	0.413	2.5×10^{-2}	1672
3.55	Vertical	37	1623	1663	38	2.1	1726	97.3	10.045	0.566	0.196	1.11×10^{-2}	1774
3.55	Horizontal	37	1451	1601	150	8.91	1519	90.20	14.7	0.873	0.180	1.1×10^{-2}	1684

Plasma Gas rate 47 l/min (60A/40N)
Cut Speed 46 mm/min ± 10%
Workpiece 304 ss, 37 mm thickness

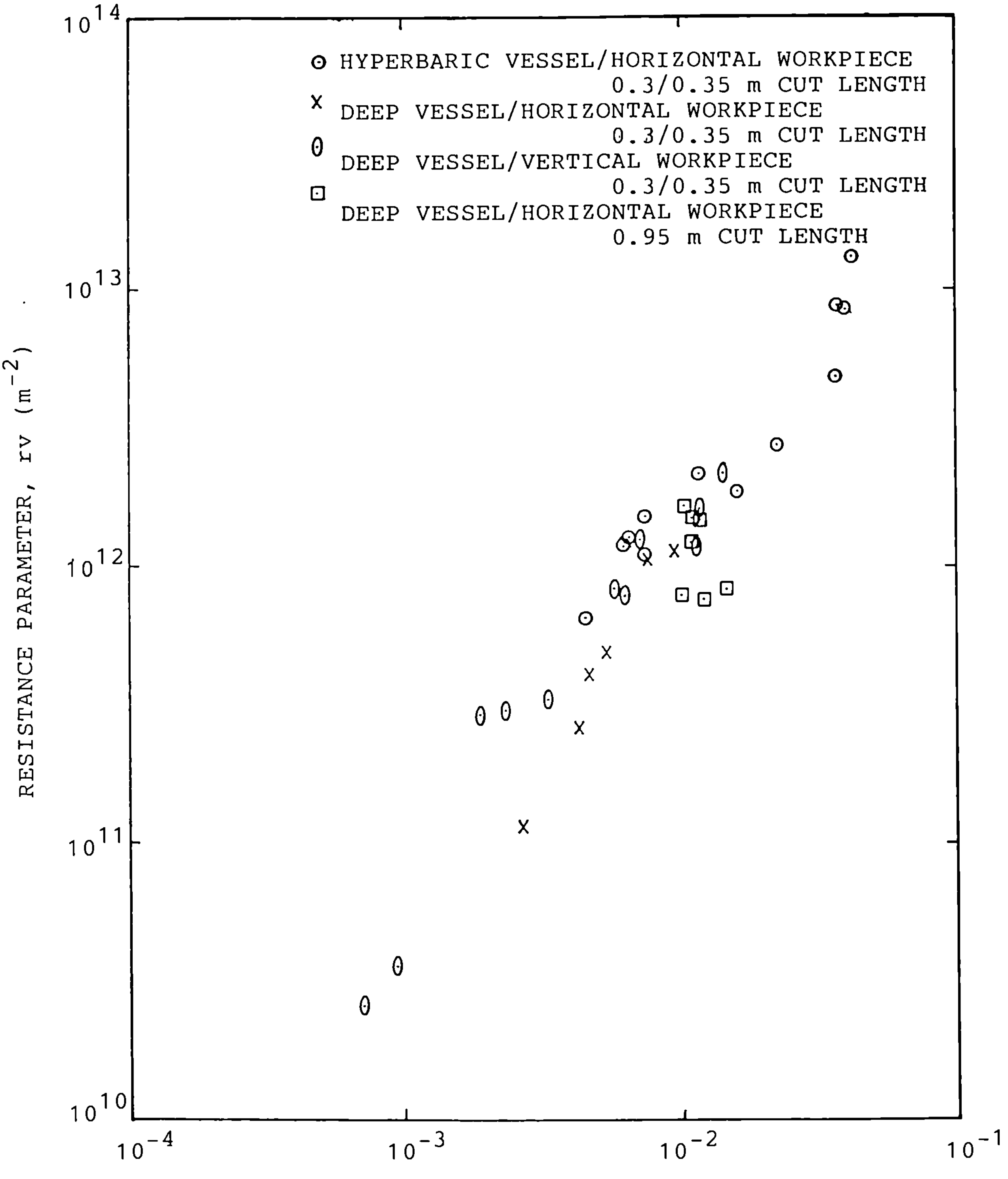

Figure 1. Filtration Resistance of Fine Suspended Particles in Water

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

Contractor: Field Automation, Paris, France
Contract N°: FI1D-0009
Working Period: January 1985 - September 1987
Project Leader: P. Thomé

A. Objectives and Scope

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

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

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

Through a supplementary agreement concluded in 1987, the work programme has been extended to the development and testing of an internal cutting tool by axial action (B.3.).

B. Work Programme

B.1. Development of methods and tools for internal arc-saw cutting of steam-generator tubes (internal diameter of about 19 mm) to produce circumferential cuts.

B.1.1. Design, construction and testing of a miniaturised cutting tool for PWR steam generator channel head.

B.1.2. Development and construction of a laboratory testing bench to check process characteristics by external and internal cutting.

B.1.3. Design and fabrication of a complete device for internal cutting, based on test results obtained under B.1.1. and complying with in-situ working limitations.

B.1.4. Performance tests on representative Inconel tubes (under water).

B.1.5. Final assessment on realistic full-scale samples supplied by EdF.

B.2. Development of methods and tools for internal arc-saw cutting of pipes (internal diameter about 200 mm) to produce circumferential cuts.

B.2.1. Design of a suitable crawler for hoisting the cutting tool.

B.2.2. Fabrication of an appropriate cutting tool and laboratory tests to determine the cutting parameters in accordance with the pipe diameter and thickness and with the tool working limitations.

B.2.3. Testing of the cutting tool.

B.3. Development of methods and tools for internal arc-saw cutting of steam generator tubes to cut along a generating line.

B.3.1. Design and construction of a miniaturised tool for internal axial cutting.

B.3.2. Performance tests on Inconel and stainless steel tubes.

C. Progress of Work and Obtained Results

Work has been completed, the final report is under publication.

3.5. Electrochemical Technique for the Segmenting of Activated Steel Components

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen, Germany
Contract N°: FI1D-0010
Working Period: January 1985 - June 1986
Project Leader: W. Stang

A. Objectives and Scope

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

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

In the research programme it will be established:

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

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

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

B. Work Programme

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

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

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

C. Progress of Work and Obtained Results

The work has been completed, the final report is under publication.

3.6. Explosive Techniques for Dismantling of Biological Shield Structures

Contractor: Battelle-Institut e.V., Frankfurt, Germany
Contract N°: FI1D-0011
Working Period: April 1985 - December 1988
Project Leader: H.U. Freund

A. Objectives and Scope

In the decommissioning of reactor systems, the removal of heavy reinforced or prestressed concrete structures, in which large quantities of concrete and steel have become activated during reactor operation, is considered as a major problem. For the safe removal of activated concrete without release of radioactive material, various techniques are being considered, in which a high level of control could be imposed. In the foregoing EC research programme, Taylor Woodrow Construction (TWC), investigated controlled explosive demolition techniques. During the same period, Battelle-Institut e.V., Frankfurt (BF) demonstrated the feasibility of an approach using "line charges" as opposed to the "point charges" used by TWC.

The present research work aims at complementing, improving and optimising the foregoing work. Extensive investigations will be executed on the adjustment of blasting parameters, material and structural effects, drilling techniques, particle distribution and on procedures for remote handling. Work is carried out jointly - based on a common and complementary work programme - with TWC (contract N° FI1D-0012).

In a supplementary agreement concluded in 1988, additional blasting tests were added to the work programme of BF (under item B.1.1.), in order to reinforce the assessment of simultaneous blasting.

B. Work Programme

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

C. Progress of Work and Obtained Results

Summary

Previous investigations with very small explosive charges which are reported in the preceding annual report /1/ had indicated an increase in concrete removal efficiency when a group of bore hole charges detonates with high precision simultaneously rather than sequentially. This effect has been confirmed with charges of optimum mass. A factor of two increase is found for the concrete loosened or removed from the residual structure. In addition a smoother removal surface is obtained. The improvement is present both in low and heavy reinforcement. The local damage to the residual structure, however, is also increased. The results are considered to bear significant practical impact, since precision timing has so far not been introduced in common blasting techniques.

During explosive concrete dismantling dust representing a mass fraction of about 10^{-4} is set free. The amount of dust can be removed substantially by a water spray which is explosively generated simultaneously with the concrete blasting. The spray method consists of a water filled balloon which is placed close to the blasted surface in order to cause efficient dust wash out. By the spray method also part of the toxic explosive reaction products is removed.

Progress and Results

1. Initiation Mode (B.1.1)

1.1 Introduction

It had been found earlier that the concrete removing efficiency is higher for simultaneous as compared to sequential blasting. While this effect had been verified for very low explosive charge mass (= threshold value for cratering/no cratering) the investigations which underly this report were performed to prove if this effect would also be present with charges of optimum mass and in the case of heavily reinforced concrete.

1.2 Experiments

Test series were prepared to compare sequential and exact simultaneous blasting with optimum mass charges and also to compare these results with previous ones obtained with minimum mass charges. Four test bodies of 1,40 m total diameter were produced. They simulate 1:1 scale sections of real biological shield structures with low and heavy lateral reinforcement. A bore hole array consisting of 7 holes in a hexagonal pattern was drilled into each test body to a depth of 20 cm. The charge mass per bore hole in all cases was 22,5 g of PETN high explosive. This mass value has been adopted as best value from single bore hole experiments. For exact simultaneous firing the charges were initiated by a ramified set of pyrotechnic cords (NONEL)[1] which were trig-

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1) The NONEL cord /2/ shows a precision detonation velocity of 2000 m/s and is thus able to provide for exact simultaneous detonation of multiple charges.

gered by an electric detonator. The sequential firing was performed by firing pairs of charges with a delay between the two detonating charges of > 6 ms. Pairs of charges were fired within 2 hours time or longer. This way overlap of the stress fields generated by the individual charges was avoided.

1.3 Calculations

Using a linear elastic model for spherical wave expansion into isotropic concrete simplified calculations were made for the vertical and lateral stresses in the plane of separation. Even though the concrete near the charges does not behave linearly elastic coarse quantitative results for the stress fields in the two firing modes can be obtained.

1.4 Results

The main results are summarized as follows:

- Exact simultaneous blasting of an array of bore-hole charges results in more efficient concrete removal than sequential blasting. This can be seen qualitatively for minor reinforcement by comparing Figures 1a and 1b: With exact simultaneous blasting the concrete layer together with the lateral reinforcement is removed, whereas, with sequential blasting the reinforcement stays in place and less concrete is removed.
- The improvement - a factor of two larger concrete removal- is present both in cases of minor reinforcement and of heavy reinforcement, see Figure 2.
- A better surface smootheness is achieved at the newly generated removal surface in simultaneous blasting than in sequential blasting.
- The damage to the residual concrete as measured by crack density is also higher in simultaneous blasting than in sequential blasting, see Figure 2.
- The firing circuit consisting of a multiply ramified set of NONEL cords can be successfully used for exact simultaneous blasting.
- Peak stress values may be higher by a factor of 3 in simultaneous firing as compared to sequential firing. This fact is regarded as responsible for the increased concrete removal efficiency, based on the stress threshold concept for concrete fracturing.
- Lateral steel reinforcement absorbs and reflects part of the stress waves weakening the stress field near the surface and enhancing the lateral stress extension, thus extending the lateral fracturing of concrete.

2. Effect of spraying (B.5.2)

2.1 Explosive water spray

A simple explosive water spraying setup has been tested. The aim is to remove concrete dust from the aerosol state immediately after its release during bore hole blasting. The setup is to be positioned close to the blasted surface where the washout efficiency is expected to be high due to the high concentration both of dust particles and water droplets. The setup is to be positioned close to the blasted surface where the washout efficiency is expected to be high due to the high concentration both of dust particles and water droplets. The characteristics of water sprays which are generated from an exploding water filled balloon of 1 l volume with a PETN type high explosive charge placed in the centre were measured. The mass of the charge was between 1,8 g and 10,2 g.

The setup was tested in concrete blasting experiments performed at TWC. Results will be described by TWC.

2.2 Results without concrete blasting

Blasting of the setup rapidly generates a spherical cloud of water mist. The following characteristics were obtained:

- A cloud radius of 1 m is reached in times between 40 ms and 15 ms for charge masses of 1,8 g and 10,2 g, respectively. Expansions beyond about 1,4.m to 1,6.m occur relatively slowly, i.e. for times above 100 ms.

- Sedimenting droplet sizes of about 0,2 to 5 mm were observed. The low explosive mass of 1,8 g gives incomplete droplet formation. For higher explosive masses all the water is transformed into a droplet spray.

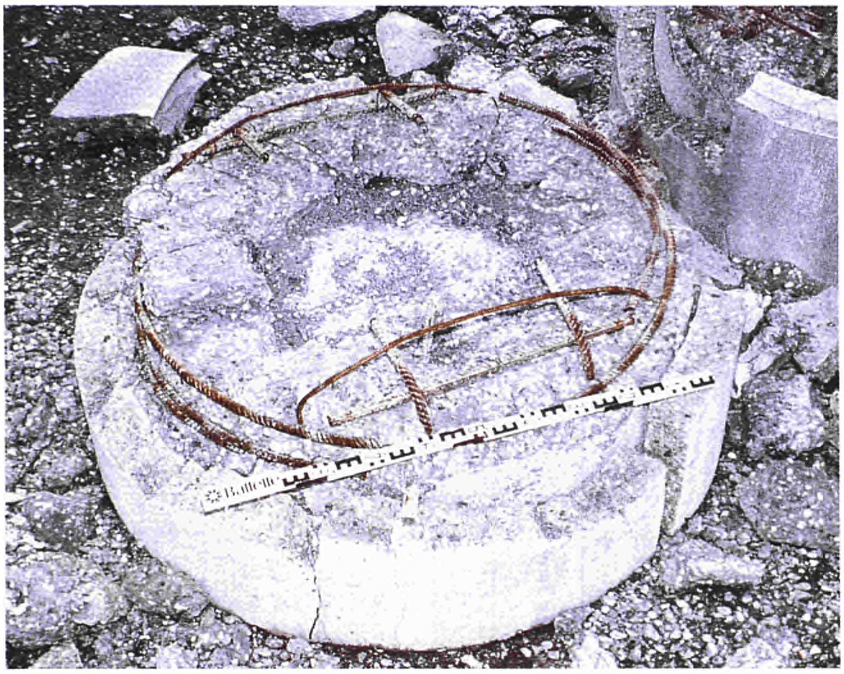
It is recommended to use a specific loading of at least 2,5 g of explosive per liter of water.

References

/1/ The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Third Annual Progress Report (year 1987), EUR 11715, 1988

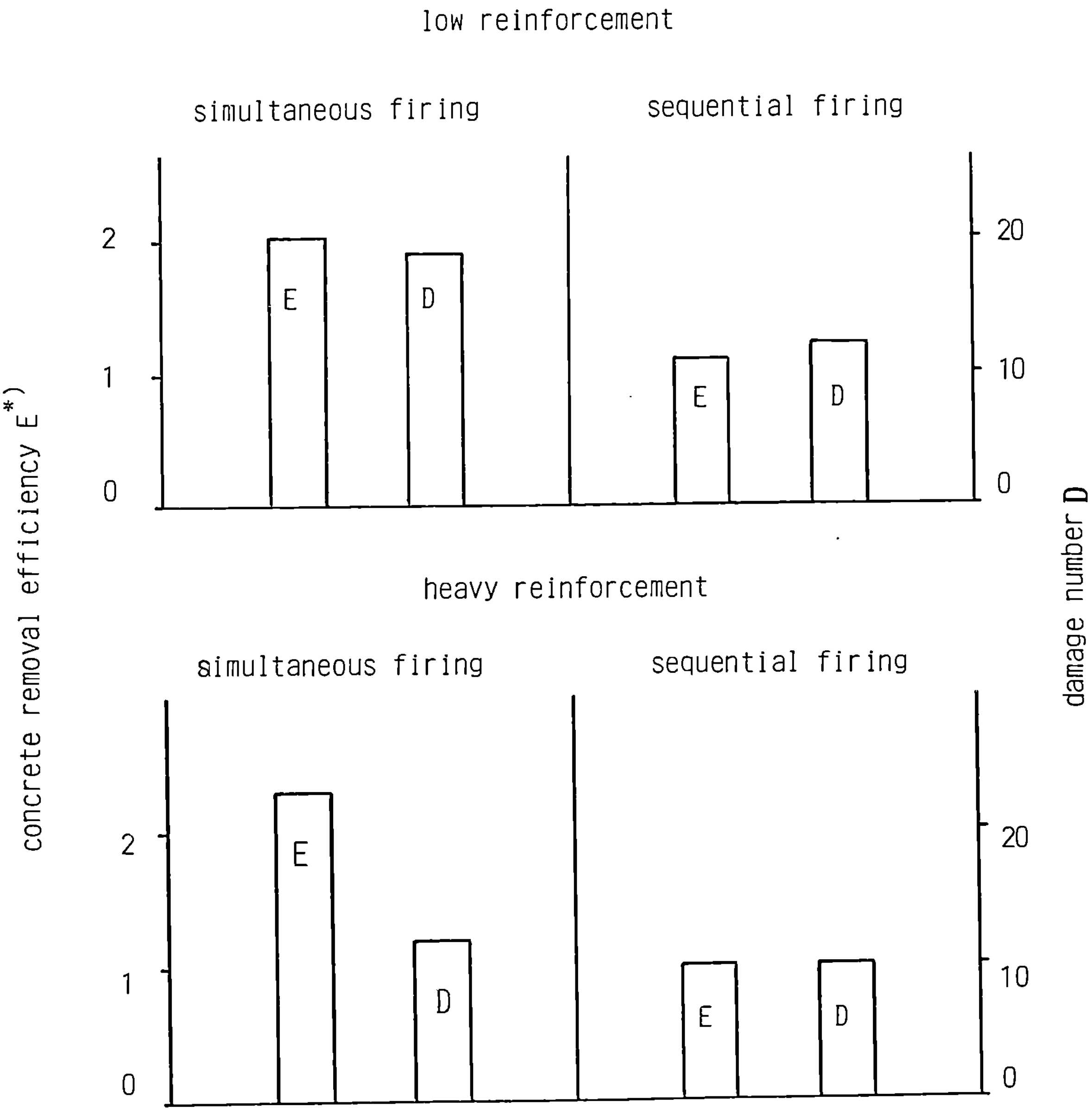
/2/ Performance and application description by the manufacturer Nitro Nobel, Gyttorp, Sweden

a)



b)

Figure 1: Comparison of concrete removal on test bodies of 1,4 m diameter
a) exact simultaneous blasting
b) sequential blasting



*) mass of concrete separated from the test body in kg versus mass of explosive in g

Figure 2 : Concrete removal efficiency and relative damage to the residual test body

3.7. Explosive Techniques for Dismantling of Activated Concrete Structures

Contractor: Taylor Woodrow Construction Ltd, Southall, United Kingdom
Contract N°: FI1D-0012
Working Period: January 1986 - December 1988
Project Leader: C.C. Fleischer

A. Objectives and Scope

In the decommissioning of reactor systems, the removal of heavy reinforced or prestressed concrete structures, in which large quantities of concrete and steel have become activated during reactor operation, is considered as a major problem. For the safe removal of activated concrete without release of radioactive material, various techniques are being considered, in which a high level of control could be imposed. In the foregoing EC research programme, Taylor Woodrow Construction (TWC) investigated controlled explosive demolition techniques. During the same period, Battelle-Institut e.V., Frankfurt (BF) demonstrated the feasibility of an approach using "line charges" as opposed to the "point charges" used by TWC.

The present research work aims at complementing, improving and optimising the foregoing work. Extensive investigations will be executed on the adjustment of blasting parameters, material and structural effects, drilling techniques, particle distribution and on procedures for remote handling. Work is carried out jointly - based on a common and complementary work programme - with BF (contract N° FI1D-C011).

In a supplementary agreement concluded in 1988, additional blasting tests were added to the work programme of BF (under item B.1.1.), in order to reinforce the assessment of simultaneous blasting.

B. Work Programme

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

C. Progress of Work and Obtained Results

Summary

Work carried out in 1988 has concentrated on work packages B2. "Material and Structural Effects", B3 "Drilling and Boring of Charging Holes", B4 "Study of the Structural Response of the Test Body", B5 "Assessment of Particle Size Distribution of Generated Dust" and B6 "Final Assessment and Evaluation of Results, including desk studies on procedures for remote handling".

Further tests have been carried out investigate the removal of material from re-entrant corners of curved surfaces with larger radii of curvature. These tests have confirmed earlier findings on the positioning of explosive charges to achieve a successful removal of material from re-entrant corners. Theoretical modifications have been carried out on a shaped charge design to produce a charge capable of boring a hole having specific required characteristics and the designed shaped charge has been tested in field trials. Work has also been carried out on mathematical modelling of the effect of firing a buried explosive charge in a concrete mass.

Particle size distribution analysis have been carried out on the dust generated during blasting. It has been shown that it is feasible to reduce the total dust particle count in the airspace by the use of fine water sprays and less efficiently through wetting of the surface before blasting. Preliminary work has been started on the development of a concept for a remote handling system for the explosive cutting technique.

Progress and Results

1. Effect of the presence of a liner (B.2.1)

Tests reported on in the preceding annual progress report /1/ had demonstrated the feasibility of debonding the liner and its anchoring system from the concrete and to break-up a layer of the concrete adjacent to the liner. However the presence of the liner made post - test examination of the extent of structural disruption beneath the liner difficult. It was therefore decided to cut the model to facilitate a closer inspection of the break-up beneath the liner.

Inspection of the sectioned model indicated that considerable break-up had occurred beneath the liner (see Fig 1). The liner had debonded from the concrete in some areas directly over the explosive charge positions. The reinforcement and concrete cover together with the anchor studs and attached concrete above the explosive charge locations has also been sheared-off from the main body of the model. The results clearly indicated the feasibility of using explosives to debond the liner and its anchoring system from the concrete.

2. Effect of Geometrical Shape of Structure (B.2.1)

Initial tests in this development programme /1/ demonstrated the feasibility of cratering and concrete removal from re-entrant corners on a model with a small radius of curvature. The test results were however found to be affected by the small radius of curvature. Further tests have been carried out on a model having a larger radius of curvature ie a test block having a diameter similar to modelled versions of an existing biological shield. These tests have confirmed earlier findings that material could be removed successfully from re-entrant corners by the correct positioning of explosive charges.

3. Drilling and Boring of Charge Holes (B.3.1)

In the preceding programme of work /2/, it was demonstrated that shaped charges may be used to produce boreholes suitable for placing cratering charges. However the dimensions of the hole produced did not meet current borehole requirements for cratering charges. Theoretical work has been carried out to modify the shape charge design to establish a charge capable of producing a hole of the required dimensions. Shape charges have been manufactured based on the modified design and these have been used in field trials. The test results indicated a significant dependency of the formed borehole characteristics on the concrete material properties, resulting in the test borehole characteristics being smaller than what had been predicted for a weaker concrete structure.

4. Structural Response of Test Body (B.4.4)

A concrete model has been loaded with an assessed transient and the response of the structure analysed using a finite element formulation (ADINA) and a finite difference formulation (DYNA3D). Both analyses assumed a two dimensional axisymmetric simulation with a small layer of mesh elements surrounding the charge represented by "compaction" material models. The results from both analyses showed the formation of a crater having characteristics similar to the crater formed on the physical model during field trials. It is anticipated that with better definition of concrete properties for the high rates of loading experienced under blast loading the agreement between predicted and model test results could have been even better.

5. Assessment of Particle Size Distribution of Generated Dust During Blasting (B.5)

Tests have been carried out to investigate the particle size distribution on the dust generated from a simulated cutting process involving the use of explosives on reinforced concrete targets. The test block was enclosed in a PVC tent to help contain the generated dust and the entrained particulate was removed using suitable fans and ducting. Standardized experimental conditions were chosen which allowed representative measurement of both the total amount of dust below a given grain size and the particle size distribution to be assessed (Fig 2).

The test results indicated that small quantities of submicron particles was found in the airspace after the explosives were fired. Further investigation into the source of the submicron particles would tend to suggest that these may have originated mostly form the explosive used and not from the concrete material. Tests were also carried out to investigate the effectiveness of dust suppression techniques. The use of an explosive water bag to produce a fine water spray mist at the exact time when the dust is generated has been shown to produce significant reductions in the total count of airborne particles in the airspace. Wetting the surface before firing was also shown to produce a reduction in the total amount of airborne particles in the airspace.

6. Remote Handling (B.6.)

In order to reduce radiation exposure and occupational hazards during decommissioning tasks in nuclear facilities and to allow intervention at an early stage after shutdown remote handling systems will be needed for the biological shield dismantling operation. Figure 3 shows an example concept sketch of the type of manipulator that may be required for the explosive cutting technique. Basic desk top studies are currently being carried out to define the likely requirements such a manipulator will need.

References

/1/ The Community's research and development programme on decommissioning of nuclear installations.- Third annual progress report, EUR 11715 EN (1988).

/2/ FLEISCHER C C, Taylor Woodrow Construction Ltd, A Study of Explosive Demolition Techniques for Heavy Reinforced and Prestressed Concrete Structures, EUR 9862 EN (1985).

Fig.1a. Example of material break-up beneath liner

Fig. 1b. Example of material break-up beneath liner

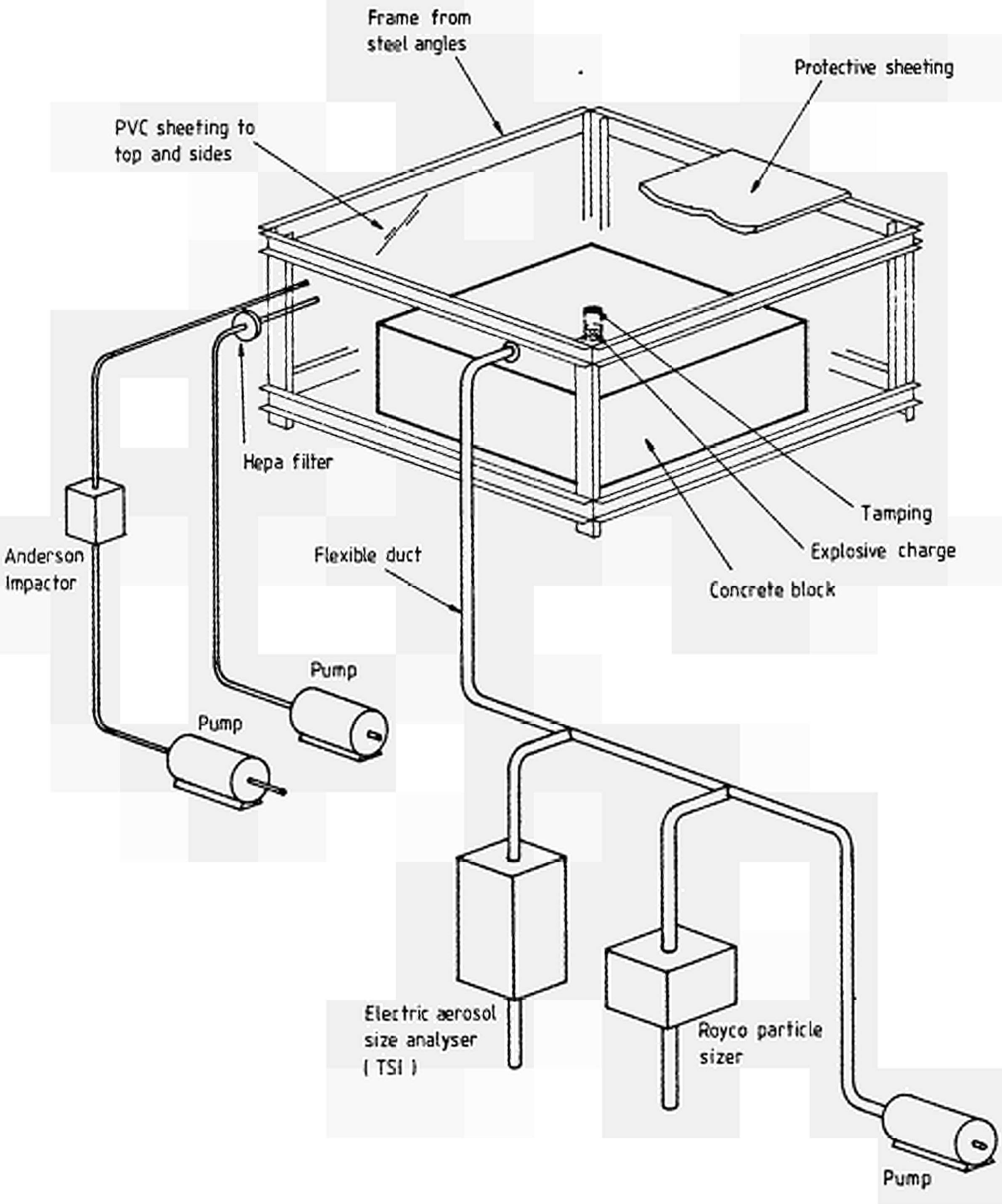


Fig. 2. Sketch of test set-up for particle size distribution analysis.

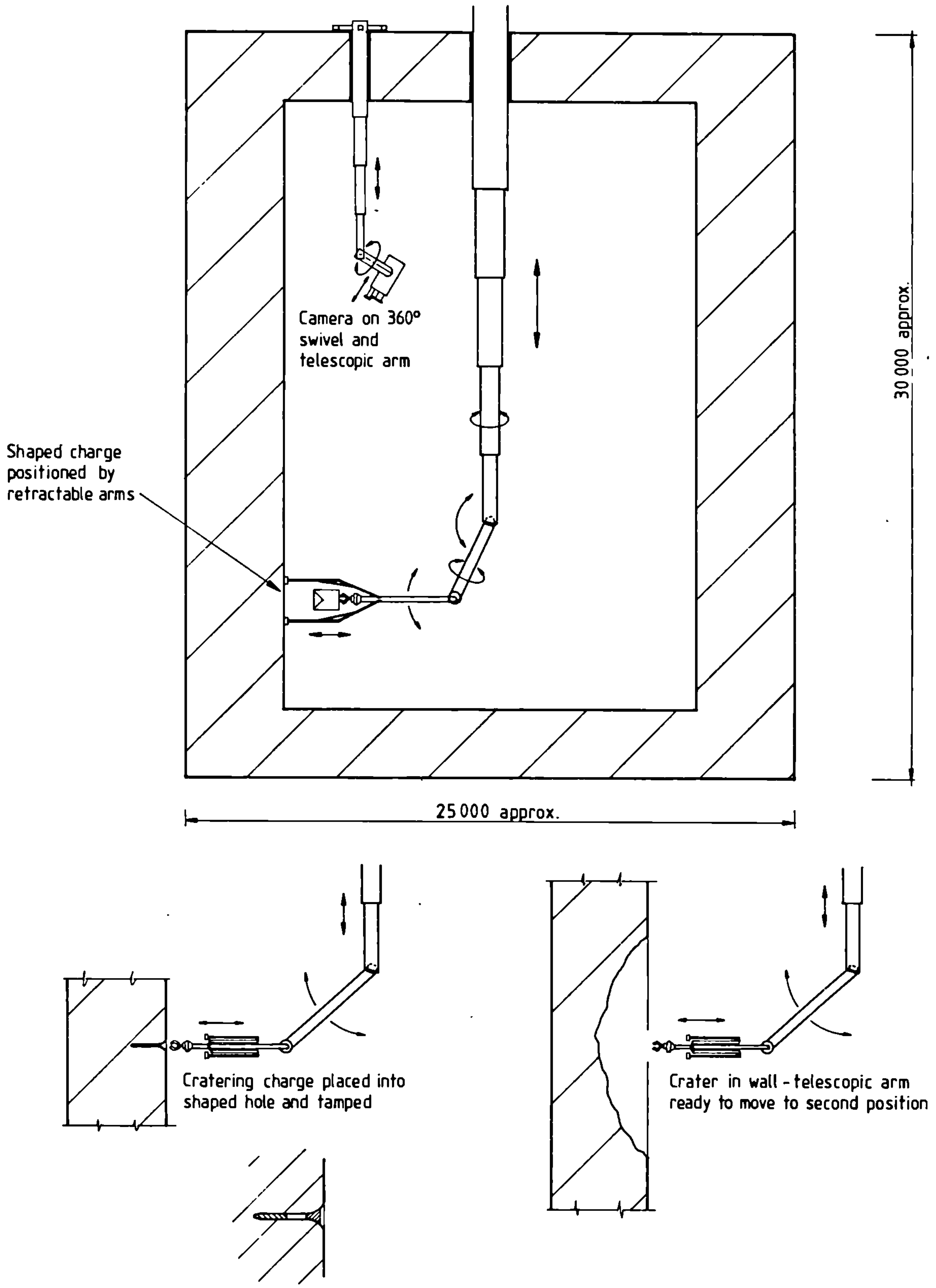


Fig. 3. Concept sketch of remotely controlled manipulator

3.8. Prototype System for Remote Laser Cutting of Radioactive Structures

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FI1D-0013
Working Period: November 1984 - December 1988
Project Leader: J. Gonnord

A. Objectives and Scope

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

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

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

In a supplementary agreement concluded in 1988, the initial work programme was extended to automatic cutting trajectory teaching as defined in working package B.4.

B. Work Programme

B.1. Design, construction and functional testing of a robot arm including remote control and command, and tests on the handling of the arising laser cutting waste.

B.2. Adaptation and coupling of the robot arm to an available laser cutting device.

B.3. Commissioning and demonstration tests of the complete facility, including laser cutting of various non-radioactive stainless steel components with handling of the arising cutting waste.

B.4. Development, construction and testing of a procedure for automatic teaching of cutting trajectories for the ROLD robot by applying CAD procedures based on a 3D laser camera and a laser beam sensor on the torch.

C. Progress of Work and Obtained Results

Summary

The last year has been devoted to arm qualification tests and also the continuing of cutting tests of 3D metallic structures. An inventory of various cutting samples has been listed.

Demonstration tests have been carried out inside a real-size mock-up of a hot-cell equipped with aerosol measurement device (CEN-FAR). This experiment was intended to show the complete system in action - the robot and the power laser - within the frame of a real intervention.

During the last semester, an automatic process has been developed for the trajectory teaching. This process features an active 3D sensor (or 3D camera).

1. Progress and Results (B.1.)

The arm has been completed with a light-alloy cylinder called "cask" which includes the mechanical set of the first axes (Thêta 1). This cask supports the arm and enables its introduction into the normalized telemanipulator hole and the proper positioning. The diameter of the cask is 250 mm and the length is 1500 mm.

The bearings of joints Thêta 2 and Thêta 3 have been modified. Former bearings have been replaced by a stiffer mounting.

2. Optical Path between CO2 Laser and the Arm (B.2.)

The optical path has been adjusted by means of a special device and an alignment method.

This device includes a swinging adjustable plaque wearing a small He-Ne laser. This plaque is situated outside the cell on the external face of the cask so that it remains accessible when the robot is engaged inside the cell. The plaque is first adjusted in a way that the He-Ne beam becomes coaxial with the first mechanical axis (Thêta 1). Therefore this assembly is used as an alignment reference which is helpful to adjust the CO2 power beam.

During our "extra muros" qualification tests, we had to deal with stability problems. We solved this drawback by improving the clamping system of the cask inside the telemanipulator hole.

3. Laser Cutting Tests (B.3.)

A large number of samples have been cut. The range of thickness involved is comprised between 1 and 10 mm. The materials concerned were mild steel and stainless steel.

Several tests were executed on clusters of pipes and large diameter pipes (exceeding 35 mm of diameter).

For the latter case, we adopted a two steps method including the opening of a window and then the cutting of the remaining part by the internal side of the pipe. This method must be retained in most cases where the nozzle cannot go all around the pipe.

Moreover, we showed that a direct linear cutting can be done on parallel groups of small pipes. Pipes up to 35 mm of diameter and 4 mm of thickness have been cut with this method.

We also set-up a list of samples : pipe, metal sheet, U-section, angle steel, etc., showing the various types of geometry practicable with the ROLD robot.

4.<u>Help for Teaching Trajectories</u> (B.4.)

Technical equipment:

A special high-quality 3D sensor has been implemented to develop an off-line programming method. This device gives geometrical 3D data on the spatial localization and the shape of both the sample and the nozzle. We can then compute directly the trajectory with a micro-computer.

Progress of work:

The definitive version of this sensor has not yet been delivered. Nevertheless, we have borrowed a specimen of the standard model, and we wrote a software able to screen the absolute coordinates X, Y, Z of whatever geometrical point of the scene, and to compute the distance between two points. This is the first step of the definitive work which must lead to an automatic programming, concerning the robot control-command and thus replacing the teaching box.

Practical process:

After a complete acquisition of the scene, data are transfered to a micro-computer which makes the scene appear on a screen like a classical image display. The operator can then select a point with the help of an electrical mouse and the coordinates are immediately displayed. After the selection of a second point, the distance would be computed and displayed.

3.9. Investigations of Applications of Laser Cutting in Decommissioning

Contractor: FIAT CIEI S.r.l., Torino and ENEA, Roma, Italy
Contract N°: FI1D-0014
Working Period: January 1986 - June 1989
Project Leader: B. Migliorati

A. Objectives and Scope

The present research work is a follow-up of work performed in the 1979-83 research programme, where it was demonstrated that laser beam cutting has a potential for a useful application to decommissioning, due to the very reduced quantity of generated aerosols and to its possibility for remote operation (Ref.: EUR 9715).

The main objectives of the present contract are as follows: execution of operational cutting tests with 7-10 kW lasers on representative materials of the Garigliano BWR, with a view to determining quantities and distributions of the generated aerosol particles, execution of feasibility studies followed by practical applications using mock-up components. The study will be completed with the assessment of the possibility of using laser cutting of specific components, considering the future decommissioning of the Garigliano BWR.

B. Work Programme

B.1. Characterisation of aerosol arisings from laser cutting.

B.1.1. Execution of cutting tests on materials representative of Garigliano BWR components, such as stainless steel, carbon steel and concrete.

B.1.2. Determination of aerosol quantities and distributions, with special attention to small particles (diameter < 0.5 μm).

B.2. Assessment of the applicability of remote operation laser cutting of specific Garigliano BWR components.

B.2.1. Execution of feasibility studies for remote laser-cutting.

B.2.2. Qualification tests on real-size mock-up components of the Garigliano BWR.

C. Progress of Work and Obtained Results

Summary

In 1988, the research on aerosol characterization (B.1.) has been carried out performing some confirmation test, evaluating the experimental results and investigating the correlation between different parameters.

The tests have been performed on stainless steel and carbon steel samples.

The results' analysis confirms the previous trends; it has been observed that the mass of aerosol produced per unit lenght of cutting, for carbon steel samples, is constantly lower than that produced, in the same conditions, by cutting AISI 304 stainless steel samples. Furthermore tests have been performed in order to investigate the correlation between particle concentration and the particle median equivalent diameter.

As far as the study of the remote transmission and control system of the Laser beam is concerned (B.2.), cutting tests have been performed with positive results at a distance of 30 m on carbon steel samples with thickness up to 35 mm.

The study of application of the Laser beam to the dismantling of the "steam drum" of the Garigliano Nuclear Power Station is in progress.

Progress and Results

1. Characterization of aerosol: (B.1.)

A series of tests were carried out with the aim of verifying the effect of the various cutting parameters on the amount of aerosol and drosses produced and confirming data obtained in the previous experiences.

The reason for continuing this investigation was that other tests were needed to find the laser cutting conditions that minimize the aerosol production rate at source.

The tests, performed mainly by cutting 5 mm thickness plates, were not limited to stainless steel (AISI 304) but also extended to some carbon steel (Fe 42C) plates.

The data confirm the previous trends, even though a certain variability can be noted, concerning the effect of cutting velocity, plate thickness and type of assistant gas on the effluent production rate. The laser power seems to be less influential than previously reported.

In case of cutting of carbon steel plates, the mass of aerosol produced per unit length of cutting is constantly lower than that produced, in the same conditions, by cutting AISI 304 stainless steel.

The reduction is in the range of factors 1.2 to 3.2 in the case of cutting by Oxygen and of a factor of about 2.6 in the case of cutting by air.

Two series of tests have been performed in order to investigate the correlation between particle concentration and the particle median equivalent diameter.

In the first series the amount of aerosol produced was varied by changing the assistent gas in such a way that different particle concentrations were obtained at the source. A correlation between particle equivalent diameter and production rate could be noted; the variations found are probably caused by different coagulation states due to different particle concentrations at the source.

A second series of tests have been performed by containing the aerosol in the cutting box without ventilation, to evaluate the granulometric modifications at high particle concentration in calm air.

Tests had been preceded by a very accurate antistatic treatment of the cutting box walls in order to avoid electrostatic effects on the particles not directly connected with coagulation phenomena. The rapid increase of the median mass diameter is confirmed, but lower than that reported in previous tests when electrostatic interactions with the walls caused artifacts.

Standard cuttings with laser beam were executed in the frame of an intercomparison programme coordinated by CEA-Saclay and Hannover University/ I.W.

Finally a series of experiences were made in order to check the feasibility and the effectiveness of a local aspiration of the aerosol produced obtaining a reduction of more than 2 order of magnitude of the concentration in the surroundings.

2. Assesment of the applicability of remote operation laser cutting of specific Garigliano BWR components. (B.2)

In order to have a more close agreement between theoretical and experimental data have been reconsidered the optical laser parameters. Therefore it was necessary to calculate the new propagation beam, utilizing the same optical system, both near and far away from the output window of the radiation source. The experimental check of the beam dimensions along the path and on the working plane have confirmed the theoretical prediction. It was obtained a 43x58 mm beam on the focalizing element and 1.2:1.3 mm on the focal plane, and more than 1 kW of power losses along the path due, principally, to the mirrors absorption.

Cutting tests, on AISI 304 steel 15 and 30 mm thick, have been performed at 30 m from the laser output with 4.1 kW and oxygen gas at 1.5 MPa. One cut has been done on a sample 35 mm thick with a 5 mm AISI 304 coating at 0.1 m/min (Figure 1).

In order to eliminate the lens breakage during the cutting process, due to some metal molten particles, a metal (copper) focalizing element was built and used. In Figure 2 is shown the optical head with the focussing part (black) and the nozzles for the assistant gas on the bottom.

A study of a laser system for cutting the "steam drum" (Figure 4), of the Garigliano Nuclear Power Plant, has been done. Starting from a commercial source it was necessary to find, with a theoretical-experimental method, the beam parameters in order to calculate the propagation data for two different final focal lengh. The sensibility of changing the source characteristics on the power density in the focal plane has been evaluated.

It has been noted that it is necessary to put special attention on the mechanical structure (Figure 3) of robot and of translation stage, on the lighting and vision system for the working area, on the memorization and on alignment remote sensing between robot entrance and the beam.

The study concerning the definition of characteristics and the installation inside the Garigliano Nuclear Power Station, of the laser apparatuses required for dismantling the "steam-drum" and of the transfer system required for evacuating the containers of the cut pieces, is in progress.

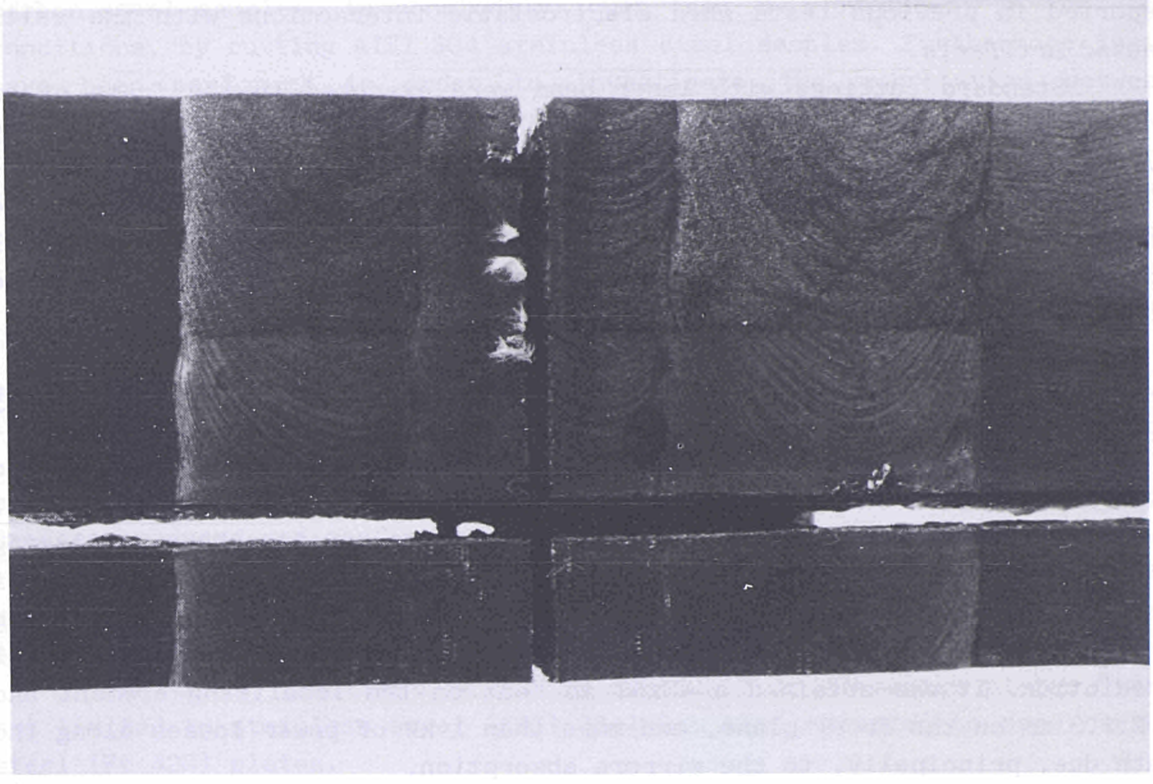


Figure 1 - Cut on one carbon steel sample
35 mm thick, 5 mm AISI 304 coating.

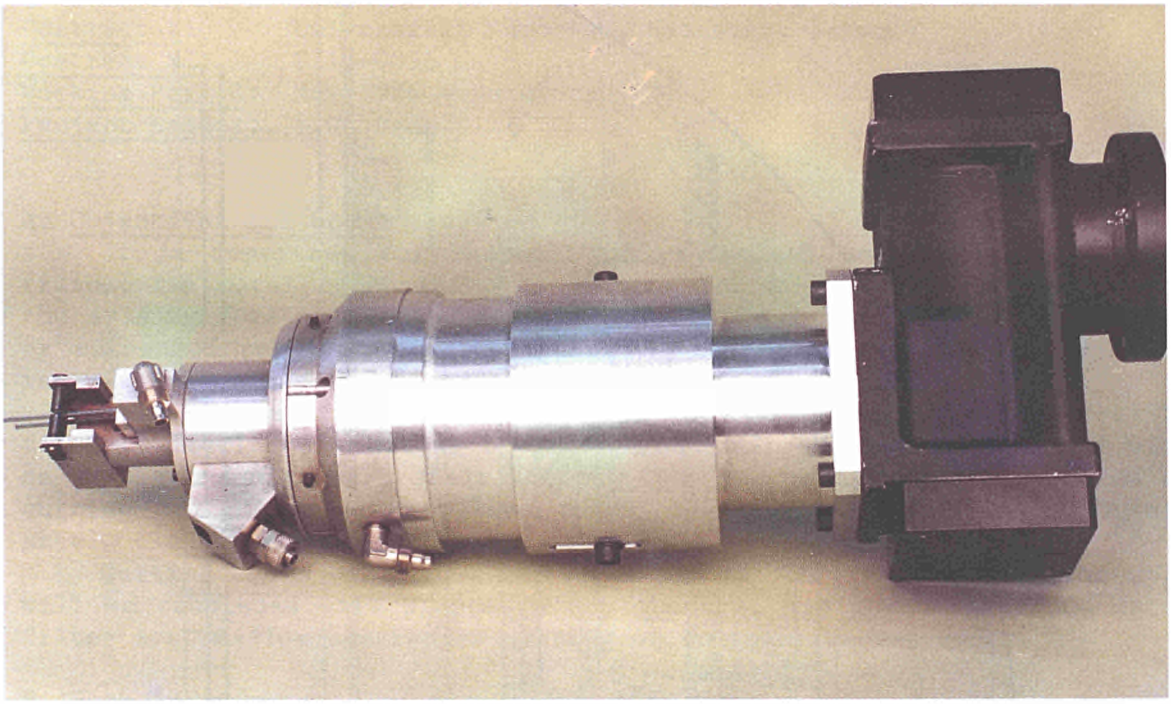


Figure 2 - Optical head

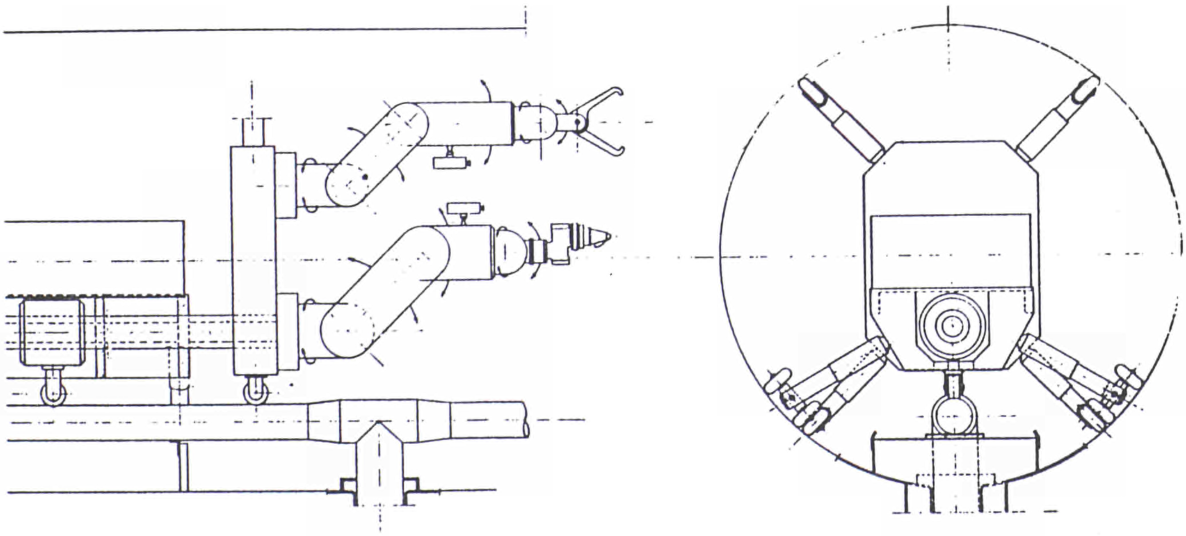


Figure 3 - Robot for cutting inside
the steam-drum

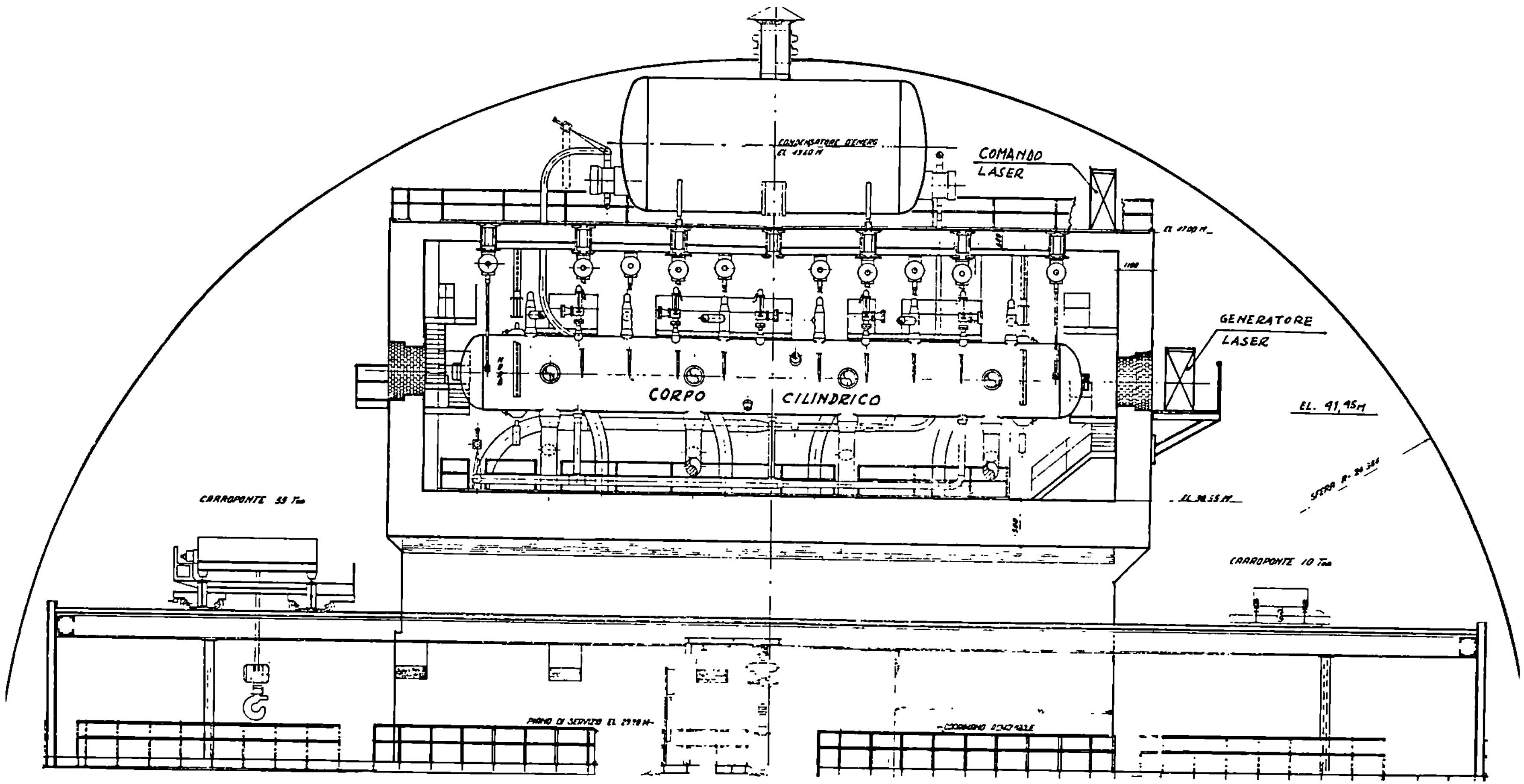


Figure 4 – Steam drum (Corpo cilindrico) of the Garigliano BWR

3.10 Spreading and Filtering of Radioactive By-products of Underwater Segmenting

Contractor: Universität Hannover, Hannover, Germany
Contract N°: FI1D-0036
Working Period: May 1986 - December 1988
Project Leader: F.W. Bach

A. Objectives and Scope

It is important for thermal and mechanical underwater cutting of radioactive metal components, to provide for an efficient collection of the arising cutting by-products, thus avoiding a reduction of visibility by suspended particles and reducing the contamination of water and the radiation level at the water surface.

This work aims mainly at studying various filter systems for efficient collection of cutting by-products in air and water, combined with an in-depth analysis of their distribution and quantities as function of cutting method (grinder, plasma torch) and cutting material (stainless steel, cladded carbon steel, aluminium).

Cutting tests will be executed on non-radioactive samples. The work will be concluded with proposals for the most appropriate air and water filter systems for underwater cutting of radioactive materials.

B. Work Programme

B.1. Modification and adaptation of the existing test facility for the provided work programme and purchasing of supplementary instrumentation.

B.2. Selection and definition of the main parameters for cutting tests with grinders and plasma arc torch.

B.3. Execution of the test programme on the distribution and concentration of particles arising in air and water with various filter systems.

B.4. Assessment of the efficiency and effective standtime of air filters and water filters.

B.5. Chemical analysis of the cutting by-products, found in air and water.

B.6. Conclusive assessment for an optimisation of filter systems for underwater cutting.

C. Progress of work and obtained results

Summary
The suspended particles in the water tank used for underwater plasma arc cutting were analysed in order to get a database for possible waterfilters. The particle size distributions were measured with an optical particle counter after cutting stainless steel, mild steel, cladded steel and aluminium.
Different systems of water filters were tested.

Progress and results

1. Measurement of cutting effluents (B. 3)

The principle of the plasma arc cutting is, that the material to be cut is molten by the plasma arc and then a kerf is formed because of the high kinetic energy of the plasma gas. The molten kerf material is spread into single particles of a great size range.
There are arising different types of cutting effluents:

- particles, which come to atmosphere as aerosols
- gaseous effluents
- suspended particles, which have a great influence for the possibility to observe the cutting process. They may reduce the quality of protection against radiation of the water.
- sedimented particles which settle to the ground of the basin
- solved material from the reaction of the hot metal with the water. This is the cause for the changing of the pH-degree and the conductivity of the water.

The cutting tests for the analysis of the cutting by-products in the water were done in a small tank made of stainless steel. One wall of the tank is made of plexiglas to control the position of the torch and the cutting process. It was possible to cut in a water depth of 100 mm needing a volume of about 50 l of water. It was not usefull to do the test in the great facility (see APR 1986) because it is possible to get high concentrations of particle in the small tank with less material.
The water used for the tests was normal drinking water. It was filtered first with a commercial 0.2 µm membrane filter before it was filled into the tank. Samples of the feed water were taken for the chemical analysis of the solved components.
The analysis of the suspended particles was done with an optical particle counter of the type HC-15, manufactured by the Polytec company. The measuring principle depends on the scattering of white light in an angle of 90° degrees. The measuring volume is defined in an optical way so that the flow of the particles is not disturbed.
The flow rate of the instrument is about 2.1 liters per minute. A speed of the particles of 5 m/s is possible.
The range of the measurement is 1.3 µm to 55 µm devided in 127 classes. The particle size analysis was done within the first two minutes after the end of cut to avoid an agglomeration of the particles.
The results of this analysis after cutting stainless steel, carbon steel, cladded steel and aluminium are shown in table I. There is a decrease of the mean particle diameter in the particle number distribution depending on the sheet thickness when cutting stainless steel.
The results of the particle measurements after plasma cutting of aluminium show an increasing of the mean particle diameter in the particle number distribution depending on the sheet thickness.

The results of the chemical analysis of the base materials as a function of the thicknesses as well as the analysis of the kerf material are given in table II. There is nearly the same composition of elements in the original material and in the sedimented kerf material.
Only the concentration of manganese in the kerf material of the stainless steel is decreasing compared to the sheet material. The same decrease is found in the concentration of magnesium in the kerf material of the aluminium sheet.

The analysis of the water quality after plasma cutting of the above materials and with different sheet thicknesses (table III) corresponds to the results of the analysis of the kerf material. There is a decrease of the manganese concentration in the kerf material of stainless steel and a rise of the solved manganese concentration.
The same happens with the magnesium in the kerf material of the aluminium sheets.

2. Filter tests (B.5)

With the results of the particle size measurements it was possible to choose some water filters for filtration tests.
The tests were carried out in a special tank where it is possible to cut sheets of 1 m length. The filtration tests were done after cutting stainless steel, mild steel and aluminium.

The particle concentration in the water after cutting was:

- stainless steel	1.4301	44 mg/l
- mild steel	St 37	34 mg/l
- aluminium	AlMg3	113 mg/l.

Figure 1 shows a schematic view to the experimental facility. A pump gets the water out of the tank through the filter. The flow rate through the filter is controlled by a rotameter. The pressure difference is taken by two pressure gauges.
The pressure curves for two different filter typs are given in figure 2 and figure 3. Both filters have the same nominal particle size (2 µm) for an efficiency of 100 %.
The filter in fig. 2 has a nominal flow rate of 11 l/min. It is a glass fiber filter. The filter according to figure 3 is constituted of polypropylen with a nominal flow of 9 l/min.
Obviously there is a great difference in the filtration capacity among these two filters depending on the filter material.
That means e.g. that the glass fiber type allows a total flow of 960 l corresponding to a filtration capacity of 42 g to reach a pressure drop of 4 bar (stainless steel, 1.4301). Under the same conditions the polypropylen type allows a total flow of only 225 l (corresponding filtration capacity 10 g) to come to a pressure drop of 4 bar.

Table I: Particle size analysis of suspended particles

Particle Size Analysis of Suspended Particles			
material	thickness	xm	x50%
1.4301	20 mm	8.1 um	25 um
1.4301	40 mm	5.8 um	32 um
1.4301	60 mm	4.5 um	22 um
St 37	20 mm	6.1 um	20 um
St 37	40 mm	3.5 um	20 um
St 37	60 mm	6.1 um	21 um
cladded steel	18 mm	5.6 um	30 um
AlMg3	10 mm	5.4 um	28 um
AlMg3	20 mm	6.7 um	27 um
AlMg3	40 mm	7.7 um	22 um

Table II: Chemical Analysis of the cut material and the kerf (SG)

Chemical Analysis											
material	mm	Al	C	Cr	Mg	Mn	Ni	Nb	Si	Ti	Zn
1.4301	20	-	0,02	17,8	-	1,56	9,0	0,01	0,28	0,13	
SG				17,4		1,5	9,6	-	0,34	-	
1.4301	40	-	0,02	16,7	-	1,7	9,4	0,05	0,32	0,52	
SG		-	-	16,6	-	1,42	9,0	-	0,35	-	
1.4301	60	-	0,03	16,4	-	1,5	10,1	0,03	0,36	0,36	
SG		-	-	17,5	-	1,36	10,1	-	0,42	-	
St 37	20	0,03	0,12	0,02	-	1,5	-	-	0,26	-	
SG		-	-	-	-	1,5	-	-	0,12	-	
St 37	40	0,11	0,14	-	-	0,47	-	-	0,31	-	
SG		-	-	-	-	0,54	-	-	0,34	-	
St 37	60	-	0,27	-	-	0,56	-	-	0,25	-	
SG		-	-	-	-	0,48	-	-	0,24	-	
Alu	10	95,2	-	-	2,54	0,32	-	-	0,18	-	0,05
SG		Rest	-	-	2,2	-	-	-	0,24	-	0,04
Alu	20	94,7	-	-	3,0	0,38	-	-	0,18	-	0,08
SG		Rest	-	-	2,0	-	-	-	0,32	-	0,06
Alu	40	94,8	-	-	3,3	0,07	-	-	0,14	-	0,04
SG		Rest	-	-	1,3	-	-	-	0,22	-	0,04

Table III: Analysis of the solved elements in the water

Wateranalysis								
material	mm	Al	Ca	Cr	Mg	Mn	Si	Zn
1.4301	20	-	55	0,15	-	0,5	4	-
1.4301	40	-	55	0,53	-	1,6	4	-
1.4301	60	-	55	0,36	-	0,8	4	-
St 37	20	-	55	-	-	0,3	4	-
St 37	40	-	55	-	-	-	4	-
St 37	60	-	55	-	-	-	4	-
Alu	10	0	55	-	5,6	-	4	0,05
Alu	20	0	55	-	6,4	-	4	0,05
Alu	40	<1	55	-	5,9	-	4	0,05
water before cutting		0	55	0	4,8	0	4	0,11

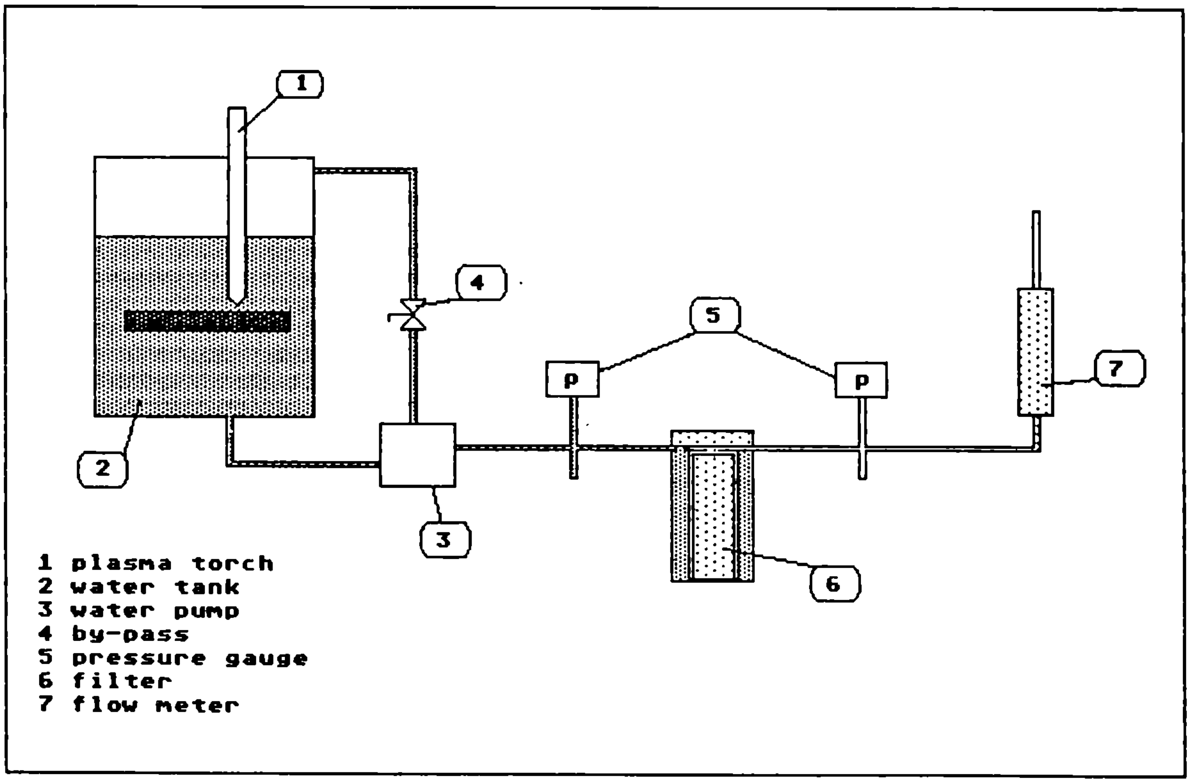


Figure 1 : Water filter testing rig

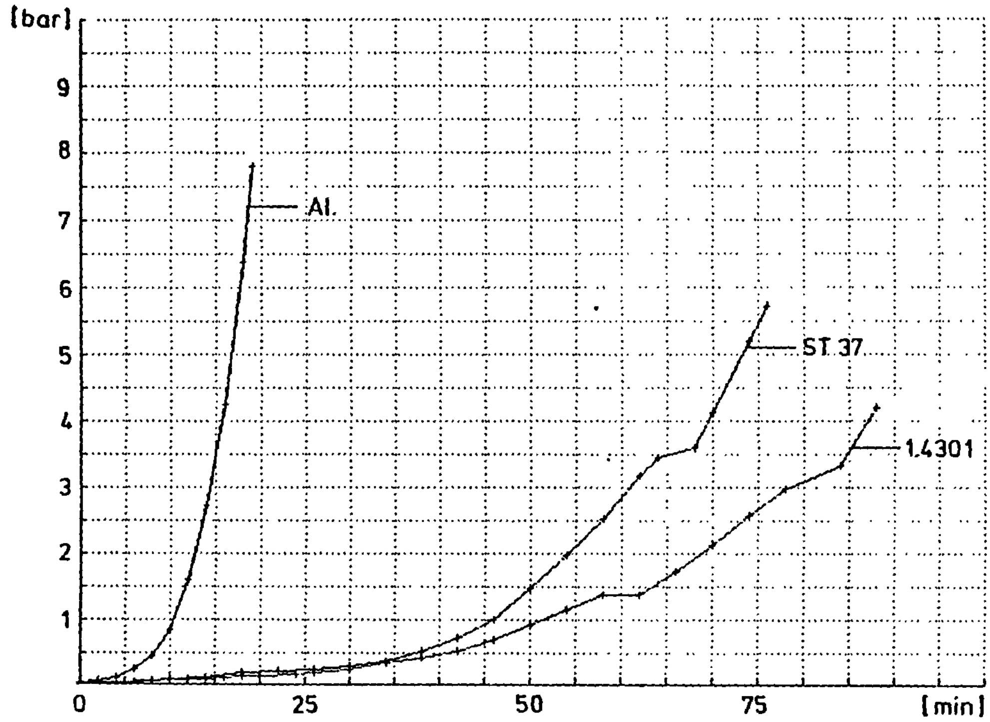


Figure 2: Pressure curve of a 2 µm glas fiber filter

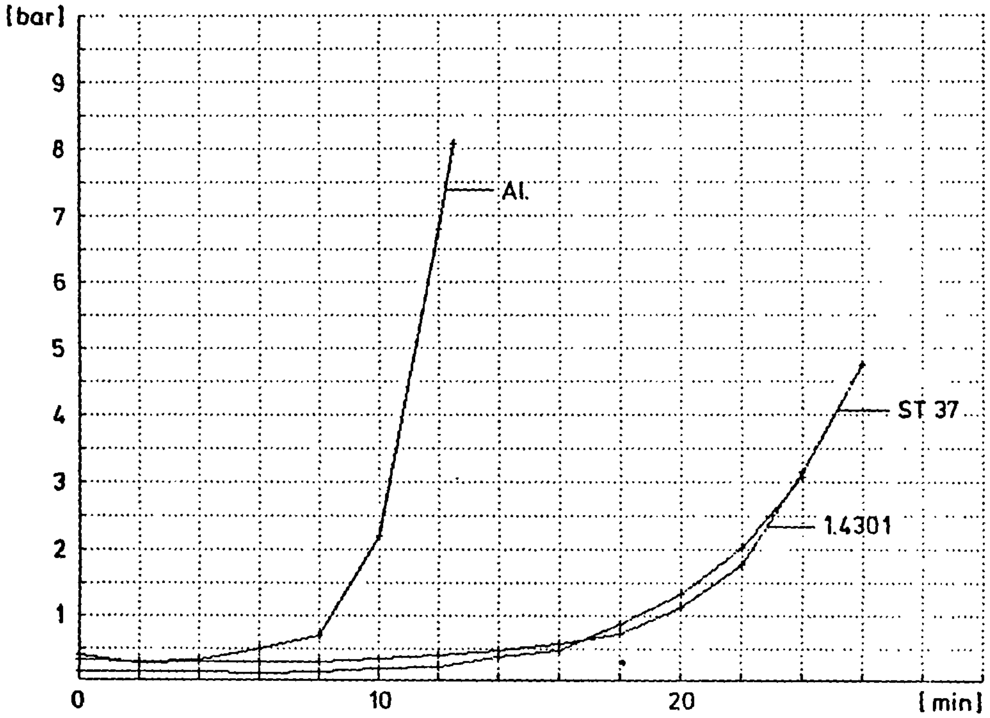


Figure 3: Pressure curve of a 2µm polypropylen filter

3.11 Development of a Prototype System for Remote Underwater Plasma-arc Cutting

Contractor: Commissariat à l'Energie Atomique, CEN Cadarache, France
Contract N°: FI1D-0037
Working Period: May 1986 - September 1989
Project Leader: R. Léautier

A. Objectives and Scope

Based on an extensive foregoing experience with underwater plasma-arc cutting, the present contract is intended to further develop this technique with the following objectives:

- remote and automated cutting of complex and thick-walled metal structures;
- evaluate the physico-chemical status of the water;
- minimise the generation of cutting by-products.

Cutting tests are executed both with non-radioactive and low-level radioactive samples of stainless steel.

In a supplementary agreement concluded in 1988, the initial work programme was extended to the development, construction and testing of a system for automatic cutting (positioning and guidance of torch, cutting control) and for remote maintenance, as defined in the new B.6. working package.

B. Work Programme

B.1. Modification of the presently available motorised cutting table, allowing for the dismantling of more complex structures.

B.2. Execution of cutting tests aimed at optimising the main cutting parameters.

B.3. Final adaptation of the cutting table, based on the experience gained in working step B.2.

B.4. Execution of cutting tests on non-radioactive stainless steel, aimed at a parametric study of the quantity and distribution of the arising cutting by-products and of the evolution of the physico-chemical status of the water.

B.5. Preparation, facility adaptation and execution of cutting tests on radioactive steel, including the preparation of an activity balance of the gaseous and liquid cutting by-products.

B.6. Development, construction and testing of a procedure for automatic cutting and for remote maintenance.

B.7. Conclusive assessment of the potential for industrial application of automated plasma-arc cutting of radioactive components.

C. Progress of Work and Obtained Results

. Summary

The work carried out in 1988, was connected with three main aims, in relation to the topic of work programme package B4, B5, B6.
Firstly, in order to give the necessary elements to the dismantler we synthetized the tests made in 1987 with a radioactive global balance in our installation associated to the ventilation system.
Secondly, some standard cutting tests with non-radioactive stainless steel plates and selected measurements apparatus allowed us to measure suspended and sedimented particles, aerosols and gases.
Thirdly, some specifications were established for the making of a torch capable of undergoing a remote maintenance, and for an assistant system allowing the operator to pilot and control the cutting.

. Progress and Results

1. Synthesis of the tests carried out in 1987 on radioactive materials (B.5.)

These works made in close cooperation with the CEA Saclay are already described in the frame of the Annual Progress Report relative to the contract n° FI 1D 007 F.

2. Standardized cutting tests on non-radioactive materials (B.4.)

These cutting tests were made in close cooperation with the CEA of Saclay, the University of Hannover and the Heriot-Watt University of Edinburgh. In these tests, sample materials, cutting parameters, methods and measurement means were identical.
The obtained results concerned the attached and suspended slags, sedimented drosses, and the mass concentration of the aerosols. The sedimented drosses represent the main part of the slags and drosses produced during the cutting (about 90%). The concentration of the suspended slags is lower than 1 mg/liter.
The mass concentration of the aerosols is about 3 to 7 per 100.000 of the secondary solid emissions. Some complementary, chemical analyses and tests with other cutting velocities will be made in 1989. So the definitive results will be presented in the final report.

3. Cutting device for remote maintenance and assistant system for the operator during cutting (B.6.)

Following the tests carried out on radioactive materials and already described in the preceding Annual Progress Report, we specified a torch of which the wearing parts could be replaced underwater by remote control (See Figure n°1). These parts will easily be separated from the body of the torch by a quick closing system.
The torch itself could also be taken out of the holder.
These operations could be made with no position adjusting.
Furthermore we studied a system to assist the operator during the difficult approach phase of the tool and the control of the cutting trajectory. (See Figure n° 2).
This system would mainly be composed of a ball-bearing mecanism keeping constant the distance between the torch and the workpiece, a pneumatic spring set with gas capacity allowing the torch to follow the cutting surface, a servo-system using the information of an inductive sensor giving the image of the pneumatic spring displacement.

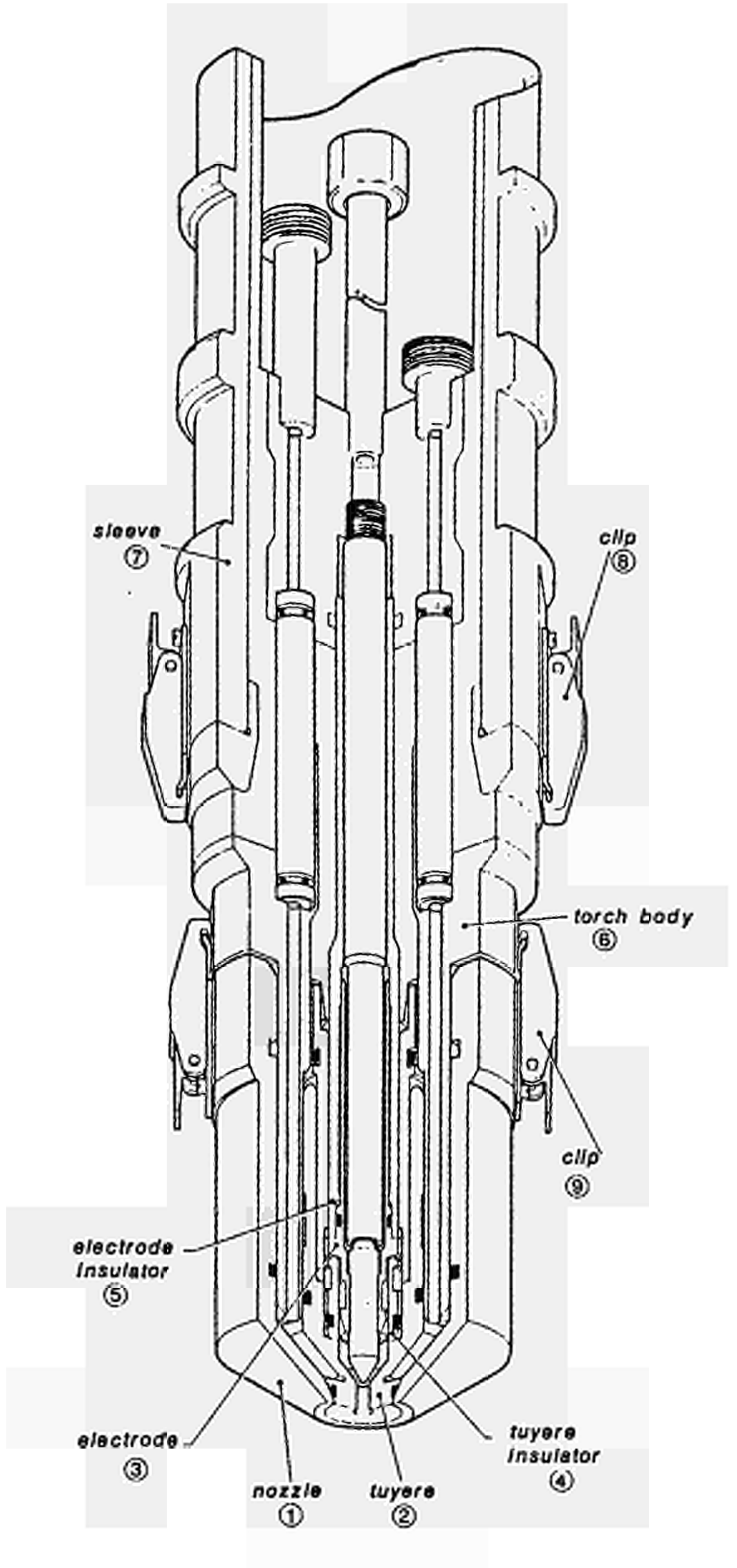


Figure 1 - Plasma torch for remote replacement of wearing parts

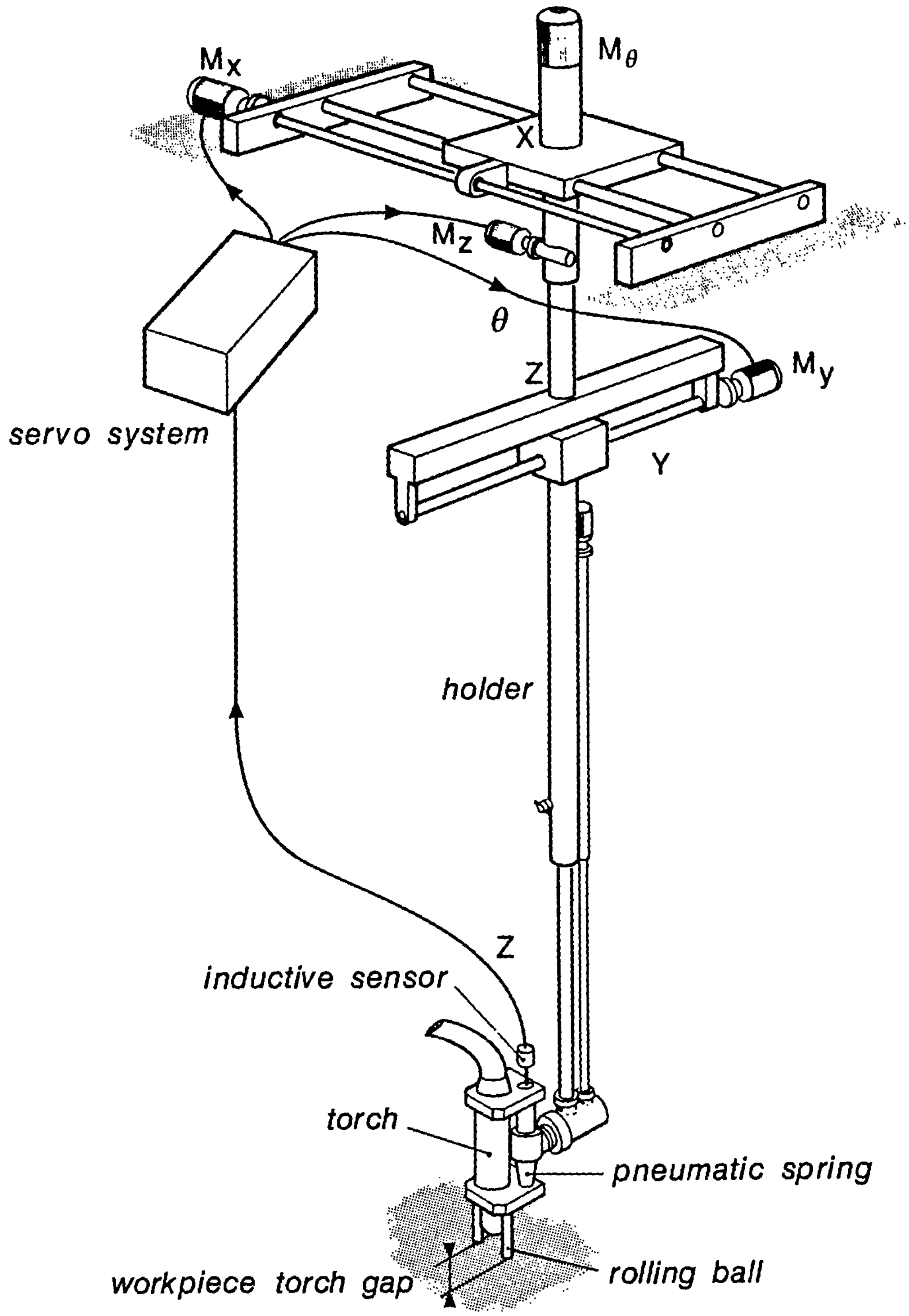


Figure 2 - Plasma torch for automated cutting

3.12 Adaptation of a Robot and Tools for Dismantling of a Gas-cooled Reactor

Contractor: Strachan & Henshaw, Bristol, United Kingdom
Contract N°: FI1D-0038
Working Period: January 1987 - August 1988
Project Leader: M. Wyatt

A. Objectives and Scope

The use of programmable computer-controlled manipulators in the nuclear industry is relatively recent. Existing programmable manipulators in radioactive areas are special-purpose designed machines, tend to lack the sophistication of industrial robots and are very expensive.

The main aim of this research is to develop a design for a manipulator suitable for use in dismantling nuclear reactors which utilises the control system, and as many mechanical features as possible, of an industrial robot.

To enable the dismantling of reactor structures, various types of tools must be capable of cutting through a variety of materials.

In association with the manipulator development a second aim is to investigate cutting tools suitable for reactor structure dismantling and a means of changing them remotely.

B. Work Programme

B.1. Preparation of a manipulator survey aimed at defining an industrial robot and computer control system with a number of given basic requirements.

B.2. Setting-up of specifications for a fully programmable nuclear manipulator, based on operational experience (see B.3.)

B.3. Acquaintance of operational experience on a hired robot.

B.4. Preparation of an appropriate preliminary manipulator design, taking into consideration specific requirements for operations under nuclear conditions.

B.5. Assessment and possible modification of appropriate cutting tools for reactor dismantling.

B.6. Design study of remote-operated tool change facilities.

C. PROGRESS OF WORK & OBTAINED RESULTS

SUMMARY - The project is concerned with the feasibility of adapting a standard industrial manipulator such that it will be able to perform efficiently within the rigors of a nuclear environment.

The technology for the type of control and the machines to perform in nuclear applications already exist in the form of industrial type robots. Development of the techniques for using these machines in a more operator sensitive environment, together with the requirements of decontamination and radiation tolerance will enable them to be used in place of expensive purpose built machines at a considerable cost saving.

A survey has been conducted into the types of industrial robot and control systems currently available to assess the most effective units for decommissioning operations. This was combined with an investigation into the peripheral systems available to existing robot equipment.

The specifications of a nuclear manipulator were defined and from this it was possible to make observations about a suitable design.

A survey was conducted into hand held tooling, thus defining the most applicable equipment for decommissioning operations. This was combined with a design study of the feasibility of a remote tool change facility for the tools highlighted in the above survey.

From this work it was possible to highlight the viability and associated costs of modifying a standard manipulator for use in decommissioning operations.

PROGRESS AND RESULTS

1. PREPARATION OF A MANIPULATOR & CONTROL SYSTEM SURVEY (B.1.)

A survey of all major manipulators and control systems has been completed. This highlighted the most suitable units for use in nuclear activities.

2. SETTING UP MANIPULATOR SPECIFICATION (B.2.)

The outline specification for a fully programmable nuclear manipulator has been defined. The aim being to identify whether a standard manipulator can meet these specific requirements with some modifications.

3. ACQUAINTANCE OF OPERATIONAL EXPERIENCE ON A HIRED ROBOT (B.3.)

This part of the programme was concerned with a detailed study of an industrial manipulator and control system. This was undertaken during a three month hire period. A test schedule was written to provide an in-depth insight into the workings of the manipulator. The test schedule focused on the problems of using an industrial machine in a nuclear environment. Particular attention was given to the problems of having man in the loop control and remote operations with visual information provided through television monitors. Further research was aimed at identifying peripheral control systems combined with a study of available sensory devices.

4. PREPARATION OF MANIPULATOR DESIGN (B.4.)

To assess the required modifications to a standard manipulator design, the specific tasks and associated radiation levels were evaluated. This highlighted the fact that the greater the levels of expected radiation the higher the number of modifications.

To fully assess the implications of radiation exposure the study of manipulator design was aimed at particular tasks and associated radiation levels. This highlighted the following modifications required and the associated costs. See Figure 1 & Figure 2.

MODIFICATIONS TO STANDARD MANIPULATOR

a) Radiation hard wiring loom.
b) Radiation hard gear lubricants.
c) Gaiters for all joints.
d) Remote tool change plate.
e) Replacement of positional encoders with resolvers.
f) Smooth profile to reduce particle traps.
g) Radiation hard paint.
h) Positive internal pressure to reduce contamination.
i) Geometry and configuration changes.

All the modifications required to a standard robot were shown to be feasible and for the lower levels of radiation exposure considerable cost savings over special nuclear manipulators were identified.

5. ASSESSMENT OF APPROPRIATE CUTTING TOOLS (B.5.)

A study was undertaken to outline the most suitable equipment for envisaged decommissioning operations. This highlighted the following equipment.

1. Oxypropane cutter with provision for iron/aluminium powder feed.
2. Bolt and cable cropper.
3. Disc cutter.
4. Internal tube cutter.
5. Heavy duty vacuum cleaner for dust and debris.

Most of the above tooling would be used on a throw away basis upon failure. However, minor modifications may be necessary for the power feed system to the tooling for remote operation.

6. DEVELOPMENT OF REMOTE TOOL CHANGE FACILITY (B.6.)

The tool change facility was achieved through a pneumatic system. The tool change components are two mounting plates; a male plate mounted to the robot arm and identical mating female plates mounted to each item of hand held tooling.

Exact location for the mating plates is achieved through hardened conical dowel pins. The lock/release mechanism is pneumatically operated by a 2-way solenoid valve. All gas and air connections are self sealing so as not to permit leakage into the working environment.

The tool change will be feasible through a library of tool locations relative to the manipulator base combined with a set of pick up and put down programmes for each of the tools.

A schematic drawing of the proposed facility is shown in Figure 3.

CONCLUSIONS

The use of industrial manipulators in nuclear environments is feasible, with all required modifications practicable with existing technology. The motivation behind the project was to identify whether industrial robots with proven control systems could be used in nuclear environments in place of specialist nuclear manipulators. This has been shown to be the case with considerable cost savings being made at the lower end of the radiation scale.

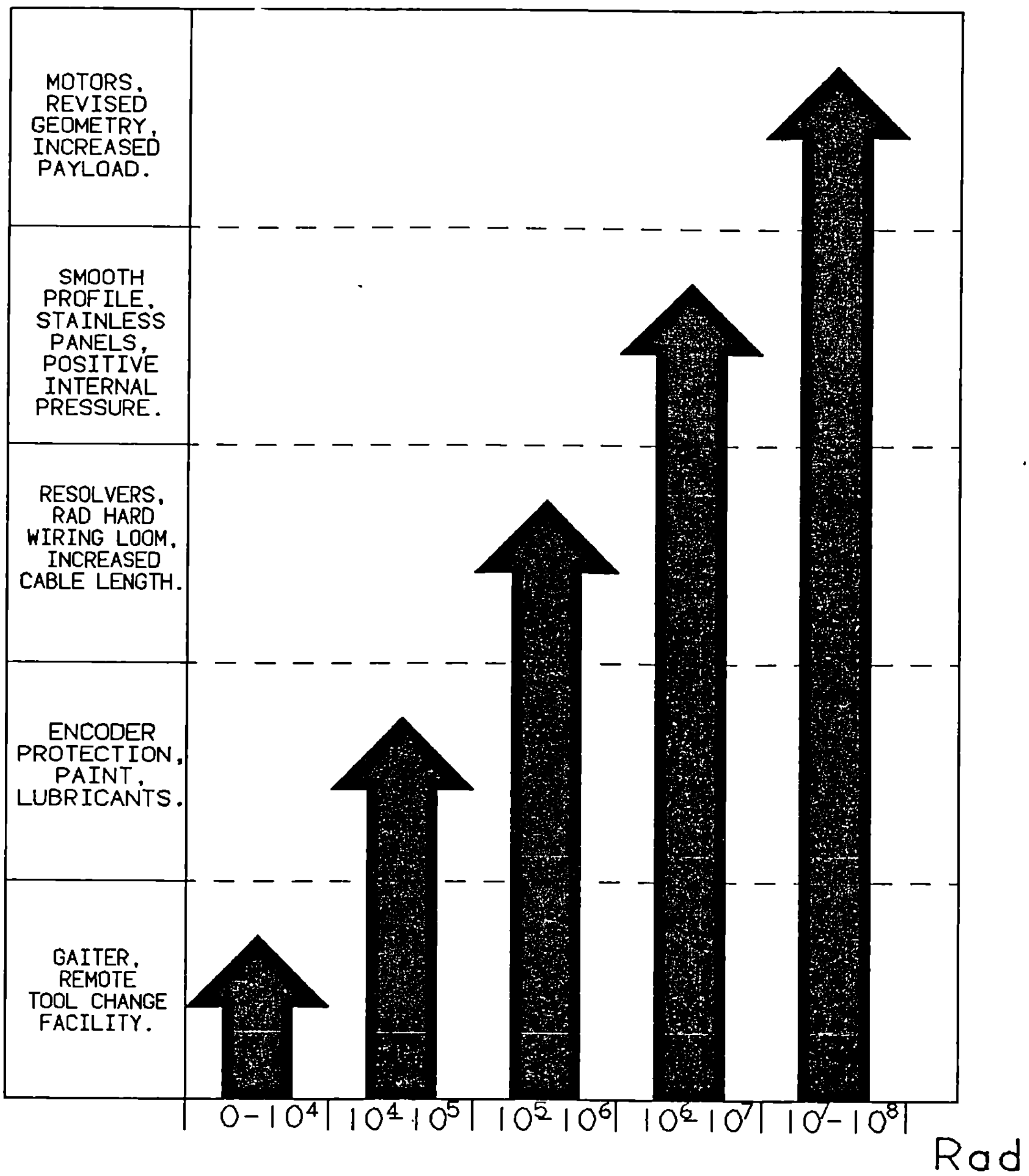


FIG. 1
MODIFICATIONS TO STANDARD MANIPULATOR VERSUS EXPECTED RADIATION LEVEL

(RAD)

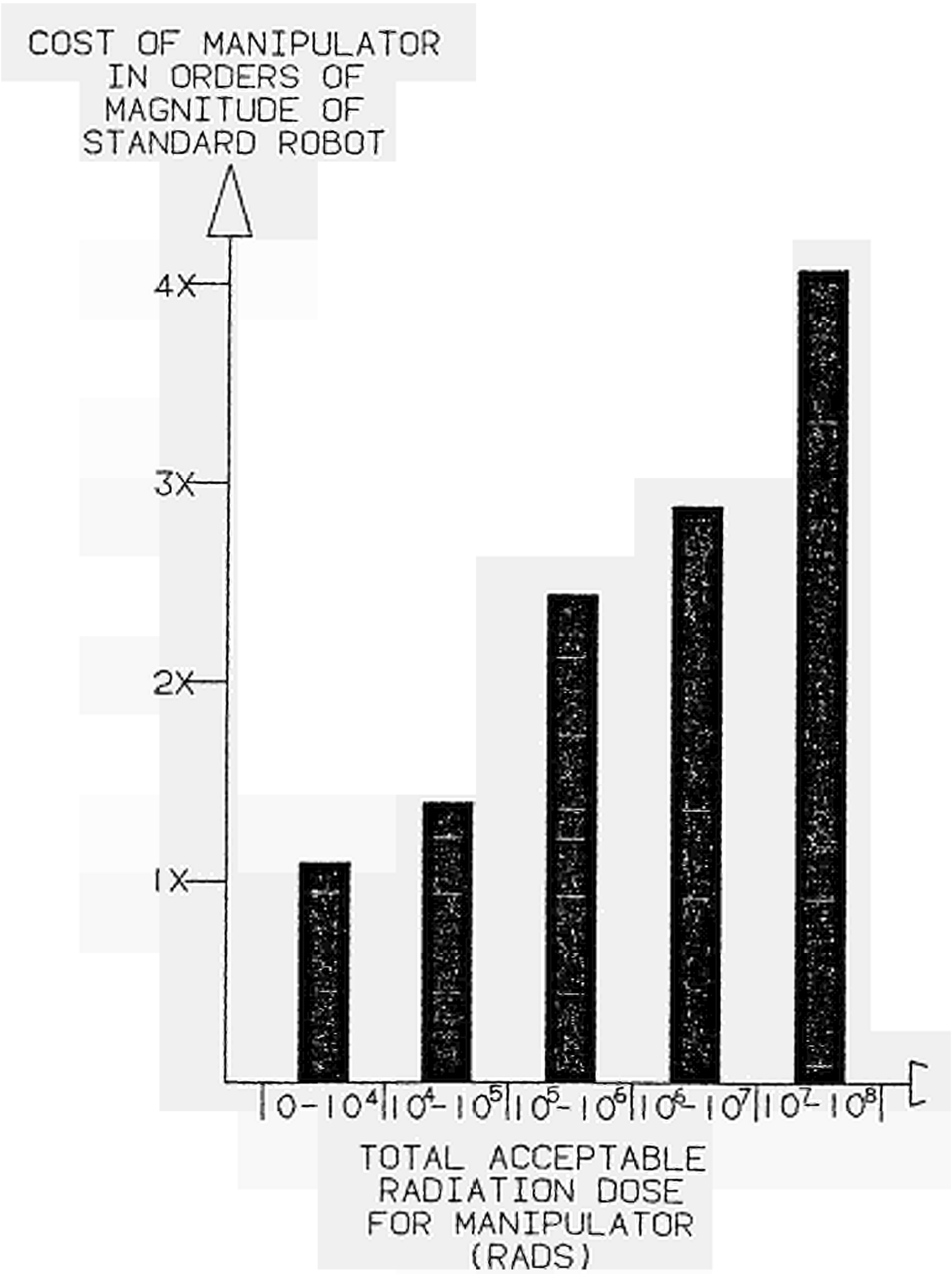


FIG. 2

APPROXIMATE COST
VERSUS
RADIATION TOLERANCE
OF MODIFIED
STANDARD ROBOT

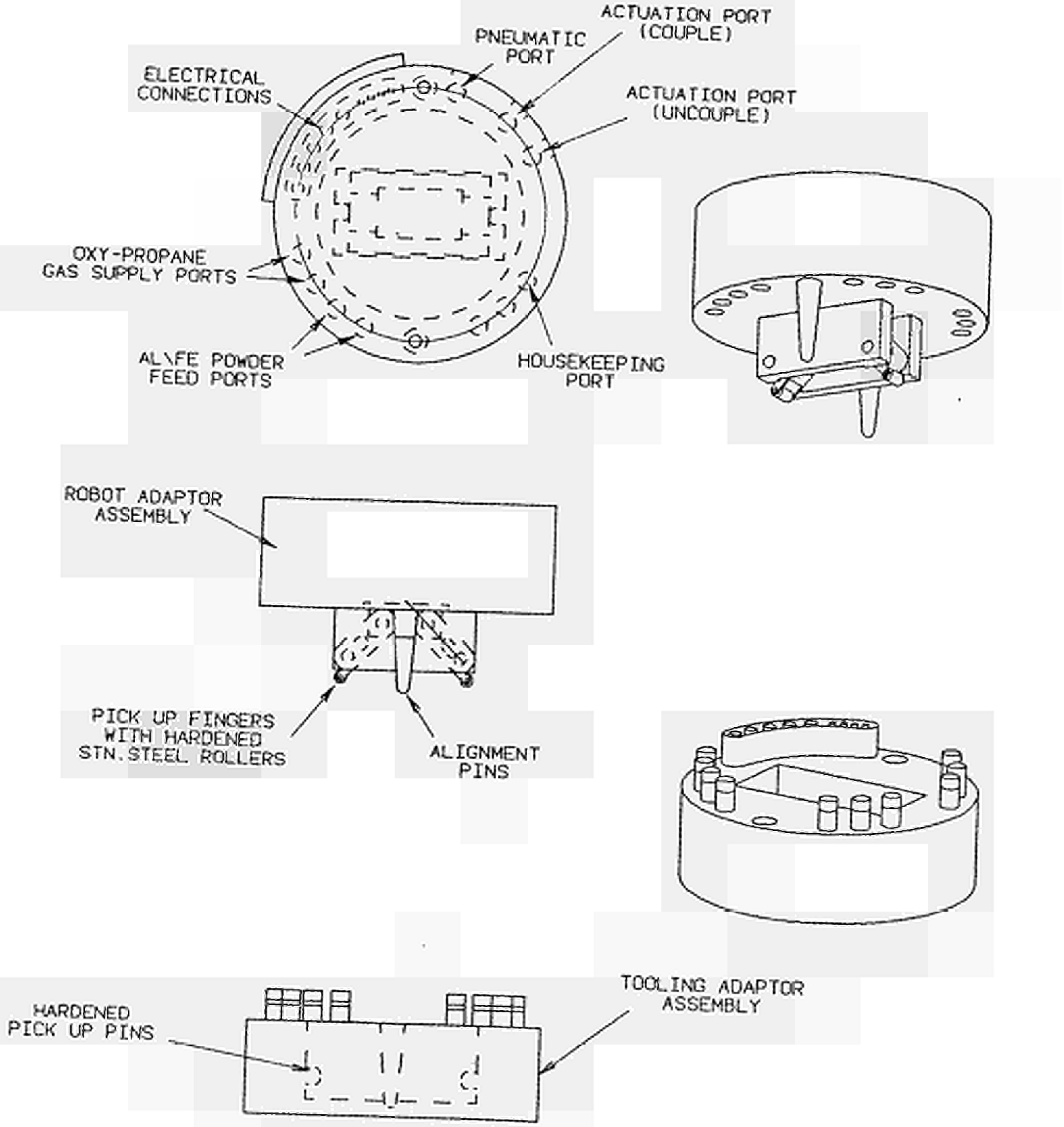


FIG. 3
REMOTE TOOL
CHANGE FACILITY

3.13 Remote Measuring and Control Systems for Underwater Cutting of Radioactive Components

Contractor: Rheinisch-Westfälische Technische Hochschule Aachen, Aachen, Germany
Contract N°: FI1D-0039
Working Period: April 1986 - June 1989
Project Leader: P. Drews

A. Objectives and Scope

Decommissioning of nuclear installations requires special techniques for the dismantling of components. Cutting of the higher-level radioactive components is preferably performed under water. To assure adequate cutting quality, some essential problems remain to be solved, such as adaptive parameter control, exact positioning of the cutting tool and control of cutting actions under water. Suitable control systems and special sensors have to be made available.

The principal aim of this research is to contribute to high-quality cutting under water by the development and application of innovative control systems and sensors appropriate for a wide range of dismantling tasks in nuclear installations.

The developed system will be tested by application to various underwater cutting procedures in collaboration with Universität Hannover (contract N° FI1D-0036).

B. Work Programme

B.1. Design and assembly of an appropriate system for underwater work piece recognition, including optical sensing, image processing and analysis, followed by practical testing with various cutting techniques.

B.2. Specification, hardware and software development for remote control of the cutting tool, providing for automatic positioning and collision avoidance.

B.3. Development of a system for the control of the cutting action, including hardware and software, and subsequent testing of a prototype.

B.4. Development and testing of an adaptive control system, assuring optimum cutting conditions for varying cutting parameters (nature and thickness of material, cutting speed and length).

B.5. Conclusive assessment of obtained results and identification of remaining tasks.

C. Progress of Work and Obtained Results

Summary

Within the period under report a new image processing system has been developed in addition to the use of a modified underwater tv-camera for optical cutting process control (plasma and abrasive wheel cutting). Workpiece recognition is performed through the comparison of actually measured objects with pretrained reference patterns allowing the determination of object location and orientation the data of which is then supplied to the handling controller.

The hardware of the process control system for collision avoidance designed has been completed and the hardware modules were tested. In extensive tests the cutting process parameters which are of influence to the sensor signals have been ascertained. A microcomputer system has been developed for the adaptation of sensors to an abraisive cutting process in order to control the relevant process parameters. Further tests made obvious that speed control of the cutting tool should be performed in dependence on rotation speed instead of workpiece thickness measuring as originally intended.

A system has exemplarily been realized (cf. fig. 1) in order to illustrate some of the individual components and sensors in cooperation. The system is capable of performing image analysis (workpiece recognition, workpiece position, etc.) as well as the control of the handling system with an inductive sensor (distance detection, edge recognition and positioning and distance control). By an additional camera the operator has the possibility of visual process observation.

Progress and Results

1. Workpiece Recognition Underwater (B1)

The underwater dismantling of nuclear power plants needs to be remotely controlled with simultaneous optical control by underwater cameras. It is especiallly this optical control that provides problems as e.g the abrasive-wheel cutting is subjected to a great range of interferences (such as e.g. dust, etc.) so that a minimum of contrast and blurred contours of camera images must be accounted for. In addition to a tv-camera therefore a new image processing system with a solid-state camera has been developed and constructed and then adapted to workpiece recognition tasks. For image pick-up, the semi-conducter camera uses an image matrix of 128*128 pixels with 60 µm a distance by Reticon. The analysis of an image starts off with the synchronous reading and storing of grey-scale values consisting of 64 greyscales. The dynamical, temporal as well as location-dependent interferences can be either eliminated or at least reduced through the application of usual filtering methods among which the median filter represents an effective, even if time-consuming operator. In subsequent image processing, the total amount of information is to be reduced to those of relevance. This is effected by an algorithm which has been developed and implemented for edge tracking generating thin, binary contours of the object to be specified from the gradient image generated by filter operators. This so determined binary contour must then be analysed as to which features are typical of the object. In order to obtain an additional feature-describing numerical value out of the course of an object-specific radius function, a special correlation algorithm has been developed and implemented.The system is able to store up to 15 different objects in a reference memory together with the respective features which are, at the same time, availabe to be worked with. Storing of these objects as reference patterns takes place in a learning phase.

After finishing object recognition, the sensor computer supplies information about the type of object, its location and also its orientation to the handling control which then uses these information to call up the object-specific motion program for correction.

2. Specifications to Control Cutting Instruments Positioning Considering Sensor Information from the Tool Area and Application of Handling Systems or Robots (B2)

The hardware of the designed process control system for cutting has been completed and the hardware modules were tested. The sensor system for collision avoidance is based on the typical parameters such as current and voltage of the pilot arc as sensor. Extensive tests were necessary for the realization of the system to determine possible influential factors to the sensor signal. Working material, its respective thickness and edge geometry, on the one hand, and nozzle diameter, approaching rate, motion rate of the pilot arc and working intervals, on the other hand, were varied. It turned out that, on moving the arc towards the workpiece surface, different working materials and varying material thickness do not have any influence on the signal, whereas the nozzle diameter strongly affects the signal. Different working materials and varying thickness, however, affect the voltage signal in those tests where the torch was guided along the workpiece edge and so does nozzle diameter and attrition, varying distance between arc and workpiece surface.

3. Control of Cutting Action (B3)

To assure high quality of the abrasive-wheel cutting, sensors must be adapted to the cutting process in order to control the relevant process paramters such as wheel rotation speed and diameter, feed rate, material thickness and the distance between tool and workpiece. This is the reason why a microcomputer system has been developed being able to perform tasks such as the control of the cutting tool, control of sensor signals and documentation. The program is modular structured consisting of a main program for cutting process control and 10 modules covering individual tasks of coordination. This structure is advantageous in that it ensures easy readability and software maintenance as well as little programming effort in adapting to a range of different problems and hardware configurations.

Extensive tests of different types of inductive and ultrasonic sensors to check the individual performance characteristics revealed that the ultra-sensor giving impulses in the shape of transient waves and in a frequency spectrum of 0.8 MHz is the most suitable one to fulfill the demands on the integration into the cutting process. If coupled to the sensor, the handling device should have an orientation accuracy of 0.5^{o} for safe employment of the sensor.

4. Adaptive Control of Cutting Parameters (B4)

Abrasive wheel cutting tests showed that it is necessary to distinguish between down and upgrinding since this affects the dynamic and static behaviour of the cutting-off unit as well as the attrition of the cutting-off wheel. The latter is also highly influenced by the depth of penetration into the material to be cut. In order to avoid one-sided attrition of the wheel and, through a constant wheel rotational speed, to ensure continuous and safe cutting also with varying plate thicknesses, the rotational speed of the wheel should be sensor-controlled. It should be maintained through the adaptive control of forward feed rate. Feasibility tests were carried out with the help of a special program developed to control the cutting process, to determine compressed-air flow, the rotational speed of the cutting wheel and traverse path.

Despite the original intention to regulate the feed rate of the tool by continuous measuring of the workpiece thickness, the above tests revealed that it is useful to effect feedrate control in dependence on the measured rotation speed. The use of the selected ultra-sonic immersion type sensor for collision avoidance provides - integrated into the overall system - efficient protection of the control system in the cutting direction.

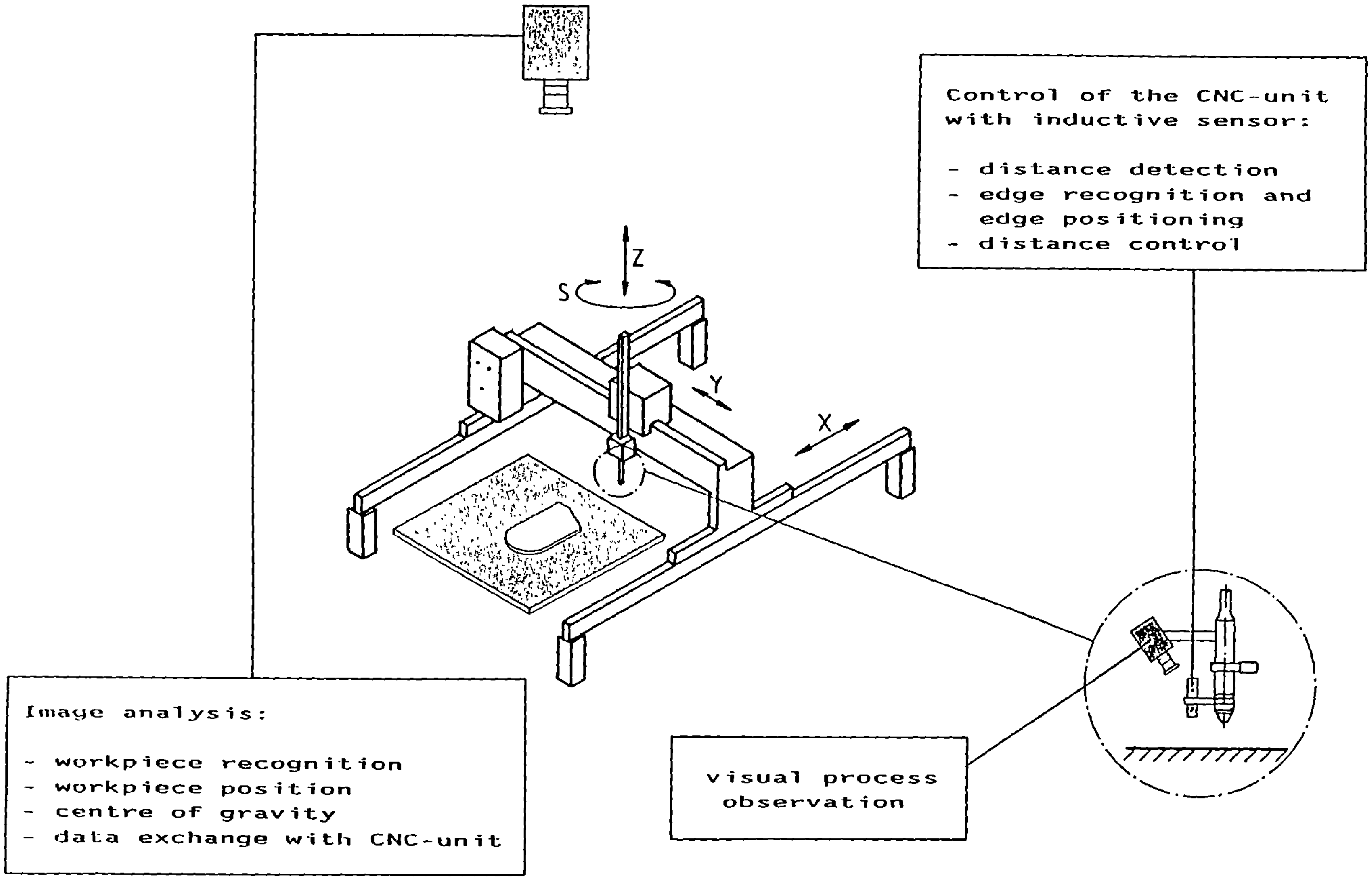


Fig. 1: Sensor-Controlled Cutting of Components

3.14 Removal of Concrete Layers from Biological Shields by Microwaves

Contractor: Building Research Establishment, Garston, United Kingdom
Contract N°: FI1D-0040
Working Period: May 1986 - March 1988
Project Leader: D. Hills

A. Objectives and Scope

The removal of the activated layer of the reinforced concrete biological shield of a nuclear reactor is an important operation in the decommissioning of nuclear power stations. The main objectives of this research are:

- to undertake a series of studies and trials in order to assess the application of microwaves in the controlled demolition of concrete biological shields of nuclear reactors and
- to undertake a feasibility design and cost study of a remotely operated prototype breaker by microwave action.

For this, former work on microwave concrete spalling will be re-assessed, and a series of laboratory trials on important parameters, such as appropriate power and frequency, useful applicators etc., will be undertaken. The results of these parametric studies will be applied to laboratory-scale tests into spalling the top 150-200 mm section of a reactor representative concrete block.

The study will result in the conclusion whether controlled removal of radioactive concrete layers by the application of microwaves will have a realistic potential for large-scale application to biological shield of nuclear power plants.

B. Work Programme

B.1. Detailed literature search on existing microwave techniques.

B.2. Theoretical analysis and computer model studies on the optimisation of power and frequency levels.

B.3. Studies on the optimisation of launching techniques for the transfer of microwaves to the concrete wall.

B.4. Theoretical analysis and computer model studies on the effect of steel reinforcement on the induction heating.

B.5. Laboratory-scale high power trials for spalling the top 150-200 mm section of a representative concrete block.

B.6. Feasibility studies for a design of a remotely operated prototype, including cost estimations.

C. Progress of Work and Obtained Results

Work in this contract has been completed, the final report is unaer publication.

3.15 Adaptation of an Existing Air-tight and Modular Workshop for Remote Operation

Contractor: Technicatome, Gif-sur-Yvette, France
Contract N°: FI1D-0041
Working Period: April 1986 - January 1988
Project Leader: B. Gasc

A. Objectives and Scope

A modular workshop for the dismantling of low-level and medium-level radioactive equipment has been developed and run successfully for some time in the La Hague Centre. It is used as an independent mobile dismantling cell receiving the equipment to be dismantled via a safety lock, with operators working inside the cell in frogman suits.

The objective of the present work is to modify the existing design of the modular workshop into a dismantling cell for high-level radioactive equipment with operators working outside by telemanipulators.

The work consists mainly in the development, fabrication and testing of following components:

- panels for specific functions such as supports for various telemanipulators and transfer locks,
- an air-tight transport cask for maintenance of important equipment and instrumentation, outside the workshop,
- an efficient system for biological protection.

B. Work Programme

B.1. Conception and preparation of premiminary designs of prototype components.
B.2. Preparation of final designs, manufacturing and commissioning testing of prototypes on the site of fabrication.
B.3. Mounting of the new components on the cell and qualification testing of the whole.
B.4. Conclusive assessment of the functioning of the newly developed components in the dismantling workshop for remote operation.

C. Progress of Work and Obtained Results

Work in this contract has been completed, the final report is under publication.

3.16 Adaptation of Abrasive Water Jet to Cutting of Radioactive Materials

Contractor: Commissariat à l'Energie Atomique, CEN Valrho, France
Contract N°: FI1D-0067
Working Period: January 1987 - December 1988
Project Leader: R. Rouvière

A. Objectives and Scope

High-pressure water cutting with and without abrasives has been currently used for some time for the manufacturing and cutting of non-radioactive materials, especially for soft materials and for concrete.

The present work aims at adapting the above technique to the dismantling of nuclear installations, taking into account the specific situation: working with radioactive materials and the need to treat the secondary waste produced during cutting of metal and concrete.

The main targets are the development of a technique for remote handling and maintenance, as well as the determination of the type and distribution of the effluents arising by the cutting of metallic and concrete samples.

This work will be executed in close co-operation with Universität Hannover (see Par. 3.17.).

B. Work Programme

B.1. Detailed definition of the test parameters and design and construction of the test facility.
B.2. Tests on standard equipment and operator formation.
B.3. Design and fabrication of components for remote operation.
B.4. Design and fabrication of components for remote maintenance.
B.5. Testing of new components and execution of remote handling tests with the manipulator.
B.6. Measurements of effluents produced during the cutting of different materials in the air with optimised parameters.

C. Progress of Work and Obtained Results

Summary

All the tests scheduled under the contract were completed in 1988. The principal results include the following:

- The high pressure water jet cutting process using abrasive additives allowed satisfactory cutting rates to be obtained in automatic operation
- Major difficulties were encountered for remote implementation of the process
- Significant volumes of liquid and solid wastes were produced, and waste recovery proved difficult
- It was difficult if not impossible to protect the environment behind the cutting zone
- Only small amounts of aerosols or water vapor were entrained in the ventilation system
- Further development for automated cutting in remote position or with sufficient visibility is needed.

Progress and Results

1. Preliminary Examination of Procedures and Construction of Test Facility (Phase B1)

Work was completed in 1987.

2. Basic Equipment Testing and Operator Training (Phase B2)

Automatic cutting tests on mild and stainless steel (500 × 500 mm specimens with thicknesses of 10, 30, 60 and 100 mm) and reinforced concrete (500 × 500 mm specimens with thicknesses of 100, 200 and 300 mm) were continued and finished in 1988.

The ADMAC machine operated satisfactorily for about 100 hours at 2400 bars. It was controlled so as to cut through the specimen in each case, while determining the maximum compatible cutting rate.

The total cutting lengths are indicated in Table I, showing that the test conditions were representative of industrial operation. All significant parameters were recorded and some were optimized.

The following conclusions may be drawn from this work:

- Garnet abrasives were the most cost-effective of those tested.
- The optimum abrasive/water mass ratio is between 15 and 25%.
- The spray nozzle/specimen distance had little or no effect on the steel cutting rate for distances between 1 and 20 mm, although this may not be true for concrete. In every case, however, the width of the cut increased with the distance.
- Cutting rates obtained are indicated in Table II: no differences were observed between mild steel and stainless steel.

3. Design and Manufacture of Components Suitable for Remote Operation (Phase B3)

Precoiled high pressure hoses rated at 2400 bars were specially manufactured for remote operation tests. The results were fully satisfactory for both flexibility and reliability.

4. Design and Manufacture of Components for Remote Maintenance (Phase B4)

The maintenance station designed and built by the CEA operated perfectly by remote manipulation: it was used to replace the cutting device in less than 10 minutes using only the MA 23M telemanipulator.

5. Remote Cutting and Dismantling Tests with a Telemanipulator (Phase B5)

These tests were conducted using the MA 23M electronic master-slave telemanipulator manufactured by *La Calhène*, see figure 1.

It was practically impossible to operate correctly because of poor visibility resulting from the heavy mist surrounding the cutting device (mist is more important when cutting is not through, which is frequent in remote operation).

Reaction forces due to the high pressure jet and to the stiffness of the HP hose did not raise any significant problems, however, and always remained below the normal operating limit of 20 daN for the MA 23M (approx 10 daN in most cases).

The remote manipulator tests also showed that the environment behind the specimen could not be protected: the bottom of the tank supporting the specimen was perforated despite the distance of 1 meter from the cutting nozzle and despite the presence of 50 cm of water in the tank.

The remote maintenance tests (cf §4) were completed without incident.

6. <u>Secondary Waste Generation</u> (Phase B6)

Test measurements showed that:

- The cell ventilation system entrains 10^{-6} to 10^{-5} of the total solid mass and 10^{-2} of the water mass in the form of vapor and spray droplets.
- From 2×10^{-3} to 1.3×10^{-2} of the total solid mass was deposited on the cell walls.
- The purification systems used (baffle type demister and electrostatic filter) were not very effective in retaining the liquid mass entrained by the ventilation system, indicating that the water is mainly entrained in vapor form.
- The solid mass deposited after cutting had a mean mass diameter of less than 80 μm and at least 85% of the abrasive particles were fractured.

Table I - Total Cut Length during Tests (in mm)

Material	Specimen Thickness (mm)						
	10	30	60	100	200	250	300
Mild steel	1386	3487	6181	1134	-	-	-
Stainless steel Z2 CN18-10	4780	4404	1169	3498	-	-	-
Reinforced concrete	-	-	-	975	2709	423	288

Table II - Mean Cutting Speed on Test Facility (in mm/min)

Material	Specimen Thickness (mm)						
	10	30	60	100	200	250	300
Mild steel or Stainless steel	225-250	73-88	37-43	20-30			
Reinforced concrete	-	-	-	100-125	12-80*	-	11

*depending on reinforcement and on cutting head/specimen distance

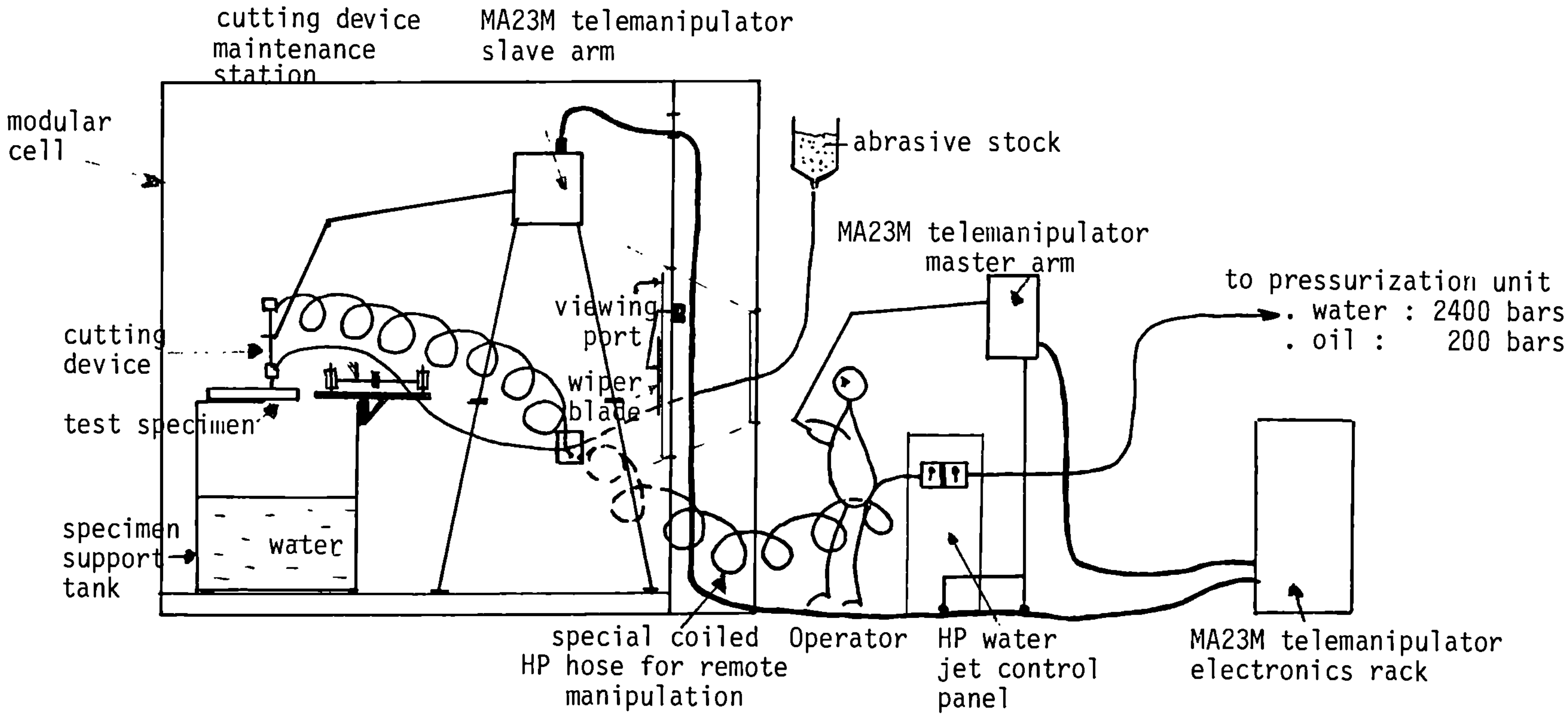


Figure 1 - Overall Layout of Remote Cutting Test Facility

3.17 Development of Abrasive Water Jet for Submerged Cutting of Steel

Contractor: Universität Hannover, Hannover, Germany
Contract N°: FI1D-0069
Working Period: July 1987 - March 1989
Project Leader: H. Louis

A. Objectives and Scope

The use of high-pressure abrasive water-jet cutting in air for non-radioactive concrete or other materials has become a current procedure.

The application of the above technique to highly radioactive metallic components under the protection of a water layer could be very attractive for the dismantling of nuclear installations, e.g. cutting of core internals.

The present work is aimed at developing and optimising this technology by parametric experimental studies, aimed at achieving high cutting performance with minimal generation of secondary waste. The parameters to be considered are such as water pressure, water depth, mass flow of water and abrasives, type of abrasives. The wear of the jet outlet nozzle and the energy loss of the jet in the surrounding water are considered as important issues.

This work will be executed in close co-operation with Commissariat à l'Energie Atomique (see Par. 3.16.).

B. Work Programme

B.1. Preparation of the test facility for underwater cutting.
B.2. Development and fabrication of the abrasive cutting head.
B.3. Studies on optimising the working distance and minimising flow rates.
B.4. Studies on the influence of the water depth on jet formation and cutting efficiency.
B.5. Studies on the influence of abrasive material on jet formation and cutting efficiency.

C. Progress of Work and Obtained Results

Summary

The cutting tests have shown the possibility to cut austenitic and ferritic steel under water with a small amount of water and abrasives.

The working distance of the abrasive water jet under water was optimised by adapting an air mantle nozzle at the cutting head. Additional investigations showed the influence of abrasive flow rate, pressure and traverse rate on depth of kerf. Investigations carried out in a pressure chamber have shown that the cutting head also in case of a higher ambient pressure produces a sufficient suction pressure. Cutting tests in a pressure chamber pointed out the influence of the surrounding medium and the ambient pressure on cutting efficiency.

Progress and Results

1. Development and fabrication of the abrasive cutting head (B.2.)

In case of submerged cutting the energy loss because of friction between the abrasive jet and the surrounding water is much higher than in air.

This is the main reason for the short working distance in case of submerged application. Figure 1 shows the differences between cutting in air and under water regarding the cutting depth. In figure 1 also the results of experiments with an air mantle for increasing the working distance are given.

To carry out these tests an air mantle nozzle was designed and tested. The setup is shown in figure 2. For cutting tests a tungsten carbide nozzle with an inner diameter of 3 mm was used. The air mantle was build up by compressed air. The use of such a nozzle causes quite good cutting results also in case of higher working distances (fig. 1).

2. Studies on optimising the working distance and minimising flow rates (B.3.)

2.1 Effect of abrasive flow rate on depth of kerf

Cutting contaminated material creates problems for the subsequent waste treatment. Not only the particles of the cut structure but also the cutting material like water and abrasives has to be treated. This fact leads to the requirement that the amounts of water and abrasives have to be optimized to reach a good cutting effiency by producing minimal secondary waste.

Figure 3 (left side) gives the results of cutting tests carried out on ferritic and austenitic steel. Both materials show the same behaviour: The depth of kerf reaches a maximum level by using a special flow rate of abrasives; increasing this flow rate causes a reduction in depth of kerf. The optimum value of abrasive flow is about 6 g/s in case of using 2400 bar and a nozzle with 0.25 mm diameter.

The results of cut-through-tests are given on the right side of figure 3. One has to notice that when cutting through the optimal abrasive flow rate is higher than in case of notching the samples.

2.2 Effects of pressure and traverse rate on depth of cut

Some investigations were done to indicate the influence of water pressure and traverse rate on the cutting efficiency. The results are given in figure 4.

Increasing the pressure up to 300 MPa the depth of kerf grows linear with the pressure. This fact is interesting for waste reduction: Doubling the pressure means doubling the reachable depth of kerf, too, but only multiplying the water flow rate by 1.4 (water flow rate is proportional to square root of pressure).

Doubling the traverse rate causes halving in depth of cut. This effect is also existing at very low traverse rates: Cutting ferritic steel with a traverse rate of 10 mm/min enables a depth of kerf of 75mm.

3. Studies on the influence of the water depth on jet formation and cutting efficiency (B.4.)

For remote-controlled operation under water the influence of the water depth on the reliability of the abrasive mixing process has to be tested. A pressure chamber was built up to realize a higher ambient pressure. In the test program the suction pressure produced by the water jet was measured for different ambient pressures.

These results showed that also in case of a higher ambient pressure it is possible to produce a low absolute pressure inside the mixing head when using the right diameter of focussing nozzle /1/.

Cutting tests carried out in a pressure chamber pointed out the influence of the surrounding medium on the reachable depth of kerf (figure 5). Tests up to a ambient pressure of 5 bar in air showed no influence on the kerfing depth, but tests in water pointed out a decrease in depth of kerf when increasing the ambient pressure.

But in case of decommissioning the water depth is less than 15 m. So the influence of the ambient pressure is not very important.

4. Studies on the influence of abrasive material on jet formation and cutting efficiency (B.5.)

Cutting tests with different abrasives pointed out that garnet and silica sand, too, give the best cutting results regarding depth of kerf on one hand and wear of the focussing nozzle on the other hand (figure 6). Spherical particles are not useful because there are no sharp edges to cut material.

References

/1/ Haferkamp, H., Louis, H. and Meier, G., Cutting of contaminated material by abrasive water jet under the protection of water shield, 9th Intern. Symposium on Jet Cutting Technology, Sendai, Japan, 4-6 October 1988. Proceedings pp. 271-287.

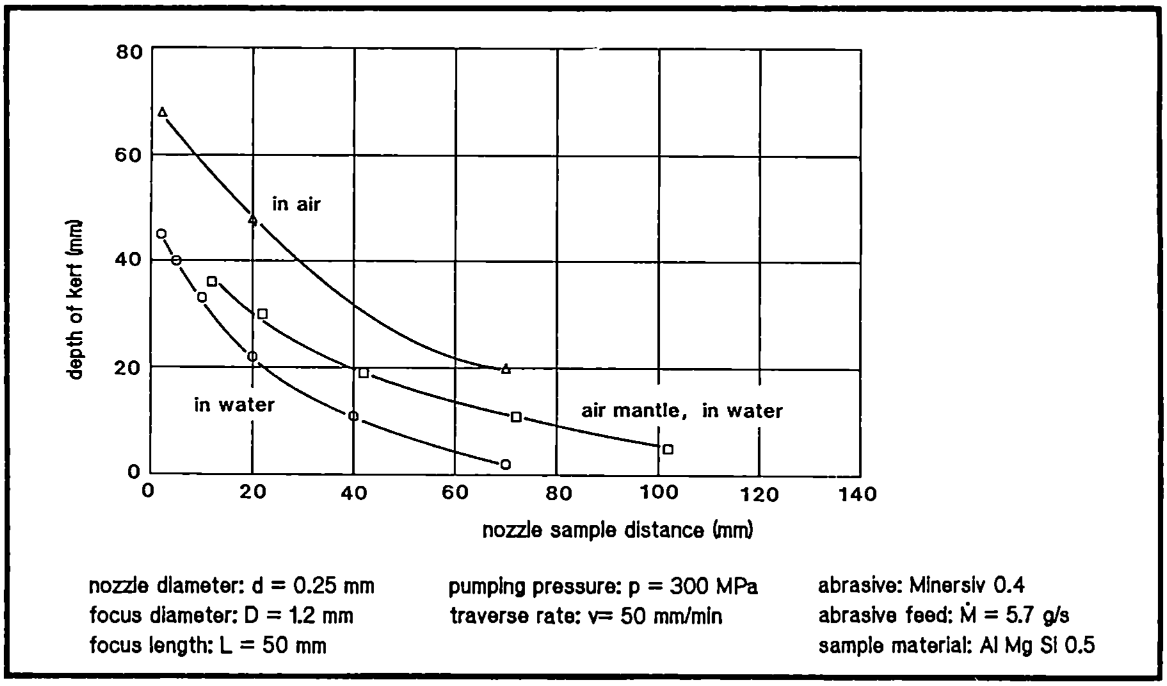


Figure 1:
Effect of surrounding medium and working distance on depth of kerf

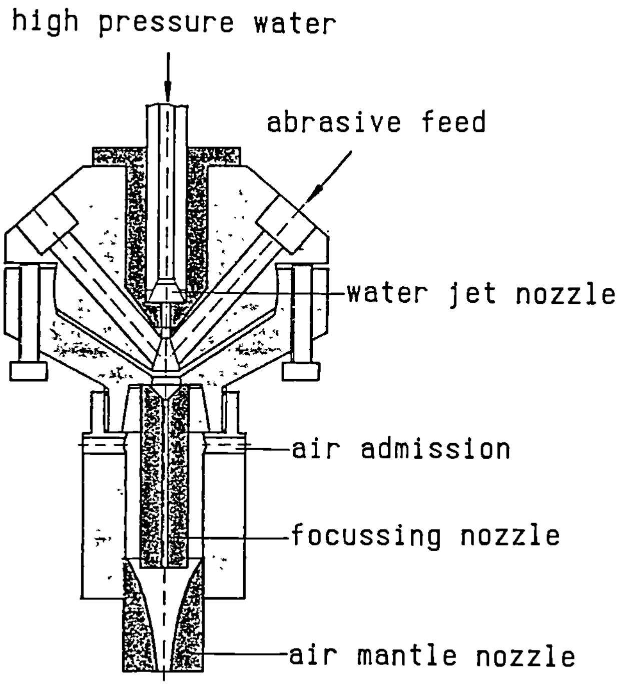
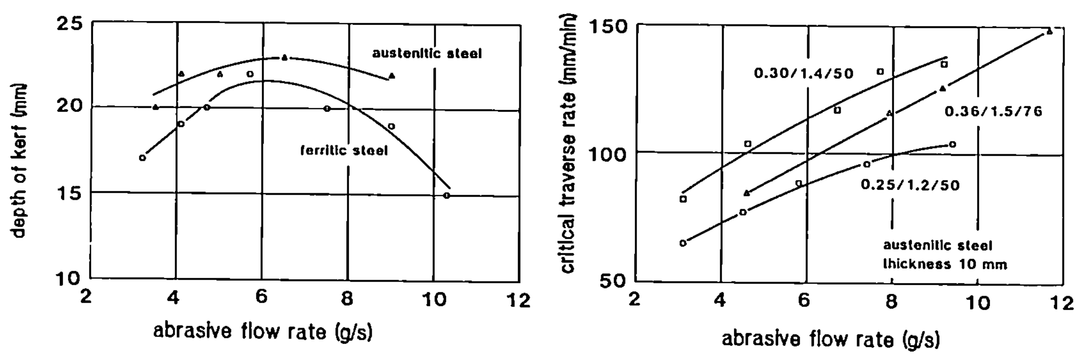


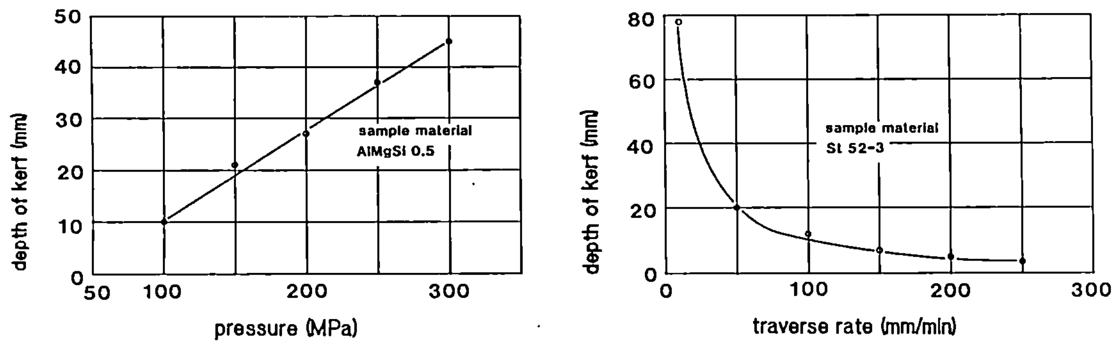
Figure 2:
Abrasive cutting head with air mantle nozzle



nozzle diameter: d = 0.25 mm
focus diameter: D = 1.2 mm
focus length: L = 50 mm
pumping pressure: p = 240 MPa
nozzle sample distance: s = 2 mm
abrasive: Minersiv 0.4
surrounding medium: water

nozzle diam./focus diam/focus length:
see diagram
nozzle sample distance: s = 5 mm
abrasive: Grenal
surrounding medium: air

Figure 3:
Effect of abrasive flow rate on depth of kerf respectively critical traverse rate



Standard parameters
nozzle diameter: d = 0.25 mm
focus diameter: D = 1.2 mm
focus length: L = 50 mm

pumping pressure: p = 300 MPa
nozzle–sample distance: s = 2 mm
traverse rate: v = 50 mm/min

abrasive: Minersiv 0.4 mm
abrasive feed: $\dot{M}_{abr}$ = 5 g/s
surrounding medium: water

Figure 4:
Effect of pressure and traverse rate on depth of cut
(L: length of focussing nozzle / D: diameter of focussing nozzle)

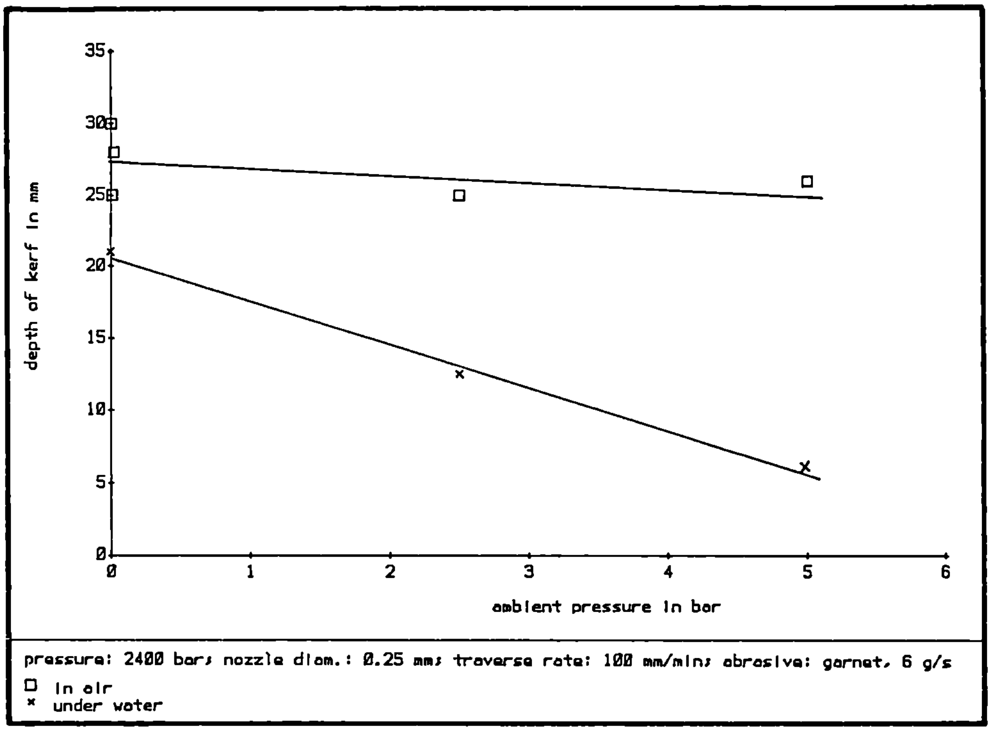


Figure 5:
Effect of ambient pressure on depth of kerf

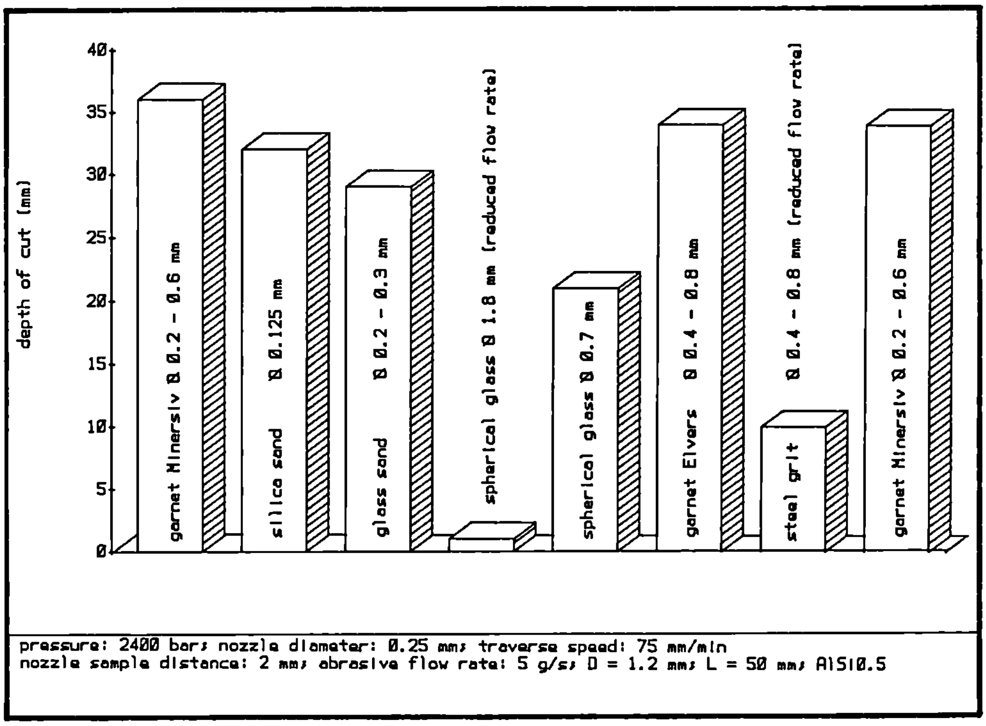


Figure 6:
Effect of abrasive material on depth of kerf

3.18 Analysis of Results Obtained with Different Cutting Techniques and Associated Filtration Systems

Contractors: Commissariat à l'Energie Atomique, CEN Saclay, France
Universität Hannover, Hannover, Germany
Contract N°: FI1D-0070 and FI1D-0071
Working Period: April 1988 - June 1989
Project Leaders: G. Pilot, F.W. Bach

A. Objectives and Scope

The present joint study by Commissariat à l'Energie Atomique, CEN Saclay (CEA) and Universität Hannover (UH) is intended to analyse the results generated in a number of research contracts concerned with cutting tests in the air and under water, under consideration of the prevailing working conditions. The same target is fixed for the associated filtration systems. The analysis should result in a large data base, making common use of all data by giving comparable information for practical dismantling operations. The study should be enlarged to a broader scope including also recently obtained results outside the present research programme, considering also supplementary cutting tools and filtration systems not covered by the present programme.

The analysis will be done in close cooperation between both contractors by work-sharing.

B. Work Programme

B.1. Preparation of a complete list of important working parameters to be considered for a useful comparison of experimental data obtained with cutting tests on metallic components, as concerns cutting performance, cutting effluent generation and performance of associated filtration systems (CEA+UH).

B.2. Analysis of generated cutting effluents in air and under water obtained with the cutting tools considered in chapter B.3. (CEA + UH).

B.3. Analysis and comparison of data on cutting performance. This analysis should also include results obtained outside the present programme (UH).

B.4. Analysis and comparison of data on various prefilters for protection of HEPA filters (CEA).

B.5. Analysis and comparison of data on water-filter systems (UH).

B.6. Preparation of a conclusive assessment on the treated results with recommendations for applications to large-scale decommissioning (CEA + UH).

C. Progress of work and obtained results

Summary

In order to compare the results of cutting tests with different cutting tools e.g. cutting performance, cutting effluent generation and performance of associated filtration systems several questionnaires have been prepared and sent to the contractors of the present programme, working with cutting tools or filter equipment.

Progress and results

Preparation of a complete list of important working parameters (B.1)

In the present research programme of the European Community there are nine contractors working with different cutting tools. The cutting tools used in these contracts are:

- plasma arc torch
- oxyacetylen torch, with and without powder
- oxypropane torch, with and without powder
- laser beam
- mechanical grinder
- high pressure water jet with abrasives

In order to compare the results of the cutting tests with these tools e.g. cutting performance, cutting effluent generation and performance of the associated filtration systems it is necessary to get very detailed informations about the test conditions.
For each cutting tool a questionnaire has been prepared and sent to the concerned contractor.
Table I gives an example for such a data sheet for the plasma arc cutting. The first part is made to get some informations about the test conditions in general (e.g. cutting at atmosphere or underwater). The second part asks for information about the torch or the cutting tool (e.g. the manufacturer and the type of the torch).
The last part is reserved for the cutting parameters and cutting results of certain tests. The contractors are asked to give their comments, remarks or any other interesting information about the used tool.

Most of the used cutting tools are working with high energy concentrations to cut metallic structures. Therefore, depending on the cutting tool, there is a large amount of cutting effluents going to atmosphere or staying in the water.
To get information about the cutting effluents, the measurement techniques for these effluents and the used filtration systems some other questionnaires have been prepared and sent to the contractors too.
Table II presents a data sheet as an example for the wanted information about the aerosol measurement techniques. The main part of this sheet consists of questions about the used aerosol measuring equipment. It is very important to have information on the physical working principle because different definitions for aerosols are used e.g. for the aerosol particle diameter.

Table I: Data sheet for plasma arc cutting

PLASMA ARC CUTTING

TEST CONDITIONS :

( ) in atmosphere   ( ) under water   depth ....[mm]
Water : ( ) deionized ( ) other   pH ....
( ) normal drinking water   conductivity ......[µS]

Material : ....   material thickness : ....[mm]
Cutting position : ( ) gravity position   ( ) vertical up
( ) horizontal   ( ) vertical down
( ) overhead

COMMENTS :

TORCH :

Specification of the torch : ....
Manufacturer of the torch : ....
Torch type : ( ) tipped electrode   ( ) flat electrode
Waterinjection : ( ) yes   ( ) no
Max. current of the torch : ....[A]
Ignition of the pilot arc : ( ) HF   ( ) other

COMMENTS :

CUTTING PARAMETERS :

Current : ....[A]   Voltage : ....[V]

Plasmagas :	( ) argon	.........[l/min]	at	.........[bar]
	( ) nitrogen	.........[l/min]	at	.........[bar]
	( ) hydrogen	.........[l/min]	at	.........[bar]
	( ) air	.........[l/min]	at	.........[bar]
	( ) other	.........[l/min]	at	.........[bar]

Diameter of the nozzle: ....[mm]
Nozzle life: ....[m cutting length]
Distance nozzle-workpiece: ....[mm]
Width of kerf: top ....[mm]   bottom ....[mm]
Cutting speed: ....[mm/min]

COMMENTS :

Table II: Data sheet about the measurement of cutting effluents in atmosphere

MEASUREMENT OF CUTTING EFFLUENTS IN ATMOSPHERE

CUTTING PROCESS: ( ) Plasma ( ) Water Jet
( ) Laser ( ) Flame Cutting

( ) in atmosphere | ( ) under water

AEROSOL MEASURING EQUIPMENT: ( ) on line ( ) off line
type: .... made by: ....
working principle
(e.g. optical counter): ....
nominal measuring range: .... [µm diameter]
This equipment gives:
( ) a size distribution ( ) the aerodynamic diameter
( ) a mass distribution ( ) the electrostatic diameter
( ) a particle concentration ( ) ....

COMMENTS:

AEROSOL CHARACTERIZATION:
Concentration in the exhaust duct: ..... [mg/m³]
at a flow rate of: ..... [m³/h]
Is a dilution necessary for measuring?
( ) no ( ) yes, a dilution of 1 : .... is used
In case of a size distribution:
mean particle diameter: .... [µm], deviation of ......
Please give the graphs of the distribution on an extra sheet!
Density of the particles: ..... [g/cm³] ( ) unknown
Aerosol emission: .....[g/m](mass/m length of cut)

GASEOUS EFFLUENTS: ( ) on line ( ) off line
H_2: .... [ppm] O_3: .... [ppm] ...: .... [ppm]
NO: .... [ppm] NO_2: .... [ppm] NO_x: .... [ppm]

COMMENTS:

3.19 Measurement of Effluents from Plasma Arc and Laser Cutting

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FI1D-0072
Working Period: April 1988 - June 1989
Project Leader: G. Pilot

A. Objectives and Scope

In the frame of comparative cutting tests with plasma arc under water (UW) and laser beam in air, being carried out by several contractors in the present Community programme, the complex measurements of the arising cutting effluents will be performed with the same measuring system and by the same team, thus avoiding discrepancies from the system and giving a better basis for comparison of obtained results.

The contractor will undertake, on the site of concerned contractors, a series of well-defined measurements on cutting effluents and evaluate the obtained results.

B. Work Programme

B.1. Measurements/evaluations of effluents during standard UW plasma arc cutting tests at Commissariat à l'Energie Atomique - CEN Cadarache.

B.2. Measurements/evaluations of effluents during standard UW plasma arc cutting tests at Heriot Watt University, Edinburgh.

B.3. Measurements/evaluations of effluents during standard laser beam cutting tests at Commissariat à l'Energie Atomique - CEN Saclay.

B.4. Measurements/evaluations of effluents during standard cutting tests with plasma arc and laser beam at Universität Hannover.

B.5. Final evaluation in cooperation with the associated contractors.

C. Progress of Work and Obtained Results

Summary

The detailed objectives of the standard cutting tests are presented. The same measuring rig that has been used on the site of the concerned contractors is described. The precise working conditions for plasma torch cutting and laser beam cutting are given.

Progress and Results

1. Introduction

The detailed objectives of the standard cutting tests are as follows:

. Gaseous effluent measurements
 - aerosols: concentration
 size distribution
 chemical analysis
 - gas: nature
 concentration

with the same measurement rig (in order to avoid uncertainties due to different measuring apparatus).

. Assessment of secondary emissions balance in gaseous and solid form (sedimented drosses, attached slags, suspended particles, aerosols).

. Assessment of installation and scale effects (laboratory or semi-industrial level).

. Comparison of gaseous effluents between two cutting tools.

The working parameters for plasma torch and laser beam have been chosen by taking into account the constraints linked to every installation and the possibility of a correct recording of values to measure after agreement of all contractors implied (CEA/CEN CADARACHE, Universität Hannover and Heriot Watt University/Edinburgh for plasma torch cutting; CEA/CEN SACLAY, Universität Hannover and FIAT CIEI-ENEA for laser beam cutting).

2. Measurement rig used

The following measurement rig given below has been used on the site of concerned contractors:

- Solid effluents measurement
 . for the aerosol mass concentration: sampling filters
 . for the aerosol size distribution: inertial and diffusional spectrometer (SDI 2001)
 . for sedimented dross, attached slag and particles in suspension in water: weighting
 . chemical analyses of elemental composition of particles in air and in water by inductive coupled plasma (ICP).
- Gas measurement
 . H_2 concentration: thermal conductivity analysis
 . NO/NO_x concentration: chemiluminescence analysis
 . O_3 concentration: ultraviolet radiation absorption

3. Arc plasma cutting

3.1. Working conditions

After consideration of common feasibilities, the three concerned contractors have retained following working conditions:

- Cutting samples:
 material: stainless steel 304

dimensions of the plates: thickness: 20 and 40 mm
width: 50 mm (100 mm for Edinburgh)
overall length: 1000 mm (400 mm for Edinburgh)
cutting length: 900 mm

preparation: cleaning with alcohol

All samples have been prepared from the same origin.

- Number of tests:

The number of tests, according working conditions and places are indicated in Table I.

For comparison, some additional tests with different cutting speeds were decided afterwards.

Cutting parameters:

- nozzle diameter: 2.5 mm
- tipped electrode, HF injection, no water injection
- plasma gas: mixture 60% argon
  40% nitrogen
- mass flow of gas mixture: 50 l/min
- intensity of electric current: 200 A for 20 mm
  250 A for 40 mm
- cutting position: gravity position
- torch-sample distance: 8 mm
- cutting speed: 60 mm/min
- water depth: 500 mm
- water quality: deionized water
- PH of the water: between 6.5 and 7.5
- conductivity of the water: between 5 and 10 µS
- ventilation flow in cutting cell: 300 m^3/h (80 m^3/h for Edinburgh).

3.2. Execution of cutting tests (B1, B2, B4)

All the experimental work on the site of concerned contractors has been done in 1988 except for the additionnal tests with higher cutting speeds (Table I) which will take place in March 1989 at Cadarache.

Some chemical analyses, necessary for the interpretation of the results, are still in progress.

4. Laser beam cutting

4.1. Working conditions

The three concerned contractors (FIAT CIEI-ENEA will execute his own measurements) have retained the following working conditions:

Cutting samples:

- material: stainless steel 304
- dimensions of the plates: thickness : 10 mm and
  20 mm (only CEA/DEMT/SACLAY)
  width: 50 mm
  overall length: 1000 mm
  cutting length: 900 mm
- preparation: cleaning with alcohol

All samples have been prepared from the same origin.

Numbers of tests:

- 2 tests with 10 mm thickness in air
- 2 tests with 20 mm thickness in air (only CEA/DEMT/SACLAY)

Common cutting parameters:

- cutting speed: 150 mm/min
- power on plate: 1 kW

- assistant gas: O_2
- oxygen pressure: 1.5 atm
- cutting position: gravity position
- nozzle diameter: 1.3 mm.

4.2. Execution of cutting tests (B3, B4)

As for plasma cutting, experimental work has been done in 1988 and chemical analyses are still in progress. For all standard cutting tests with plasma arc cutting and laser beam, more than 350 samples (filters, solutions) have be to analysed.

Table I: Number of tests according working conditions and places

Conditions / Place	e = 40 h = 0.5 V = 60	e = 20 h = 0.5 V = 60	e = 40 h = 1 V = 60	e = 20 in air V = 60	e = 40 h = 0.5 V = 180	e = 40 h = 0.5 V = 300
Cadarache	3	3	-	2	2	2
Hannover	3	3	-	2	2	2
Edinburgh	3	3	2	-	-	-

e = thickness of the plate (in mm)

h = water depth (in m)

V = cutting speed (in mm/min)

4. PROJECT N°4: TREATMENT OF SPECIFIC WASTE MATERIALS: STEEL, CONCRETE AND GRAPHITE

A. Objective

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

B. Research performed under the 1979-83 programme

The following research work has been performed:

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

C. 1984-88 programme

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

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

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

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

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

D. Programme implementation

Eight research contracts relating to Project N°4 were executed in 1988, of which two were completed.

4.1. Melting/Refining of Contaminated Steel Scrap from Decommissioning

Contractor: British Steel Corporation, Moorgate Rotherham, United Kingdom
Contract N°: FI1D-0015
Working Period: January 1985 - June 1989
Project Leader: D.S. Harvey

A. Objectives and Scope

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

In a supplementary agreement concluded in 1988, the initial work programme was extended to the development of a system for monitoring of the radioactive scrap at arrival and during storage and treatment at the melting facility, see additional working package B.8.

B. Work Programme

B.1. Tests on the 5t electric arc furnace with appropriate nuclear scrap and simulated contamination.

B.2. Investigation of caesium retention in 10t induction furnaces using normal acid slag and low level radiotracer.

B.3. Melting of activated/contaminated steel waste in a 6t experimental Basic Oxygen Furnace (BOF) in order to examine the Co-retention in slag when Co is present as surface contaminant.

B.4. Pre-furnace assessment of the contamination of steel waste by monitoring.

B.5. Investigation of the slag/metal chemistry to identify specific radionuclides (Nb-94, Ni-63, Sr-90, Sb-124, Eu-154 and Am-241).

B.6. Investigation of the transfer of radioactivities to furnace and refractories with particular view to the concentration effect of the nuclides.

B.7. Evaluation of radiation exposure (individual and collective) of involved persons and of radioactive emissions to the environment for long-term operations; cost/benefit optimisation for re-cycling and disposal based on results obtained.

B.8. Conception of a system for detection of radioactive sources in steel plant scrap materials.

B.8.1. Examination of detection strategies by computer modelling.

B.8.2. Definition of radioactivity levels to be detected, account being taken of potential health hazards, and calculation of the conditions required to detect these levels.

B.8.3. Assessment of various detection strategies.

B.8.4. Cost estimation.

C. Progress of Work and Obtained Results

Summary

There have been a number of experiments to try and define the behaviour of caesium during melting of steel in the induction furnace. The results show that retention of caesium is greater in acidic slags than in basic slags as expected. It was found however, that the results were very variable, and the reason is believed to be the difficulty in obtaining uniform melting of slag in induction furnaces.

A study of the retention of cobalt, caesium and europium in refractory bricks showed that only caesium was mobile at steelmaking temperatures. It was mainly volatilised from basic refractories but it was mainly retained in acidic refractories.

Detection of radioactive materials entering steelworks is being studied using a detector at the road weighbridge. The equipment can detect 37 M Bq of cobalt 60 in a typical load of scrap. The limit of sensitivity of the system will be set by the variability between the different loads of scrap which have to be examined, rather than by the sensitivity of the detector.

Progress and Results

1. Investigation of Caesium Retention in an Induction Furnace (B.2)

A number of melts were performed to try and define the behaviour of caesium during melting of steel in an induction furnace. The retention of the caesium was found to be approximately 70% of that added when an acidic slag was used. The degree of retention did not markedly decrease with time and it was not dependant on the temperature of the steel.

A chemically neutral slag achieved less consistent retention of caesium. There were indications that loss of caesium from the slag was continuous and that at higher temperatures of the steel the loss was greater. Chemically basic slags yielded very erratic retention of caesium and the loss of caesium from the slag was continuous.

The results were generally in accordance with the known chemical properties of caesium.

The inconsistencies in the results are thought to have arisen from the inhomogenous nature of the slag. In induction furnaces the slag is not always fully liquid and can have chemical and thermal inhomogeneities.

Experiments in the induction furnace with sodium silicate suggest that under favourable conditions this material could retain up to 80% of the caesium released from the steel. Sodium silicate has a low melting point and forms a complete cover over the molten steel so preventing escape of caesium. It is also a low cost material and the slag produced is chemically stable.

2. Interactions with Steelmaking Refractories (B.6)

There have been experiments with chemically basic (MgO) and chemically acidic (Al_2O_3 44% SiO_2 51%) refractory bricks. Cavities in the bricks were filled with powdered material of the same composition which had been impregnated with either cobalt 60, caesium 134, or europium 154. The bricks were then heated at 1600°C for several hours, cooled and sectioned. Autoradiography of the sections showed that cobalt 60 and europium 154 had not been mobile within the brick, nor had they been volatilised. Caesium 134 had been mainly (75%) volatilised from the magnesia brick but very little (6%) had been volatilised from the acidic brick. It was concluded that retention of radioactivity within steelmaking refractories was likely to be confined to a thin layer on the surface of the refractories.

3. Workplace Acceptance and Benefit v Detriment (B.7)

Discussions on the large scale melting of contaminated scrap continued but eventually the proposal was rejected by local steelworks. There is now no expectation that a melt will take place and the possibility is not being pursued.

4. Detection of Radioactive Sources in Steelplant Raw Materials (B.8)

4.1 Definition of Radioactivity Levels to be Detected (B.8.2)

The levels of radioactivity which need to be detected can be defined most easily by reference to the requirement that the products of the steelmaking process should not exceed an activity of 0.37 Bq/g. In practical terms the acceptable input of metal-seeking activity such as Co60 would be in the region of 50 M Bq. For slag seeking activity such as Eu 154 it would be 5 M Bq and for volatile activity such as Cs137 it would be in the range 0.5 to 5 M Bq.

4.2 Assessment of Various Detection Strategies (B.8.3)

A survey of the steelplant has shown that the 4 areas suitable for detection are the road and rail weighbridges, the magnet cranes and the basket filling areas. The first two areas would give poorer sensitivity of detection, but they have the advantages that they will monitor the scrap at its point of entry to the steelworks and that the development of monitors presents less technical difficulty.

An experimental monitor has been developed and used on the steelworks. It has been able to detect separately 37 M Bq of Co60 and 185 M Bq of Cs137 deliberately concealed in a typical load of scrap.

A prototype unit suitable for continuous use on the steelworks is now being developed.

The work has shown that sensitivity of detection will be limited by the variability between loads of scrap delivered to the steelworks. Under these circumstances there is no benefit in developing a monitor of extreme sensitivity. There is benefit in using several detectors in an array, and this possibility will be examined.

4.2. Melting of Radioactive Metal Scrap from Nuclear Installations

Contractor: Siempelkamp Giesserei, Krefeld, Germany
Contract N°: FI1D-0016
Working Period: November 1984 - June 1989
Project Leader: H. Deipenau

A. Objectives and Scope

This research is based on the results and experience of work carried out at Siempelkamp in the framework of the first five-year (1979-83) programme (Ref.: EUR 10021). The preceding research work proved that it is possible to melt down contaminated scrap by means of a modified industrial furnace device in compliance with the legal limits and regulations.

This research work, therefore, aims mainly at the behaviour of radionuclides during the melting procedure, with regard to various material qualities and the harmless recycling of melted-down metal parts coming from refurbishing and decommissioning of nuclear installations.

Through a supplementary agreement concluded in 1987, the initial work programme is extended by items B.7. to B.10.

B. Work Programme

B.1. Planning and design of the melt device taking into account an existing furnace.
B.2. Construction of the needed melt device components.
B.3. Melt work using as scrap contaminated carbon steel, stainless steel and its mixture.
B.4. Evaluation of melt results.
B.5. Technical, economical and radiological consequences.
B.6. Extrapolation to large nuclear power plants and comparison with alternative modes with a view to the economical and environmental aspects.
B.7. Melting of contaminated galvanised sheet material.
B.8. Melting of non-ferrous metal (e.g. copper and brass) to investigate the behaviour of relevant radionuclides (e.g. Co-60, Cs-137) during the melting process.
B.9. Investigation on adding radioactive carbon to the steel melt process to obtain cast iron of suitable quality for e.g. disposal containers.
B.10 Investigation on the long-term behaviour of the furnace liner, the charging device and the filter system after melting of about 500 t of contaminated steel waste (over two years) with particular view to activity concentration in the different parts of the melting plant.

C. Progress of Work and Obtained Results

Summary

The long-term behaviour of contamination in the working area and the activity contents in the air of the melting shop were analysed. In the point of view of radiation protection the measurements of surface contamination near the furnace are necessary to make the decision for decontamination. Within the programme "Melting of non-ferrous metals" laboratory investigations were carried out to ascertain the chemical composition of the materials with regard to their behaviour at melting. Melting tests with the insert of different kinds of carbon issued from filters and of graphite reflector blocks were carried out. The contents of non-admissible elements in the mould were measured.

Progress and Results

1. Radiological long-term behaviour of the melting plant (B.10.)

In the period of investigations in 1987 and 1988 the melted scrap reached a value of 575 Mg. The scrap was melted in 14 campaigns.
The specific activity of the melted scrap in 1988 lay in the range from 1 Bq/g to 8 Bq/g. These values are lying in the area of former experiences.

1.1 Melting of scrap with special nuclides

One melting campaign was of special interest.
A mass of 8 Mg carbon steel with contamination of Sr 90/Y 90.
The surface contamination was distributed to the following nuclides:

Sr 90/Y 90	97 %
Cs 137	2 %
Co 60	1 %
Eu 154 less than	0.5 %

The main value of the activity was lower than 3 Bq/cm^2. The distribution of the nuclides after melting is listed in table 1.
The main activity was found in the dust of the bag filter and the cyclone. The current investigations by KfA, Jülich, showed that Sr 90/Y 90 were condensed on the inner walls of the filter pipes. The results will be presented later on.

1.2 Investigation of the environmental exposure by dust depositions near the melting furnace

After normal operation of the melting furnace without radioactive scrap the surrounding area is cleaned once a week or the area is cleaned directly after melting of radioactive scrap. The measurements of the surface contamination on the roof of the furnace-control-board showed activity levels from 0.7 to 3.4 Bq/cm^2. The limit of the admissible surface contamination in a surveillance area in Germany lies at 3.7 Bq/cm^2. Thus the measurement of the surface contamination is a necessary control system to make the decision for decontamination. A sum-up of activity over several melting days takes place.

1.3 Investigation of the activity load in the air in various areas of the melting shop

During the melting process dust samples were gathered by means of an aerosol sampler. The activities of the paper filters were examined with a Geli-detector. The dust samples were taken at a height of 1.8 m and in different distances of the furnace.
The activity in the air near the furnace during the pouring was detected

in the range of 0.3-0.4 Bq/m^3. The other values were below the detection limit of 0.01 Bq/m^3.
The maximum activity load in the air was below the limit for derived air concentrations for surveillance areas which amounts to 14 Bq/m^3 according to the German "Strahlenschutz-Verordnung".

1.4 Measurements of dose-rates at components of the filter plant

The measured values of the dose-rate at different points of the melting facility showed the highest values in the range of 1-2.5 µSv/h for the tube filter followed by the cyclone.
The obtained dose rates are below the admissible value (2.5 µSv/h) of the Siempelkamp license. During melting the boundary area of the filter facility must be marked. After cleaning up the dust deposits the dose rate decreases to the usual amounts during the following melting days.

2. Preliminary tests with non-ferrous metals (B.8.)

The investigations on the melting of insulated copper motor windings showed that the flue gas of the insulation needs a filter capacity of 12000-20000 m^3/h. The existing filter facility has a capacity of 5400 m^3/h. The high flue gas volume will lead to a great contamination in the surrounding furnace area. Therefore the melting of non-ferrous metals will be made in a new melting shop which will be built at Siempelkamp at the beginning of 1989.

3. Investigation on adding radioactive carbon to the melt (B.9)

According to the iron-carbon-diagramm between 2 % and 4 % carbon can be bound in form of graphite lamellas or as graphite nodules in the cast iron.
The melt tests showed that radioactive filter carbon had a percentage of carbon (70-80 %) and a high portion of ashes (10 %). In addition the chemical analysis showed 5.4 g boron per kg filter carbon. As the admissible percentage of boron in cast iron lies at about 0.0005 % it is not suitable to charge the filter carbon into cast iron melts.
Reflector graphite was very pure. The percentage of carbon was 99.6 % and the portion of boron was 0.003 %. The test melt showed that great blocks must be crushed to reduce the holding time of 40-50 minutes. About 95 % of the carbon was bound in the melt. The only disadvantage is the swimming-up on the melt caused by the low density. The suitability of reflector graphite is given from the point of view of its behaviour at melting. The maximum length of the graphite blocks should be 10-15 cm.
The radiological impact of reflector graphite must be proved in the actual case.

Table 1: Activities after melting

Sample	Activities Total Gamma Bq/g	Activities Total Beta Bq/g
melt	0.058	0.06
slag	0.68	15.7
bag filter dust	41.49	35.4
cyclone filter dust	45.2	48.4

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

Contractor: Commissariat à l'Energie Atomique, CEN Grenoble, France
Contract N°: FI1D-0017
Working Period: January 1985 - December 1986
Project Leader: G. Tanis

A. Objectives and Scope

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

To date, no technique is known to lead to an efficient separation of cobalt at reasonable cost and without involving, as intermediate steps, an increase of the amount of waste. The present research consists in a feasibility evaluation of separating cobalt from stainless steel using vapour phase transport. This process offers the following advantages:
- no additional amount of waste, even for a transient step,
- repeatability allowing high separation factors to be reached.

It has never been applied to alloys such as stainless steel and the conditions of application would have to be assessed by theoretical and experimental research ending with a first estimate of the feasibility of the suggested process.

B. Work Programme

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

C. Progress of Work and Obtained Results

The work has been completed, the final report is available as EUR report No. 11296.

4.4. Immobilisation of Contamination of Large Waste Units by Polymer Coating

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

A. Objectives and Scope

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

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

B. Work Programme

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

B.2. Mechanical, thermal and radio-diffusion optimisation of polymer coating with particular respect to geometry, surface conditions and nature (metal, concrete ...) of the components.

B.3. Study of a mobile projection apparatus suitable to be adapted for application in the nuclear area and able for projection thick coatings on large components as given in B.2.

B.4. Preparation of a projection area at pilot plant scale to demonstrate the feasibility of the procedure on components > 1m and on large dimension low level waste.

B.5. Application of the procedure within the frame of a dismantling project in France or another EC country.

C. Progress of Work and Obtained Results

Summary

Work carried out during 1988 has consisted mainly in :
- building a painting-cell in accordance with regulation of painting process usuable in industry and especially in nuclear plants
- giving example of practical applications of the "Journal Officiel" decrees for painting with thick polyurethen products
- preparing a file for sending large scale wastes towards a ANDRA storage Center
- continuing the measurements of physico-chemical parameters on polyurethen layers, especially diffusion of tritiated water and other radionuclides.

Progress and Results

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

1.1 - Polymeric formulation

This part of the programme is achieved. Progress have been obtained for a better knowledge of different kinds of polyurethen. Relations between organic structure of polyurethen systems and their properties have been established. The choice of a well-adapted polyurethen is now possible as a function of the kind of the practical application in a nuclear plant.

1.2 - Mechanical and Physical Characteristics

The major parameter for assuring guaranties of confinement in an aqueous system, as in a subsurface storage, is the value of the diffusion coefficients of radionuclides. The results have been obtained with tritiated water and other radionuclides as ^{60}Co and ^{137}Cs, on thin membranes of commercial polymers.

The other mechanical characteristics have been also determined in order to answer to specifications of ANDRA.

2 - Optimization of polymeric formulation (B.2.)

Comparative projection applications with the spraying machine have been in progress for last 6 months. To the three formulations (BAULE, SOUPLETHANE, DEBRATHANE) choosen en 1987, a new polyurethen has been added (BAYTEC).

3 - Study of a mobile projection apparatus (B.3.)

The study of functions of a projection cabin has been continued. Painting process has been examined and compared with safety regulation applied in the fields of painting and nuclear industry, that are published in the "Journal Officiel". The geometric arrangement of the cabin and the rate flow of ventilation have been studied in accordance with the requirements of inflammability, toxicity and radioactivity rules.

4 - Feasibility study (B.1., B.4.)

The last point of the programme concerns the acceptability by ANDRA of large scale waste coated by a polyurethen. Evacuation of large scale wastes towards an ANDRA storage Center cannot be done by using normalized containers.

It requires the preparation of a file for the acceptance as "colis particulier".

The file includes the evaluation of several physical parameters and especially diffusion coefficient. The work has been continued to complete the file particularly with regard to polyurethen BAYTEC. Several files of the "dossier particulier" have been sent to ANDRA.

The concerned wastes are e. g. coming from dismantling the AT1 plant in La HAGUE.

4.5. Treatment of Active Concrete Dust by Slurry Setting Method

Contractor: Taylor Woodrow Construction Ltd.
Contract N°: FI1D-0042
Working Period: April 1987 - April 1989
Project Leader: P. Dawson

A. Objectives and Scope

Previous work (Ref.: EUR 9568) established the principle of setting concrete dust from a slurry with a suitable sodium silicate solution.

The objective of this research is to evaluate the viability of a method of immobilising radioactive concrete dust arising from the dismantling of reinforced concrete structures of nuclear installations. The concept to be investigated involves the following steps:

- mix concrete dust with an excess of sodium silicate solution to make a slurry;
- cast the slurry directly into the disposal box and allow it to solidify;
- fill with grout any spaces, due to shrinkage, between the solidified slurry and the walls of the box, and any space under the lid.

An additional option is to demonstrate that lumps of concrete rubble, or possibly steel components available at the same time, may be pre-packed in the box and the slurry injected to fill the spaces.

The research is directed mainly at the biological shields of early Magnox reactors and the pre-stressed concrete pressure vessels (PCPV) of later Magnox and Advanced Gas-cooled Reactors. Structures of other commercial nuclear power plants in the European Community, in particular the PCPVs of French Gas Graphite Reactors and the biological shields of Light Water Reactors, will also be considered.

B. Work Programme

B.1. Review of operational requirements to cast a slurry into disposal boxes with particular view to cost effectiveness.

B.2. Re-consideration of binding material (i.e. sodium silicate) with respect to the thermal conditions of the reaction.

B.3. Provision of two types of relevant concrete and of steel boxes of various sizes to be filled with concrete slurry.

B.4. Large-scale casting of the boxes by the slurry method and analysis of the physical properties and the shrinkage of the material.

C. Progress of Work and Obtained Results

Summary

The year 1987 concluded with the preparation of a detailed task definition for the job, and the commencement of a literature survey on silicate - concrete interaction. The past year has seen the completion of this literature survey, and a second study to characterise the product of concrete demolition debris. The latter study was also undertaken largely through the literature although some limited experimental work has been involved. The main activity of the year has however been an experimental study to determine the extent to which silicate/concrete debris at slurry consistency will harden. Variables investigated have included original concrete composition, grade and concentration of silicate used, mix preparation, and maximum particle size of debris that will harden adequately.

The laboratory program has presented two problems. Firstly, although the resources used have been no greater than envisaged, testing has been slower than planned. We have therefore requested a time-only extension. Secondly, the range of concrete demolition debris particle sizes that set is not great. Typically the entire fraction below 5 mm will solidify. The use of this fraction (in conjunction with sodium silicate) to grout boxes of larger concrete debris or metal components, is probably feasible, but is unlikely to be cost effective. Part of the programme has however been to consider the disposal process for concrete demolition debris, and it is now clear that for conventional grouting of active debris in boxes, removal of the fire material will first be necessary. The silicate process is a viable method of encapsulating and immobilising this material, thus validating the work already undertaken, and the remainder of the programme.

The work will conclude with the completion of the concrete study of the site logistics for the concrete fires disposal process, and an experimental study of the engineering properties (strength, durability etc) of the hardened slurry.

Progress and Results

1. Literature Survey on Concrete/Silicate interaction (B.2.)

Although there is obviously an extensive literature available on silicates, there is little on the concrete/silicate interaction. Even uses such as surface hardiness are reported from an experimental standpoint rather than a theoretical view. The literature gives little insight into why there are various optimum and pessimum levels of silicate concentration to set debris at certain consistancies.

2. Characterisation of Concrete Demolition Debris (B.1.)

A study of the data available on concrete demolition debris indicates that although (as would be expected) the maximum particle size is strongly dependant on the method of demolition used, there is limited evidence that if only the smaller size fractions are considered (say < 20 mm) the gradings of the debris produced by all fragmentous demolition methods are similar. A more theoretical approach based on fracture mechanics and the strain rates associated with differing demolition methods, provides support to this view.

Finally, debris grading produced by crushing of concrete samples cast in the laboratory were shown to be similar to gradings in 'real' demolition. This fact together with data comparing the effects of old and young concrete, has allowed the use of simulated demolition debris in the experimental programme.

3. Experimental Programme on concrete/silicate interaction (B.1.,B.2.)

A detailed study has been undertaken to determine the range of proportions over which debris/silicate slurries set and harden. Variables include silicate solution composition and concentration, silicate to debris ratio, concrete type (and age) and maximum particle size of fraction tested. Most demolition debris has been prepared from concrete samples cast in the laboratory. These were passed through a jaw crusher, and sieved to remove material above a certain size. All the material passing say a 5 mm sieve of a certain size is referred to below on the '5 mm size fractions'.

By far the most important conclusion to emerge from this study is the importance of the size fraction. Fine fractions (say < 1 mm) always set whilst larger fractions ($\geqslant 10$ mm) would not set. Intermediate fractions could be set by optimising silicate grade and dose. This important finding for slurries contrasts with earlier work on briquettes, for which a much greater range of sizes would harden.

A list of important results is included in Tables 1-3. (Attached).

4. Logistics of the slurry setting method on an active site (B.4.)

This study has only just commenced. A review of the various proposals for grouting active concrete demolition debris into disposal boxes strongly indicates that some form of removal of fines will be necessary to facilitate grouting. Prior research into prepacked aggregate concrete systems, in a previous CEC Contract, is proving a useful source of input data for this aspect of the current project.

Table I.   30% PFA, 1 Year Old Concrete

Solution SiO_2/Na_2O Ratio	Debris Fraction	Ratio Debris:Silicate (by wt.)	Initial Gel	Final Gel	Nature of Material After Final Gel	Other Observations
Crystal 79 3.3	Complete	1.6, 1.4, 1.2, 1 and 0.8:1	None	None	All particles sunk to bottom of container with layer of silicate solution on top. No gelation at all	After one week a crust formed on the top of the silicate solution. No other gelation had occurred.
Crystal 100A 2.0	Complete	1.2, 1 and 0.8:1	None	None	As above.	
Crystal 79 3.3	<5mm	1.2:1	1½ hour	Not observed (set overnight)	Hard. Could be broken up with difficulty.	
		1:1	Not observed (>3½ hours)	Not observed (overnight)	As above.	Thin hardened layer of silicate solution on top.
		0.8:1	Not observed (>3½ hours)	Not observed (overnight)	Not as hard as 1.2 and 1:1. More easily broken up.	Thick viscous layer of excess silicate solution.
Crystal 100A 2.0		1.2:1	Not observed (>3½ hours)	(overnight)	Easily broken up. Crumblecrete.	Tacky silicate layer on top.
		1:1	Not observed (>3½ hours)	Not observed (overnight)	Easily broken up. Crumblecrete.	Runny silicate layer on top.
		0.8:1	Not observed (>3½ hours)	-	Easily broken up and remixed with silicate solution.	Large and runny silicate layer.
Crystal 79 3.3	<2.36mm	1.2:1	1½ hours	4 hrs (nearly set); set hard overnight	Hard; broken with difficulty.	Very tacky thin silicate layer on top.
		1:1	2 hours	Not observed; had set overnight	Hard; broken with difficulty.	Viscous silicate layer on top.
		0.8:1	2 hours	Set overnight	Fairly hard. Easily broken.	Viscous silicate layer on top.
Crystal 100A 2.0		1.2:1	2 hours	Not observed; set overnight	Hard broken with difficulty	Viscous silicate layer on top.
		1:1	2 hours	Not observed; set overnight	Easily broken. Crumblecrete.	Viscous silicate layer on top.
		0.8:1	2 hours	Not observed; set overnight	Very soft set	Runny silicate layer on top.
Crystal 79 3.3	<600µm	1.2:1	5 minutes	4 hrs (1½ hrs crumblecrete texture)	Hard. Could not be broken; surface could be marked.	After leaving for 24 hrs all samples had set very hard.
		1:1	5 minutes	4 hrs	Hard. Broken with difficulty.	
		0.8:1	10 minutes	Soft set 4 hrs set overnight	Easily broken.	
Crystal 100A 2.0		1.2:1	5 minutes	Soft set 4 hrs set overnight	Hard. Very difficult to break.	
		1:1	5 minutes	Very soft set 4 hrs	Easily broken.	
		0.8:1	4 hours	Not observed	Crumblecrete texture.	
Crystal 79 3.3	<150µm	1.2:1	5 minutes	20 mins	Hard.	Very viscous mixture.
		1:1	5 minutes	20 mins	Hard.	Very viscous mixture.
		0.8:1	5 minutes	30 mins	Hard.	Viscous mixture after 1 minute.
Crystal 100A 2.0		1.2:1	10 minutes	1 hr		
		1:1	10 minutes	1 hr		
		0.8:1	½ hour	1 hr		

Table II. OPC Concrete - Approx. 2 Months Old

Solution SiO_2 Ratio Na_2O	Debris Fraction	Ratio Debris:Silicate (by wt.)	Initial Gel	Final Gel	Nature of Material After Final Gel	Nature of Slurry after Initial Mixing	Other Observations
Crystal 79 3.3	Complete	1.6, 1.4, 1.2, 1 and 0.8:1	None	None	All particles sunk to bottom of container with layer of silicate solution on top. No gelation at all		
Crystal 100A 2.0	Complete	1.2, 1 and 0.8:1	None	None	As above.		
Crystal 79 3.3	< 5mm	1.2:1	½ hour	Overnight (6-8 hours)	Can be dug out with spatula.	Viscous	
		1:1	40 minutes	Overnight (6-8 hours)	Can be dug out with spatula.	O.K.	
		0.8:1	Overnight	Overnight (6-8 hours)	Overnight set. Can be broken and re-mixed with silicate; after this product hardens.	Runny	Layer of silicate solution on top hinders final set.
Crystal 100A 2.0		1.2:1	2 hours	Overnight (6-8 hours)	Can be dug out with spatula.	O.K.	
		1:1	3 hours	>12 hours	Can be dug out with spatula.	Runny	
		0.8:1	Overnight (6-8 hours)	>12 hours	Overnight set. Can be broken and re-mixed with silicate; product not hard after 1 week.	Runny	Layer of silicate solution on top hinders final set.
Crystal 79 3.3	<2.36mm	1.2:1	5 minutes	Overnight (6-8 hours)	Hard; surface can just be marked with spatula.	Viscous	
		1:1	10 minutes	Overnight (6-8 hours)	Hard; surface can just be marked with spatula.	O.K.	
		0.8:1	½ hour	Soft set Overnight	Hardens after a few days.	Very runny	Layer of silicate solution on top hinders final set.
Crystal 100A 2.0		1.2:1	2 hours	12 hours	Hard; surface can be marked with spatula.	O.K.	
		1:1	3 hours	Not observed; > 12 hours	Hard; surface can be marked with spatula.	O.K.	
		0.8:1	4 hours	Not observed; > 12 hours	Can be broken up and re-mixed after overnight set. Slightly soft after 1 week.	Very runny	Layer of silicate solution on top hinders final set.
Crystal 79 3.3	>600μm	1.2:1	immediate	15 minutes	Hard; easily broken with spatula.	Solid mass	
		1:1	immediate	25 minutes	Hard; easily broken with spatula.	Very viscous	
		0.8:1	5 minutes	40 minutes	Hard; easily broken with spatula.	O.K.	
Crystal 100A 2.0		1.2:1					
		1:1	not yet carried out (waiting for containers)				
		0.8:1					
Crystal 79 3.3	>150μm	1.2:1			Powdery mixture obtained instantly on mixing.	Dry, powdery	
		1:1				Dry, powdery	
		0.8:1	immediate	5 minutes	Hard, can be broken with spatula.	Solid mass	
Crystal 100A 2.0		1.2:1					
		1:1	not yet carried out (waiting for containers)				
		0.8:1					

Table III. OPC Concrete 2 Months Old

Solution $\frac{SiO_2}{Na_2O}$ Ratio	Debris Fraction	Ratio Debris:Silicate (by wt.)	Initial Gel	Final Gel	Nature of Slurry after Initial Mixing	Nature of Final Product
Crystal 66 3.65	<600μm	1.2:1	immediate	5 minutes	Solid mass	Hard; can just be broken.
		1:1	immediate	10 minutes	very viscous slurry	Hard; easily marked and dug out.
		0.8:1	2 minutes	1½ hours	Slurry	Hard; easily marked and dug out.

4.6. Investigations into the Melting of Radioactive Metal Waste in a Controlled Area

Contractor: Noell GmbH, Würzburg, Germany
Contract N°: FI1D-0043
Working Period: October 1986 - December 1989
Project Leader: U. Birkhold

A. Objectives and Scope

The melting of radioactive metal waste has several advantages in comparison with other procedures, i.e.: reduction of the waste volume to be disposed of, safe enclosure of the radionuclides in the metal matrix, safe and exact determination of the radioactive inventory and, under certain conditions, harmless reuse of the metal.

The aim of the investigations is the testing of the melting procedure on various types of waste metal, with surface contamination up to 500 Bq/cm^2 and specific activity up to 200 Bq/g, under permanent operation conditions and in a controlled area.

B. Work Programme

B.1. Investigations on the distribution of radionuclides in melt, slag, furnace liner and filter dust.

B.2. Investigations on secondary waste, in function of the processed material, and on behaviour of the filter system.

B.3. Investigations on committed doses and activity release as consequences of the melting work.

B.4. Overall evaluation of the melting technique and comparison with alternative techniques (decontamination, compaction, direct disposal).

C. Progress of Work and Obtained Results

Summary

Commissioning of the melting plant was essentially completed in autumn 1988. The cooling system will be converted, due to new waste water regulations within KfK. Commissioning will be restarted in March 1989 and the research programme will be started in late spring.

The dismantling work in the NPP Niederaichbach has begun in the contaminated area in November 1988. The first lot of material for melting is ready.

Progress and Results

Due to the fact that no meltings were performed, no results were generated for the relevant work programme.

4.7. Behaviour of Actinides and Other Radionuclides that are Difficult to Measure, in Melting of Steel

Contractor: Kraftwerk Union AG, Erlangen, Germany
Contract N°: FI1D-0044
Working Period: January 1987 - June 1989
Project Leader: E. Schuster

A. Objectives and Scope

Various types of contaminated piping, valves, heat exchangers and vessels are removed from nuclear facilities in the course of decommissioning. Depending on their origin, these components are contaminated with various radionuclides, e.g. alpha-emitters, pure beta-emitters, and gamma-emitters. Unrestricted or otherwise non-hazardous reuse of these components is possible if the residual activity concentrations are below the limits authorised.

To achieve this goal, decontamination processes have to be used in general. In many cases, chemical decontamination of large components with complex surface geometry cannot be performed economically. Recycling can be achieved in many cases using melting processes. Thus the non-hazardous reuse of beta-, gamma-contaminated material which accumulated in the course of repairs and refittings of nuclear power plants has been demonstrated by the contractor in co-operation with Siempelkamp Giesserei GmbH & Co, Krefeld.

The aim of this research programme is to extend the melt decontamination process to materials which are contaminated with actinides and radionuclides that are difficult to measure. The distribution of these radionuclides in the metal and the slag will be determined and direct measuring techniques or representative sampling techniques will be developed.

B. Work Programme

B.1. Literature review related to radionuclide deposition on components, chemical separation procedures for iron and nickel, basic radionuclide data and evaluation of authorised activity limits.

B.2. Sampling of material and test melts at laboratory scale using well known activity quantities and accompanied by an appropriate measurement programme for original material, metal, slag and off-gas.

B.3. Development of direct measuring techniques for alpha-emitters in melt and slag, taking into account the alpha-energy of the emitting nuclides and the sample geometry.

B.4. Development of measuring techniques for pure beta-emitters, such as C-14 and Sr-90, expected to be found in metal and off-gas, and in slag, respectively.

B.5. Development of a sampling technique and simple chemical separation procedures for nuclides decaying by electron capture, such as Fe-55 and Ni-59, emitting weak X-rays which cannot be measured directly.

B.6. Large-scale melt in a commercial foundry of alpha-contaminated material to demonstrate the transferability of the laboratory results to industrial scale.

B.7. Evaluation of results from both laboratory tests and large-scale tests with respect to alpha-activity distribution in metal, slag and off-gas, the most suitable measuring technique and costs.

C. Progress of Work and Results Obtained

Summary

Eight melt experiments have been performed in this reporting period. Six melts were run using inactive steel St37-2 with addition of various radioactive indicators. Two melts were performed with uranium contaminated scrap, originating from a fuel fabrication factory.

Five melt experiments have been evaluated completely, while three melts are still under evaluation.

Even with a rather high uranium quantity added (up to 3.3 E-2 wt %) a decontamination factor of 75 was reached in one experiment. This is in the same order of magnitude obtained in a former americium tracer experiment. A high decontamination factor in connection with a rather high uranium contamination seems to depend on the addition of suitable slag forming materials.

The results of melt down runs with scrap from a fuel fabrication factory show, that the remaining α-contamination of the ingot is well below the limiting value of 0.1 Bq/g.

Progress and Results

1. Preparation of traced scrap (B.2.)

All together eight melt experiments have been performed in course of the reported period. The parameters of the various melts are summarized in Table I . Six melt experiments were run with the addition of radioactive indicators, while 2 melt down runs were performed with uranium contaminated scrap, originating from a fuel fabrication factory.

The radioindicator isotopes were accomodated in a drilled hole in one section of the inactive steel. The hole was plugged after insertion of the radioindicator to avoid any uncontrolled losses during the melt process.

2. Results of the laboratory-scale melt experiments (B.2)

The melt experiments 1 to 5 (see Table I) have been evaluated completely, the results will be discussed in the following. Melt experiments 6 to 8 are still under evaluation.
The evaluation of the uranium content was performed by α- and γ-measurements as well as by neutron activation analyses. The poor agreement of the three different determination methods resulted in detailed investigations on the isotope composition of uranium indicator compounds used in the experiments. The resulting influence on the uranium evaluation by activity measurements is still subject of evaluation. Hence the behavior of uranium in the melt experiments 1 to 5 of Table I is only assessed by determination of uranium content via neutron activation analyses. The results are compiled in Table II showing that the addition of slag forming materials increases the release of uranium into the slag and thus decreases markedly the uranium content in the ingot (see Table II,melt 3). The overall decontamination factor of melt 3 was 75. This is the same order of magnitude determined by the preceding americium tracer experiment /1/. This high decontamination factor in connection with a rather high uranium contamination of the starting material seems to depend on the addition of suitable slag forming materials.

Melt 4 and 5 were performed with scrap originating from a fuel factory. This scrap was contaminated with 3.4 % enriched uranium. For reasons not known up to now melt 4 contained many shrinkage cavities in the ingot, hence the evaluated analytical values of this melt should not be taken for the assessment of the decontaminaton behavior. Melt 5 shows a rather good decontamination behavior though no additional charge of slag forming material was applied. The average value of uranium retained in the ingot is only by a factor of 2 higher compared to the natural uranium content of inactive steel St37-2 which was used as starting material in the three uranium indicator experiments 1 to 3 (see Table II). An additional amount of natural uranium originates from the melt additives applied in the melt experiments, their α-content is shown in Table III.

The natural uranium content of 0.8 μg/g Fe measured in several inactive steel samples of the quality St37-2 corresponds with 0.02 Bq α/g Fe. Paying regard to an additional contribution by melt additives to the natural uranium content of the ingot, the measured 1.6 μg U/g Fe in Melt 5 which corresponds with 0.04 Bq α/g Fe may be a value that cannot went markedly below.
The limiting value of 0.1 Bq/g for the unrestricted release of melted scrap is on the one hand by a factor of about 2 higher compared to the above mentioned value but on the other hand this is a rather low margin to the natural α-activity content of the used steel.

References

/1/ The community's Research and Development Programme on Decommissionning of Nuclear Installations Third Annual Progress Report (Year 1987) EUR 11715, p. 176-180.

Table I: Material, Additives and Boundary Conditions of Laboratory Melt Experiments in 1988

Melt No	Material	Added radionuclides	Additives	Boundary Conditions of melt
1	steel St37-2; 3250 g	1 g elem. U	C: 3 % SiC: 2.5 % slag former: 1 %	all melts: temperature 1480 - 1530 °C cover gas: argon, 10 % H_2 cooling time: 6 h dwell time 15 min
2	steel St37-2; 3200 g	1.15 g UO_2	C: 3 % SiC: 2.5 % slag former: 1 %	dwell time 25 min
3	steel St37-2; 2000 g	0.75 g UO_2	C: 3 % SiC: 2.5 % SiO_2: 2.7 % CaO: 2.7 % Al_2O_3: 1.4 %	dwell time 45 min
4	contaminated austenitic steel scrap; 3000 g	U (3.4 % U235) contamination	C: 3 % SiC: 2.5 % slag former: 1 % VL63M: 0.1 % DISPERSIT: 0.1 %	dwell time 40 min
5	contaminated ferritic steel scrap; 3700 g	U (3.4 % U235) contamination	C: 3 % SiC: 2.5 % slag former: 1 % VL63M: 1.2 % DISPERSIT: 0.1 %	dwell time 40 min
6	steel St37-2; 3150 g	Sr85: 3.7 E+6 Bq	C: 3 % SiC: 2.5 %	dwell time 30 min
7	steel St37-2; 3150 g	Sr85: 3.7 E+6 Bq	C: 3 % SiC: 2.5 % SiO_2: 1 % CaO: 1 % Al_2O_3: 0.5 %	dwell time 30 min
8	steel St37-2; 3180 g	C14 elem. 1 mg 3.7 E+6 Bq	C: 3 % SiC: 2.5 %	dwell time 30 min

Table II: Uranium Content from Neutron Activation Analysis

Melt No	Added Uranium	Melt position	μg U/g Fe analysis 1	μg U/g Fe analysis 2	Average of analysis 1 and 2 μg U/g Fe	Average of analysis 1 and 2 ≙ Bq* α/g Fe
1	U elem.	1	93	146		
		2	161	-	130	3.3
		3	150	102		
2	UO_2	1	4.5	10		
		2	37	-	27	0.7
		3	52	30		
3	UO_2	1	2	4		
		2	7	4	5	0.1
		3	6	6		
4	cont. scrap (3.4 % U235)	1	10	9		
		2	5	10	9	0.2
		3	16	4		
5	cont. scrap	1	0.8	0.7		
		2	6	1	1.6	0.04
		3	0.7	0.6		
1-3	inactive starting material steel St37-2		-	0.8	0.8	0.02

*calculated by 99.27 % U238
0.72 % U235 ≙ 25300 Bq α/g U
0.005 % U234

Table III: Uranium Quantities of Melt Additives

Additive	Bq/g (alpha-meas.)
slag former	25.3 Bq/g
SiC	7.05 Bq/g
coal	1.67 Bq/g
Dispersit	19.94 Bq/g
VL 63 M	35.24 Bq/g

4.8. Conditioning and Disposal of Radioactive Graphite Bricks from Reactor Decommissioning

Contractor: Commissariat à l'Energie Atomique, CEN Valrhô, Bagnols-sur-Cèze, France
Contract N°: FI1D-0064
Working Period: January 1987 - December 1988
Project Leader: J.R. Costes

A. Objectives and Scope

The decommissioning of gas-graphite reactors in the EC (e.g. French UNGGs, British Magnox reactors and AGRs, and reactors in Spain and in Italy), will produce large amounts of graphite bricks.

Evaluations of the radioactivity inventory of graphite moderators, made in France and the UK (ref. EUR 9232), show that this graphite cannot be accepted without particular conditioning by the existing shallow land disposal sites.

The aim of the study is to examine the behaviour of graphite waste and to develop a conditioning technique which makes this waste acceptable for shallow land disposal sites.

B. Work Programme

B.1. Definition of the site-specific conditions to be taken into account, in relation with the French waste disposal agency (ANDRA).

B.2. Study and fabrication of a particular device suitable for machining cylindric samples of 50 or 80 mm diameter out of graphite blocks of the G2 reactor.

B.3. Characterisation of the graphite on the basis of three samples (one being used as a reference).

B.4. Study of impregnation procedures using bitumens or polymers capable of penetrating several millimetres into the graphite.

B.5. Study of appropriate conditioning procedures of the graphite before final disposal.

B.6. Study of the industrial feasibility of the procedures and evaluation of costs.

C. Progress of work and obtained results

Summary

As stipulated in the contract, 18 kg of graphite core samples with an outside diameter of 74 mm were removed from the G2 gas-cooled reactor at Marcoule. The samples were furnished to CEA laboratories for analysis and continued examination. At the same time, the LETI laboratory at Grenoble carried out a tomographic comparison among nonradioactive graphite samples, some of which were impregnated with epoxy resin. The results confirm the mercury porosimeter findings, i.e. that impregnation to the center of the graphite bricks can be seriously considered. In addition to the radiochemical analysis methods, the ORIS laboratory at Saclay conducted a series of autoradiographic examinations on small radioactive graphite samples. The photos clearly reveal a ^{14}C irradiation background on which impurities are highlighted by irradiation spots due essentially to ^{60}Co.

Progress and Results

1. Extraction of graphite samples from a GCR core (B.1., B.2.)

Following approval of the proposed procedure by the Reactor Safety Commission, a test was performed under nonradioactive conditions to ensure that the metal structures (12 mm + 20 mm thick) could be perforated in complete safety.

In order to prevent contamination of the reactor building and to limit dispersion of inactive or activated dust particles from the reactor block, a containment lock was set up and maintained under negative pressure from the moment the thimble was drilled through until the operating zone was decontaminated on completion of the task.

When the thimble was open, air was drawn into the reactor block by suction pressure (20-60 mm H_2O). When the thimble was closed by its plug or a vinyl sleeve air was drawn from the lock by a Cobra blower across a HEPA filter.

After setting up the tooling required for this operation, drilling began on the 12 mm thick end wall of the steel thimble in the northwest power measurement chamber at 1:00 pm on January 21, 1988, lasting until 9:00 am on January 25. No cooling provisions were required during the 10 hours of actual drilling.

The 20 mm backing steel plate was drilled from 9:15 am to 2:30 pm on January 25 (the actual drilling time was 3½ hours). Air cooling was provided in view of the negligible swarf dispersion hazard because of the work configuration.

A specialized firm was called in on January 26 to inspect the situation. The images obtained indicated satisfactory results and confirmed that the opening was perfectly centered for taking graphite core samples.

Graphite core drilling began on February 8. Initial tests showed that the tool penetrated well into the graphite at a rate of about 1 meter in 20 minutes. Air blasts were used to facilitate tool rotation and prevent jams.

No problems were encountered during removal of the samples given the low activity levels. Thirteen samples were taken in four successive operations, representing a total length of 2.46 m inside the reflector and moderator : eleven 200 mm cores and two 130 mm cores. The sampling procedure was completed at noon on February 9.

Externally, all the samples were in excellent condition. Since the samples were cored along the guide tube centerline, however, eight of the samples were scalloped at the channel positions.

The contact dose rates for all the cores samples reached 1 G/h, and the mean dose rate sustained by the operators was 20 µG/h at 1 meter. The corresponding values after the samples were placed in vinyl wrappers and industrial packing cases (5 mm lead + 2 mm steel).

No atmospheric contamination was measured on the HEPA filter at any time during the sampling procedure.

For a total sample mass of 18 kg the overall activity was 2.0 GBq.

The core samples were distributed to the CEA laboratories involved in this study

2. X-Ray Tomography of standard and resin-impregnated non radioactive graphite samples (B.3.)

Four graphite prisms measuring 50X50X100 mm were stamped "G" or "R" as follows :

- type "G" 1 control specimen (G_t)
  1 specimen (G_i) impregnated with epoxy resin (10.5wt%)
- type "R" 1 control specimen (R_t)
  1 specimen (R_i) impregnated with epoxy resin (12.9wt%)

The system was an experimental single-detector setup recording digital transmission data on magnetic tape to be sent to the data processing center. The X-ray source was used with a small focal point (1.8x1.8 mm). The beam width at mid-height was 370µm. The four samples were placed side-by-side. The sample integration period was 20 msec.

Each image pixel was assigned a relative density between 1 and 1024 with the highest numerical value corresponding to the highest density.

Individual zones of the image can be statistically analyzed using the mean pixel density values for the desired areas.

The following conclusions may be drawn from the examination of these images :

- visual observation of the images, notably those with the smallest sampling pitch (80µm) reveals relatively extensive porosity and incipient cracks. Comparisons between the control specimens and impregnated specimens suggests that impregnation was successful both lengthwise and transversely ;
- statistical image analysis confirms that the impregnation was successful to the core, and provides an impregnation ratio very close to the specification value.

Moreover, examination of the porosities suggests that the material deficiencies do not correspond to air bubbles but rather to volume with high local porosity.

This technique can be considered suitable for the development of an industrial impregnation system and for random sample quality control purposes.

3. Qualification of autoradiographic examinations (B.3.)

Before the large active graphite samples described in paragraph 1 were available, it was considered advisable to qualify an autoradiographic method on the small active specimens then available. This type of examination is planned to provide data on radioactivity distributions in the samples and on possible superfical heterogeneity.

A feasibility study was therefore carried out with 5 cylindrical specimens ranging from 8 to 12 mm in diameter, 16 to 17 mm high and weighing from 2.5 to 4 g.

The radionuclides in the graphite can be divided into three groups according to their radiographic yield :

- ^{41}Ca, ^{55}Fe and ^{59}Ni do not emit β particles or conversion electrons, and have a negligible effect on the result.
- ^{3}H is a low-energy (18.6 keV) β emitter.
- All the other emitters (^{14}C, ^{36}Cl, ^{63}Ni, ^{60}Co, ^{93m}Nb, ^{133}Ba, ^{134}Cs, ^{137}Cs, ^{154}Eu and ^{155}Eu)

contribute to the photographic impression to roughly the same extent, assuming equal emission surface areas. In fact, two radionuclides predominate: ^{14}C (with approximately 1.6×10^{5} Bq per specimen) and ^{60}Co (approximately 9×10^{4} Bq per specimen). Together they account for about 90% of the activity attributable to the radionuclides of the third group.

The image obtained during the autoradiography process can thus be assumed to be formed by the following factors :

. as a uniform gamma haze visible around the edges of the specimens
. a homogeneous region due to β emission from ^{14}C
. spots due to β particles from fission and/or activation products, and which can be considered to result from ^{60}Co.

However, autoradiography is unable to select the energy level of the β particles responsible for the photographic image.

4.9. Separation of Contaminated Cement Stone and Non-contaminated Concrete Aggregates

Contractor: Netherlands Organisation for Applied Scientific Research, Delft, The Netherlands
Contract N°: FI1D-0068
Working Period: January 1987 - June 1989
Project Leader: A.P.S. Reymer

A. Objectives and Scope

In a nuclear installation, concrete in various building structures may become contaminated during operation. When the installation is withdrawn from service and eventually dismantled, the contaminated concrete must be conditioned and disposed of as radioactive waste.

Preliminary research carried out by the contractor indicates that the contaminating substances penetrate only into the porous cement stone and not into the mostly watertight aggregates.

The objectives of this research programme are to check the beforementioned indication and, if it is confirmed, to develop, on laboratory scale, a process for separating the constituents of contaminated concrete into a contaminated and a non-contaminated part in order to reduce the amount of radioactive waste.

B. Work Programme

B.1. Checking of the hypothesis that only cement stone is contaminated by analysis of contaminated concrete samples from various nuclear installations.

B.2. Laboratory-scale development of a separation process by e.g. thermal shock and screening sheet on samples of various types of relevant concrete, including contaminated samples as far as available.

B.3. Development of a washing process for contamination-containing aggregates (if any), based on an investigation of various washing fluids.

C. Progress of Work and Obtained Results

Summary

On the basis if previous work (Part B.1.: Checking the hypothesis that only cementstone is contaminated) the development of a separation technique for contaminated concrete on laboratory scale is started (Part B.2.). In this programme three types of concrete were separated into cementstone (hardened cement) and aggregates by means of milling and thermal shock. In the milling process the following parameters are important: filling ration, use of liquids and the diameter of the milling vessel. For three types of concrete, with different mixture compositions, these parameters are optimized. For wet milling, the best filling ratio is 35%, the optimum water scale is 121 mm.
For thermal shock, different relative humidities were assessed. A relative humidity of 30% gives the best results. A process of heating the concrete sample under various conditions has been initiated.

Progress and Results

Development of a separatioj process (B.2., B.3.)

The development of a separation process was started in order to separate the concrete into cementstone and aggregates. Three types of concrete with different mixture compositions were used. To separate the concrete, three techniques were checked on their performance:
- milling technique
- thermal shock
- surface active additives.

Research was done on the effectiveness of various milling techniques depending on different parameters, of which the most important are:
- filling ratio of the vessel
- use of liquids
- diameter of the milling vessel
- use of additives (detergents)
- use of milling balls.

The influence of the relative humidity and the influence of milling time after the thermal shock was checked, as well as the influence of additives on the milling efficiency.

For the milling technique, the best results were obtained with a diameter of 121 mm, a water amount of 20% and a filling ratio of 35%, see figures 1, 2 and 3.

A relative humidity of 30% during thermal shock gave the best results (measured each 15 minutes), see figures 4 and 5.

The fineness modulus given in figures 1, 2 and 4 is determined by adding the different cumulative residues on the sieves divided by 100 (Rengers-Antonisse method). It gives the aggregate size distribution.

The additives used in the process are used in two solutions: 0.01% and 0.1%. From the results, a solution of 0.01% D.N.P. is chosen, see figure 6.

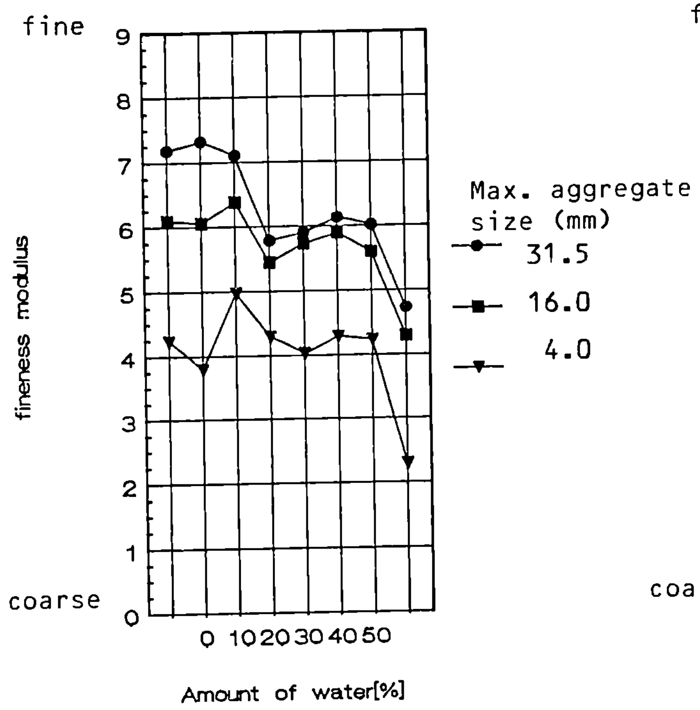


Figure 1: Influence of water content on milling

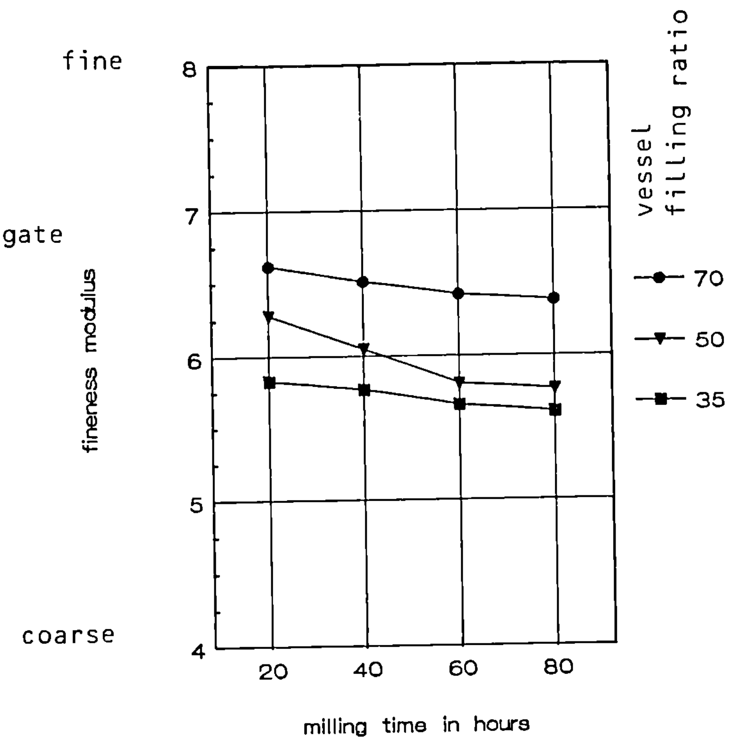


Figure 2: Influence of filling ratio on milling

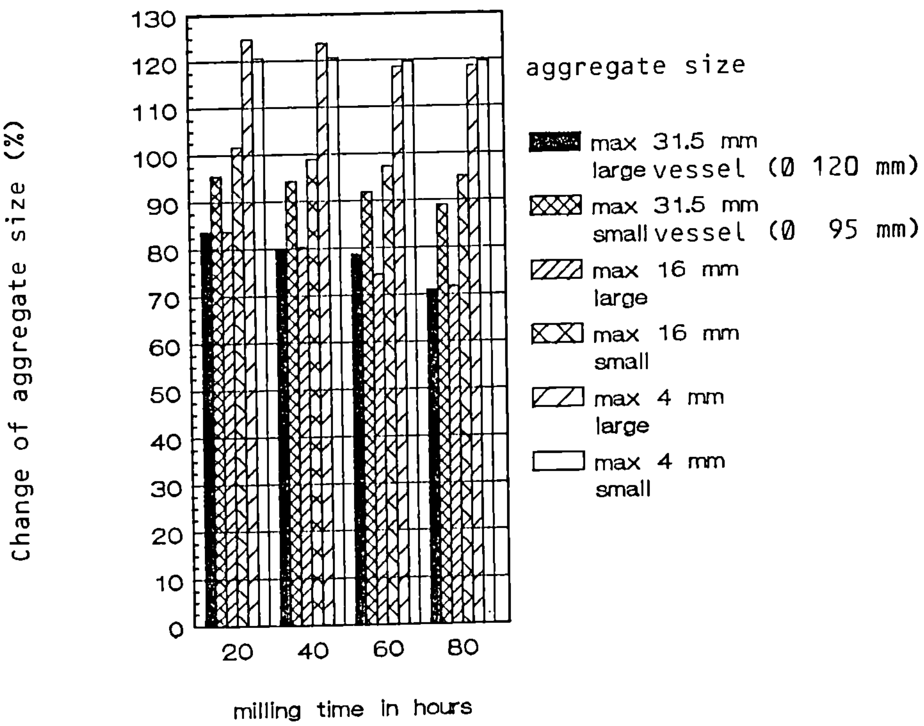


Figure 3 : Influence of vessel diameter on milling

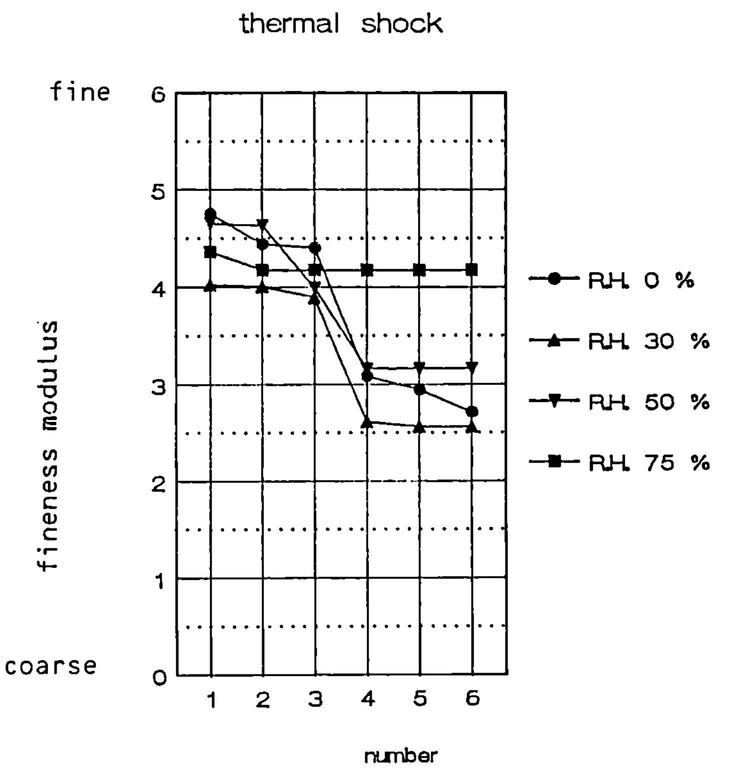


Figure 4: Influence of relative humidity and thermal shocks on fineness modulus

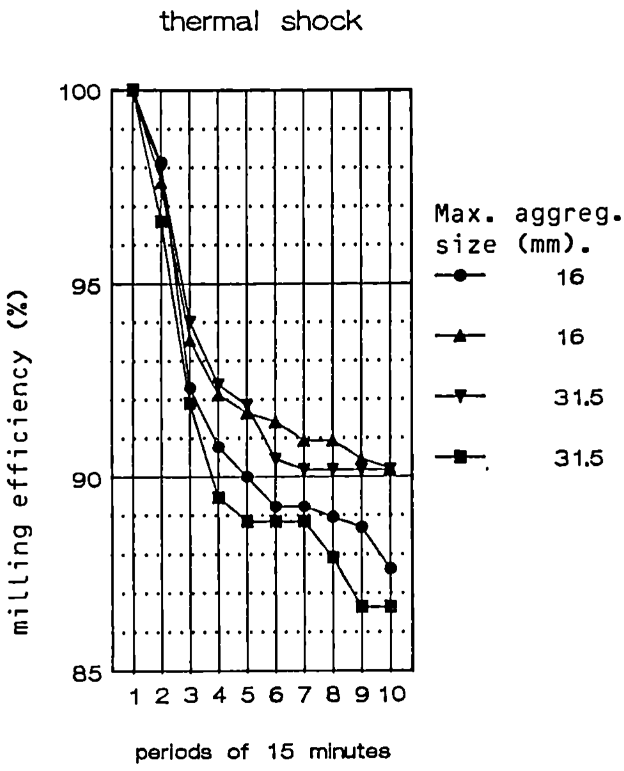


Figure 5: Influence of milling time on milling

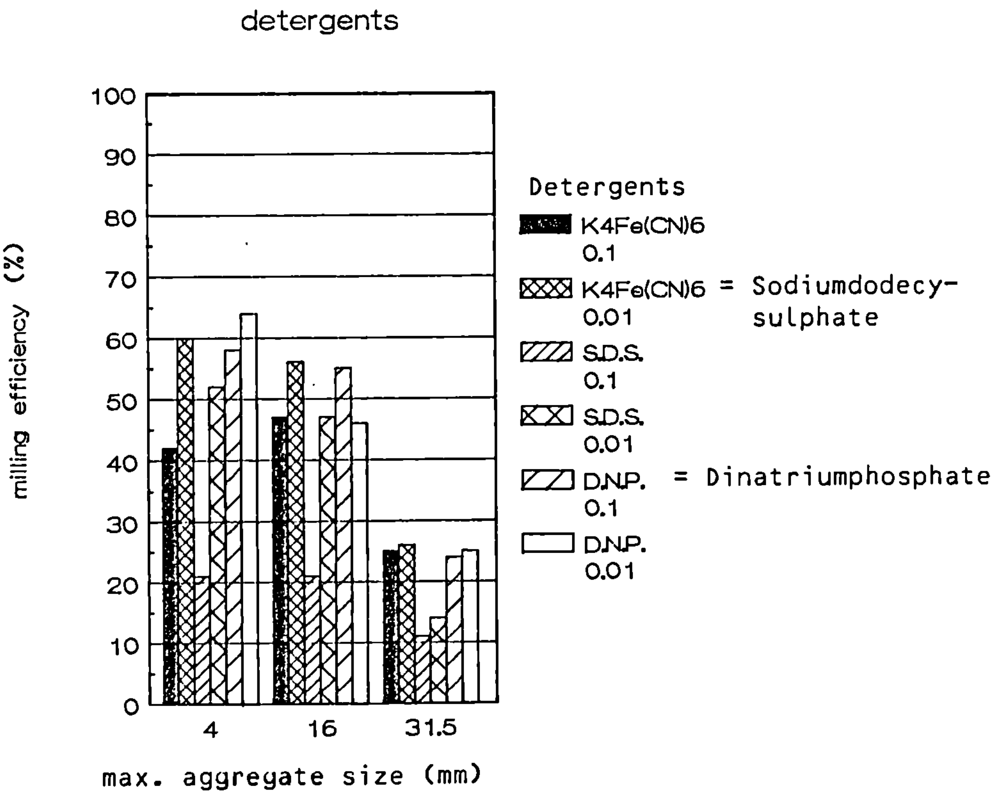


Figure 6: Influence of detergents on milling

5. PROJECT N°5: LARGE CONTAINERS FOR RADIOACTIVE WASTE PRODUCED IN THE DISMANTLING OF NUCLEAR INSTALLATIONS

A. Objective

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

B. Research performed under the 1979-83 programme

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

C. 1984-88 programme

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

D. Programme implementation

Three research contracts relating to Project N°5 were executed in 1988, of which two were completed.

5.1. Design and Evaluation of Large Containers for Reactor Decommissioning Waste

Contractor: United Kingdom Atomic Energy Authority, AEE Winfrith, United Kingdom
Contract N°: FI1D-0045
Working Period: July 1986 - November 1988
Project Leader: M.S.T. Price

A. Objectives and Scope

The system study carried out under the first five-year programme, led to the evolution of design concepts of Type B and Low Specific Activity (LSA) transport containers and to an evaluation for the number of containers required to transport decommissioning waste from a pressurised or boiling water reactor, as well as the associated transport costs and radiological detriment.

The aim of the research is to check these design concepts in relation to the influence of manufacture, handling and disposal on design and transport hazards. The examination of transport hazards will lead to the identification of appropriate package performance.

B. Work Programme

B.1. Project definition.

B.2. Effect of the manufacture on design for large waste containers made out of reinforced concrete and for ferrous metal packages.

B.3. Transport hazard survey to evolve various accident scenarios and to identify the most extreme accident scenario.

B.4. Effect of disposal on design taking into account the environmental impact of the waste.

B.5. Definition of performance criteria for package design based on the ALARA approach.

B.6. Quantitative assessment of proposed concrete and steel package design concepts using simple computer-aided methods, and revision of the original concepts if necessary.

C. Progress of Work and Obtained Results

Summary

The structure of the work programme for the study of large transport containers for decommissioning waste has remained unchanged although the timescale for completion was extended to 30 November 1988. The main tasks performed during the period related to the preparation of package design criteria (Task 5) and the assessment of package design (Task 6). The study has concentrated on Industrial Package designs since Type B containers can rely on experience in the design of irradiated fuel transport flasks.

In carrying out Task 5 the following aspects have been examined:

Factors Affecting Package Design

waste inventory
container manufacture
influence of lifting features
waste loading
temporary storage
transport
disposal

Factors influencing package specification

'mandatory' requirements
design targets
additional analyses covering fire and impact behaviour

Completion of Task 6 has involved the interpretation of the package specification as a means of defining preliminary package design concepts. A minimum package design has been assessed and designs capable of giving enhanced performance have been examined for their performance and cost.

For countries which do not have radioactive waste disposal sites in operation there is considerable uncertainty about the cost of disposal. Under these circumstances it is concluded that there is merit in choosing a design concept which is relatively insensitive to disposal cost variations. The most robust design of large Industrial Package is shown to be one with a returnable shield.

Progress and Results

1. Task 5 (B.5.)

A draft version of the report on Task 5 was issued for comment in February 1988 and discussion took place throughout March. The final version of the Task 5 report was then prepared and the conclusions of this phase of the study are summarised below:

The aim of Task 5 was to identify aspects of manufacture, transport and repository design which impose constraints on the design of large packages for the transport of decommissioning waste. These constraints were summarised in a design specification for packages to form both the conclusion of the Task 5 report as well as the brief for Task 6 - the preliminary development of package design concepts.

The constraints on package design were divided, for discussion purposes, into those arising from technical aspects of decommissioning, the costs of the various operations involved, and the social-political consequences of decommissioning. Work done by other participants in the project had contributed information in relation to the manufacture of packages in both concrete and ferrous metal (Tasks 2A and 2B), transport and possible transport accidents (Tasks 3), and constraints from final disposal operations (Task 4).

The Task 5 study considered the technical constraints according to the sequence of decommissioning operations beginning with the waste arisings and package manufacture through filling, handling and transporting the package to final disposal in the repository. Economic constraints have been similarly broken down into manufacturing, filling, temporary storage, transport and final disposal costs. Socio-political aspects have been examined chiefly in relation to package transport.

Although the amount of shielding required for the majority of waste is low, the very high costs of disposal encourage the minimisation of volume in the repository. Thus any cost savings at final disposal must be balanced against increased complexity (and hence cost) of package manufacture, handling and transport. The performance of a package in an accident will also depend on the type of package adopted. The Task 5 report indicates the factors to be considered when comparing preliminary package designs in Task 6.

The Task 5 study concluded with a package specification which also formed the brief for Task 6. The specification was split into two sections; the first presented the mandatory requirements for packages while the second presented design targets. Thus all package designs must satisfy the first part of the specification, but they do not necessarily have to meet all the requirements in the second part.

The so-called 'mandatory' requirements include compliance with the IAEA transport regulations with respect to dose rates, package robustness and so on, provision of a fifty year temporary storage life, a maximum weight of 65 t and external dimensions to satisfy the railway loading gauge (that for the United Kingdom being the most restrictive of the three nations considered).

The design targets were presented as requirements which enhanced package designs might satisfy to some degree; all the points would be beneficial to a package design but their benefit must be weighed against cost. The design targets included provision of additional shielding to allow hands-on handling of the package, minimisation of release from the package during the first one hundred years after disposal, simplicity of handling and filling operations, and consideration of stacking stability as well as additional impact loads representing feasible handling and transport accidents. The appearance of the package during transport must also be considered. Finally it was suggested that the performance of a package when subjected to the IAEA tests for Type B packages should be evaluated, although the package would not be designed to such standards.

Work on Task 6 began in mid-April 1988. Once the Package Specification had been finalised (as the end of Task 5), the aim of Task 6 was to develop a range of package designs which satisfied mandatory requirements fully but meet the design target to a greater or lesser extent. Both concrete and ferrous metal and a combination of the two were considered for package manufacture and both self shielded and returnable shielding concepts were developed.

Four variants of the self shielded package concept have been prepared for comparison. They would be manufactured in varying combinations and thicknesses of concrete and steel to give a shield thickness equivalent to 100 mm concrete (equivalent to 30 mm steel). Only one concept of returnable shielded package has been examined in detail. This is a cemented internal package with 75 mm thick walls and a 150 mm base within an 8 mm welded steel plate returnable shield with internal stiffeners. The cost of the five package concepts had been evaluated on a comparative basis as a function of the disposal cost and the results are given in Table I

The inferences from Table I are that, at the lower end of the range of disposal costs, manufacturing costs are dominant such that the best package will be the self shielded concrete package (SS 100/000). At the higher end of the disposal cost range, the best package is the one most efficient in volume terms, ie the self shielded metal package (SS 000/030).

The concept with the most shallow gradient overall is RS 075/008 returnable shielded concept and this is illustrated in Figure 1.

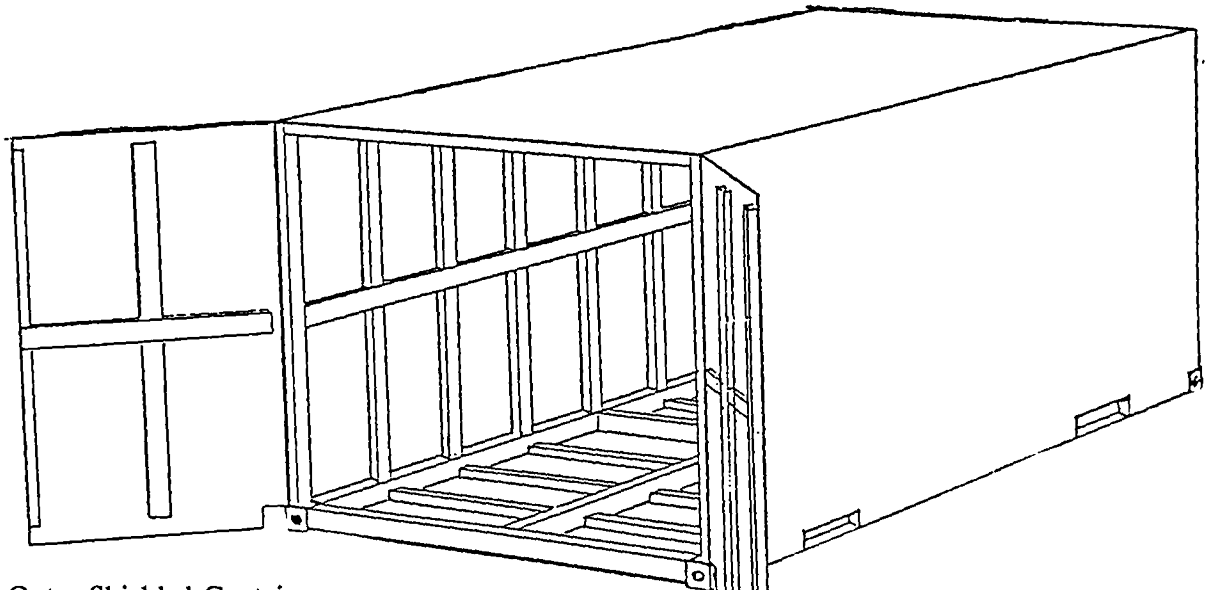
Because, in a country such as the United Kingdom, disposal of decommissioning waste which requires shielding has not commenced it is important to consider how any package design can be improved to achieve enhanced performance. It is obviously difficult to predict what will be the nature of the IAEA Transport Regulations in the period, say, 2005-2010 when the UK disposal site will first be available. Nevertheless, the final part of the study has examined the cost of enhanced performance. It has been shown that the addition of timber to the basic returnable shield improves the impact performance slightly and the thermal performance significantly at little additional cost.

Table I: Comparative Costs of the Package Concepts*

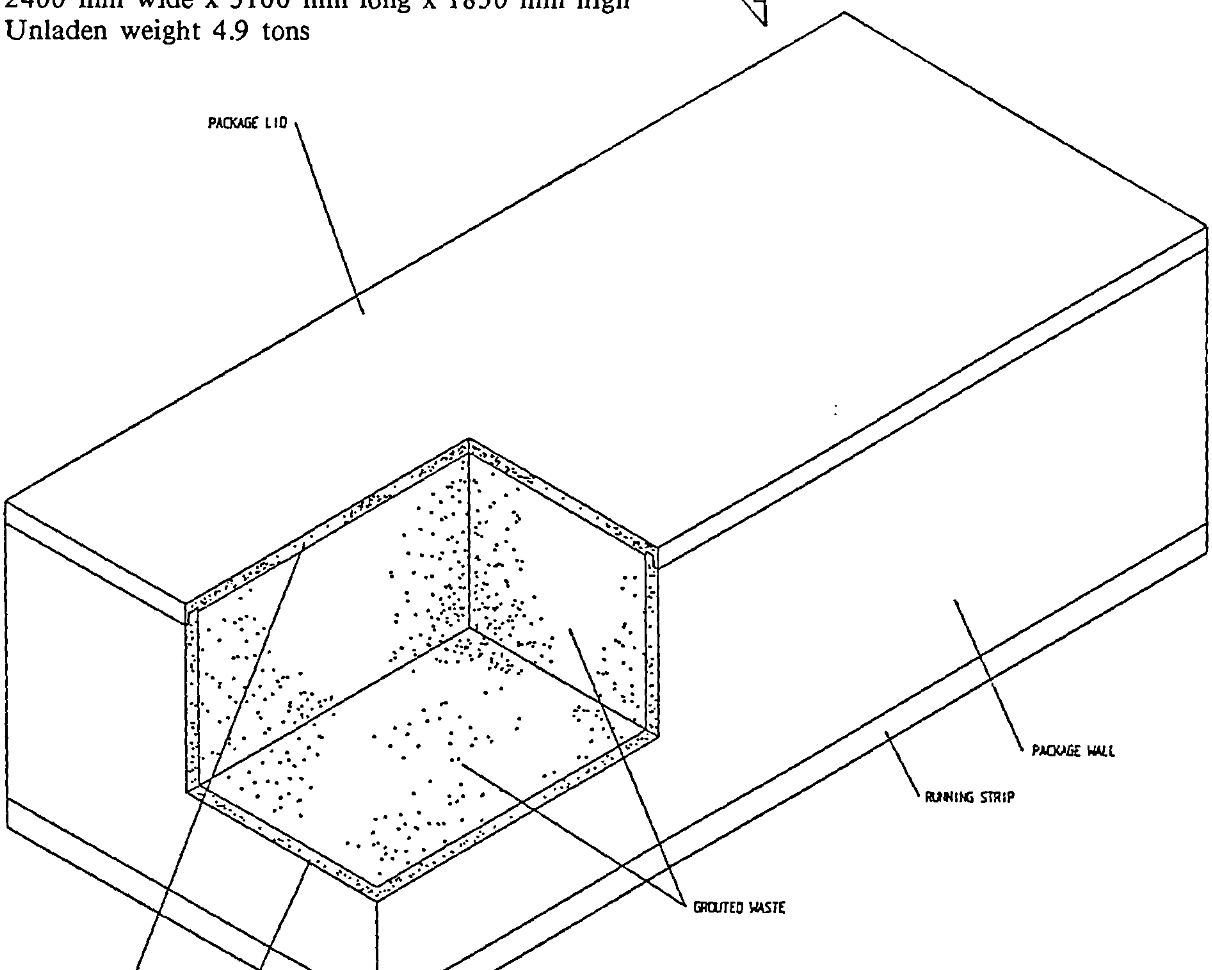
Design Concept**	Disposal Cost (£/M^3)			
	600	1400	2500	7000
SS 100/000	281.1	603.7	1047	2862
SS 075/008	312.2	612.5	1025	2714
SS 055/015	361.4	652.2	1052	2688
SS 000/030	482.9	749.6	1116	2617
RS 075/008	287.5	600.0	1030	2788

* Comparative costs (in £ million) are the costs of aspects of the total decommissioning cost which vary from concept to concept; ie package manufacture, transport, disposal and total number of packages.

** The first two letters, either RS or SS, indicate whether the package is self shielded or has returnable shielding. The following six numbers indicate shielding thickness, the first three the thickness of concrete in mm and the second three the thickness of steel in mm.



Outer Shielded Container
2400 mm wide x 5100 mm long x 1850 mm high
Unladen weight 4.9 tons



Internal Concrete Container holding grouted waste
2260 mm wide x 4900 mm long x 1660 mm high
Weight (full) 60.0 tons

Figure 1: Returnable Shielded Industrial Package for Intermediate Level Waste.

5.2. Large Waste Containers made of Fibre-reinforced Cement

Contractor: Société Générale pour les Techniques Nouvelles, Saint-Quentin, France
Contract N°: FI1D-0046
Working Period: June 1986 - December 1988
Project Leader: C. Jaouen

A. Objectives and Scope

The storage in large containers of radioactive wastes issued from the dismantling of nuclear facilities must be taken into account for establishing a general methodology of decommissioning. Since 1980, SGN and EVERITUBE have been promoting medium-sized cement-based containers, reinforced with asbestos fibres, for conditioning low-level and medium-level radioactive wastes.

The objective of this research is to develop large cement-based containers, reinforced with various fibre materials other than asbestos, with the technology used for fabrication of asbestos cement pipes.

The research will be based on the current components to be disposed of, with respect to the recent improvements in the disassembling of large metal components. Drums and other conventional unshielded containers already used for decommissioning of nuclear facilities will also be taken into account.

The containers to be developed are subject to limitations of external dimensions and weight allowing them to comply with international regulations for road and railway transportation. These containers should generally be used without additional shielding, for storage of low-level and medium-level radioactive materials.

B. Work Programme

B.1. Compilation of container requirements, activity levels, transport and disposal conditions, in particular for large components.
B.2. Selection of appropriate cement/fibre composites based on all available relevant information and experiences.
B.3. Experimental evaluation of the selected materials at pilot plant scale with respect to relevant criteria.
B.4. Definition of main parameters of a range of large containers, compatible with existing transportation means and storage/disposal facilities.
B.5. Development of a prototype container and recommendations for further research.

C. Progress of Work and Obtained Results

Summary

During this period, tests were carried out on the specimens of the following selected materials: Everite n° 1, Everite n° 2, Lafarge Nouveaux Matériaux (LNM) and Farsil mortar.

The LNM material was selected and the criteria for the large container were then defined. Manufacturing has started and delivery is scheduled for February 1989.

Progress and Results

1. Selection of material (B.3.)

Following materials were selected for testing:

a) Everite N° 1:
- Cellulose: 4.5%
- Polyethylene: 1.5%
- Alkali-proof glass fibres (CEM FIL type): 3%
- Portland cement (CPA - HPR): 87%
- Silica fumes: 4%.

b) Everite N° 2:

Identical to formulation N° 1 but with 5% glass fibres of the alkali-proof type and 85% Portland cement (CPA - HPR).

c) Lafarge Nouveaux Matériaux (LNM)/Tréfileries de Conflandey

Fibre-reinforced mortar comprising a mix of cement, aggregates with specific size and resistance, superfines and additives, and reinforcement steel fibres (length: 17 mm; diameter: 0.2 mm; tensile strength: 135 kg/mm^2; proportion: 4 wt%).

d) SGN Formulation - Farsil mortar (compatible with the container molding process)

Alkali-proof glass fibres (Saint-Gobain) and concrete matrix comprising blast-furnace slag cement (CLK) with silica fumes and additives:
- Water: 21.1%
- Silica: 16.6%
- CLK cement: 45 - 60.2%
- Pozzolith 400 N: 0.45%
- Alkali-proof glass fibers: 1.65%
- Cement to water ratio: 0.35%.

Table 1 shows the clear superiority of the LNM material selected for the research programme.

2. Definition of the large container (B.4.)

The selected container is cylindrical, 1000 mm in diameter, 1500 mm height and 60 mm wall thickness. Notches are provided at the lower part so that it can be lifted by a truck (see Figure 1). The external finish is very smooth and consistent with non-contamination checking by wipe test.

The container is molded with a bottom.

3. Fabrication of a prototype (B.5.)

The container without lid will be delivered to the laboratory by the end of February 1989.

No satisfactory solution has been found so far for closing the container (lid).

Further developments are still desirable, namely:

a. Satisfactory container closing: the contractor hopes developing an expansive mortar complying with French containment requirements (ANDRA and CEA).

b. Fabrication of a larger container (1000 mm in diameter, 3000 mm high).

c. **Fabrication of a few containers of different sizes.**
d. **Performance of a test programme on these containers (full scale).**
   - resistance to thermal cycling,
   - immersion,
   - free fall,
   - load test,
   - leak test (following thermal cycling).

TABLE 1 - COMPARATIVE TABLE

TESTS	LNM	EVERITE	FARSIL MORTAR
Compression (crushing)	119.3 MPa	(*)	79.5 MPa
Compression (crushing) after thermal cycling	160.4 MPa	(*)	96.4 MPa
Compression (crushing) after irradiation Age of specimens : 5 months	174 MPa	Strength decreased by 70 %	Important decrease (- 24 %) 62.2 MPa
Compression after corrosion	172 MPa	Strength decreased by 30 %	90 MPa
Porosity to mercury	7.26 %	—	15.71 %
Porosity to water	6 %	17.5 %	21.1 %
Measurement of permeability to water	—	—	$1.1.10^{-19}$ m^2
Measurement of permeability to gas (CEA standard)	8.10^{-20} m^2	—	$3.3.10^{-17}$ m^2
Corrosive test (**)	Good resistance	Good resistance but corrosion is not uniform	Good resistance
Density	2.5 approx.	1.7	2

(*) Crushing : a comparison with other materials is not easy. Crushing resistance can be compared with that of asbestos cement, i.e. 30-50 MPa (calculated value), which corresponds to 80 MPa if the compressive force exerted on the specimen is parallel to the layers and 120 MPa if it is perpendicular.
(**) Corrosive tests : 1. Test of chemical corrosion with acetic acid (norme NF P16-304). 2. Test of chemical corrosion with nitric acid during 50 days.

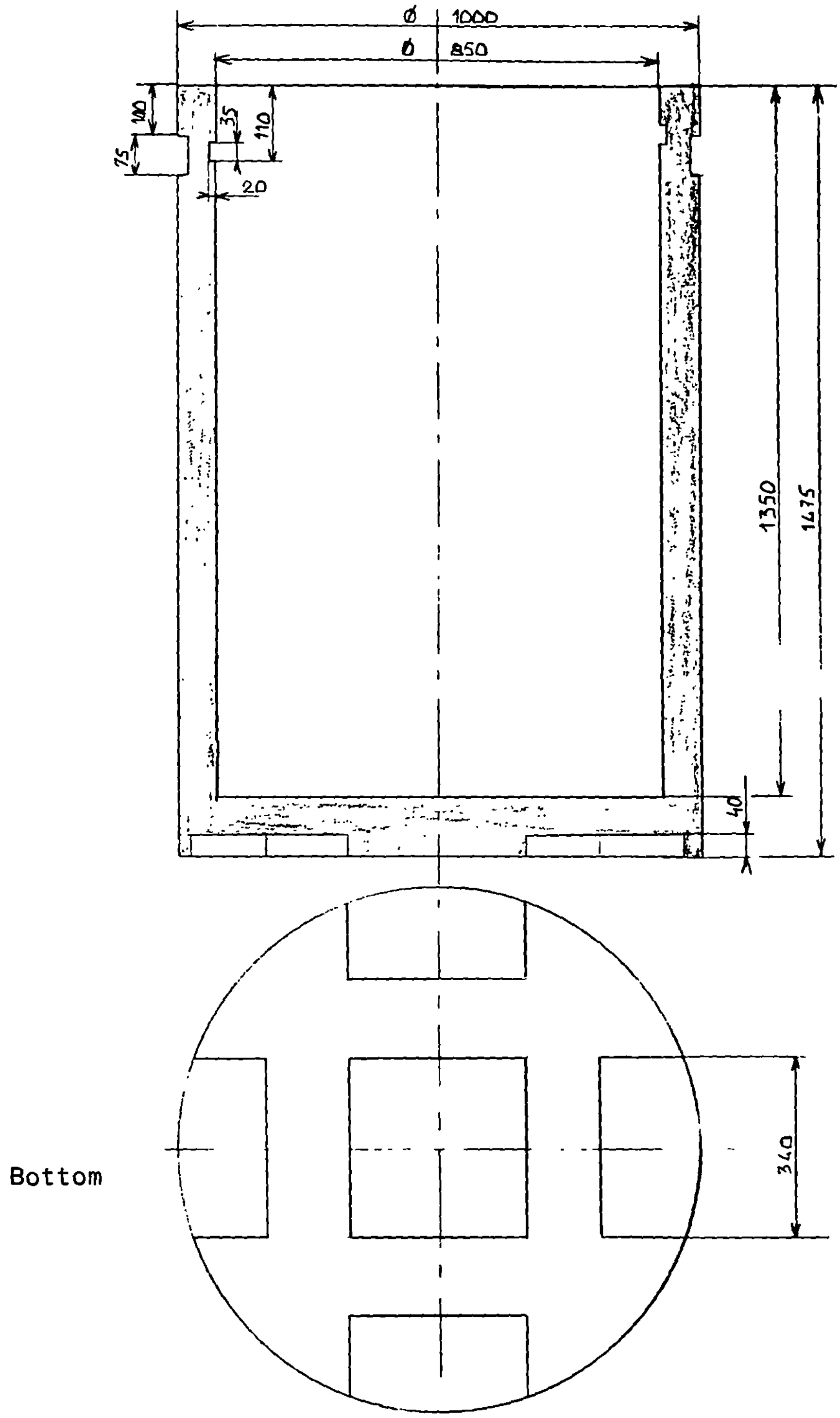


Figure 1: Container design

5.3. Large Waste Containers Cast of Low-level Radioactive Metal Scrap

Contractor: Siempelkamp Giesserei, Krefeld, Germany
Contract N°: FI1D-0047
Working Period: May 1986 - March 1989
Project Leader: H. Deipenau

A. Objectives and Scope

Radioactive waste coming from dismantling of large reactor components should be transported in larger containers than those already used, in order to save cutting work and, consequently, radiation exposure of personnel. The use of radioactive steel for manufacturing transport and disposal containers reduces the volume of waste to be stored, and also metal consumption.

A reference container will be chosen in agreement with requirements for the KONRAD disposal site and suitable for fabrication out of low-level radioactive steel (specific activity up to 74 Bq/g).

In a supplementary agreement concluded in 1988, the initial work programme was extended to studies related to the qualification requirements for waste containers, presently under development at the Physikalisch-Technische Bundesanstalt (PTB). This will be applied to the available prototype container WaCo I, and implemented in the additional working packages B.5. to B.8.

B. Work Programme

B.1. Optimisation of type A cast steel containers, taking into account all relevant requirements for safe transport and disposal in the Konrad mine.

B.2. Design of a prototype container based on the previous optimisation.

B.3. Fabrication of the prototype container with lid and all accessories (plugs, sealing, screws ...) and testing under boundary conditions as given by IAEA and German regulations.

B.4. Establishment of a radiological measurement programme and measuring of all relevant activities occurring before and during fabrication, and on the finished container.

B.5. Design studies and experiments aiming at optimising the prototype as regards leaktightness and the potential for repairs.

B.6. Five experiments on the prototype with a view to assessing leaktightness and mechanical integrity.

B.7. Definition of the original status and backfitting of the prototype with above developed methods.

B.8. Radiological studies into the health physics consequences for given release rate and activity inventories.

C. Progress of Work and Obtained Results

Summary

The "Cast-iron container" was tested successfully according to the IAEA-Transport Regulations for Type-A-Containers /1/ and the preliminary conditions for disposal in the KONRAD-mine /2/.
Methods for the repair of defect corner fittings during production and transportation were developed.
Experiments with different types of seals were carried out to reach a leakage lower than 10 E-5 Pa m^3/s. The graphite seals did not reach the requirements of the preliminary conditions of the KONRAD-mine.

Progress and Results

1. Testing the container (B.6.)

The tests were carried out under the inspection of Physikalisch-Technische Bundesanstalt (PTB), Braunschweig and Bundesanstalt für Materialforschung und -Prüfung (BAM), Berlin.
The following test programme was accepted by PTB and BAM:
- Lifting at the bottom corner fittings with a test load of 40 Mg. (Deformation of the corner fittings was not observed.)
- Stacking test with a test load of 100 Mg. Deformations were not observed.
- Drop test with a height of 0.8 m.
  The container with a maximum load of 20 Mg falls onto a steel plate lying on a concrete foundation.
  At the first impact test the container dropped onto the two cast steel corner fittings; at the second impact test the container falls onto the corner fittings made of cast iron. Some deformations up to 1.7 mm were measured but they did not have a negative influence. The mechanical integrity of the container was proved by a pressure test.
- Side-wall-spreader test results from the conditions of transport in the final depository KONRAD. The test weight was 40 Mg, that means a test weight of 20 Mg for each corner fitting. Deformations were not inspected.
- Fire test (800 ° C, 1 h)
  The preliminary conditions of the depository require that after the mechanical impact (drop 0.8 m) and the following fire the penetration of oxygen to the waste product must be limited. An open flame is not allowed. The temperature curves are shown in figure 1. The integrity of the container was proved by a pressure test. No cracks were detected.

2. Design studies for container repairs (B.5.)
   - Different design studies were made. They showed that the corner fittings can fail during fabrication and transport. Two cast iron corner fittings were machined and replaced by cast steel fitting corners to demonstrate the procedure for repair.
   - Optimizing the container leak-tightness
     The emplacement of the seals requires a greater contact area between lid and container body.

3. Tests for leak-tightness (B.6.)

Different seals were tested: viton, graphite, polymer (NBR). The graphite seal did not fulfil the required leakage rate of less than 10 E-5 Pa m^3/s.

4. Pyrolytic decomposition of powder resins (B.8.)

During the fire test (800° C, 1 h) the inner wall of the container reached temperatures in the range from 250° C to 300° C. Therefore the produced gas volume during pyrolysis of the powder resins in the range from 200° C to 300° C was measured. Specific value: 100 ml/g (Figure 2).

5. Conclusion

The tested container has demonstrated that it is possible to produce a cast iron container out of nuclear scrap fulfilling the conditions as type-A-package according to the IAEA-Regulations for the Safe Transport of Radioactive Material and the preliminary conditions of the final depository KONRAD.

References

/1/ IAEA Safety Series No. 6, 1985
Regulations for the Safe Transport of Radioactive Material
/2/ PTB-Braunschweig, Anforderungen an endzulagernde radioaktive Abfälle (Vorläufige Endlagerungsbedingungen - Schachtanlage KONRAD - Stand November 1986)

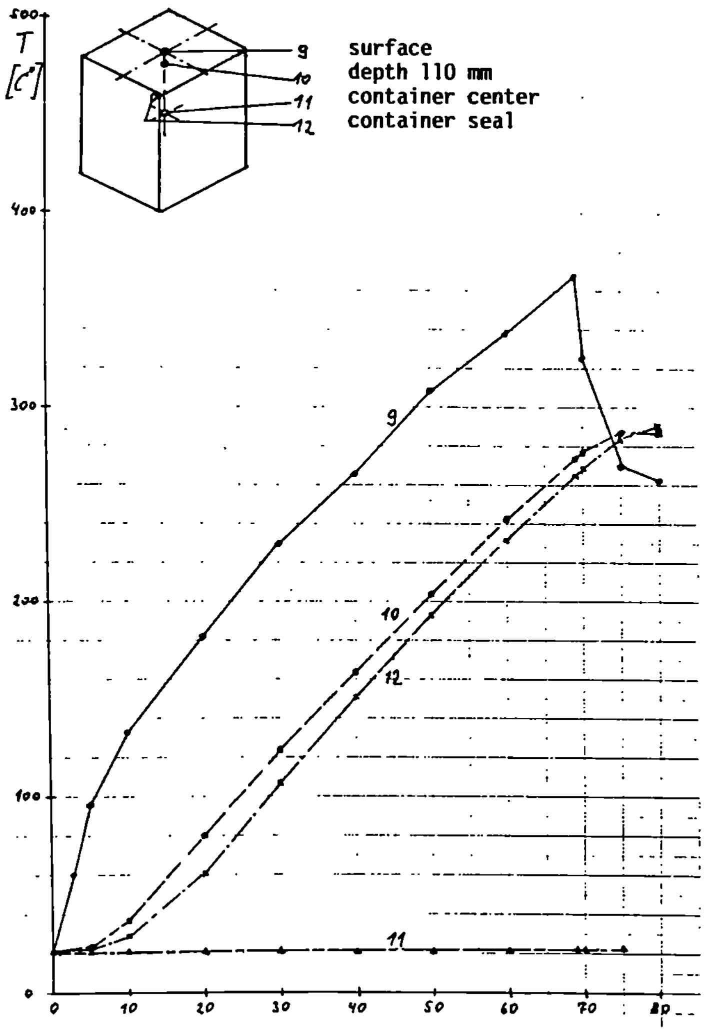


Figure I: Temperature profiles at the seal, position 12, fire duration 1 h, 800° C

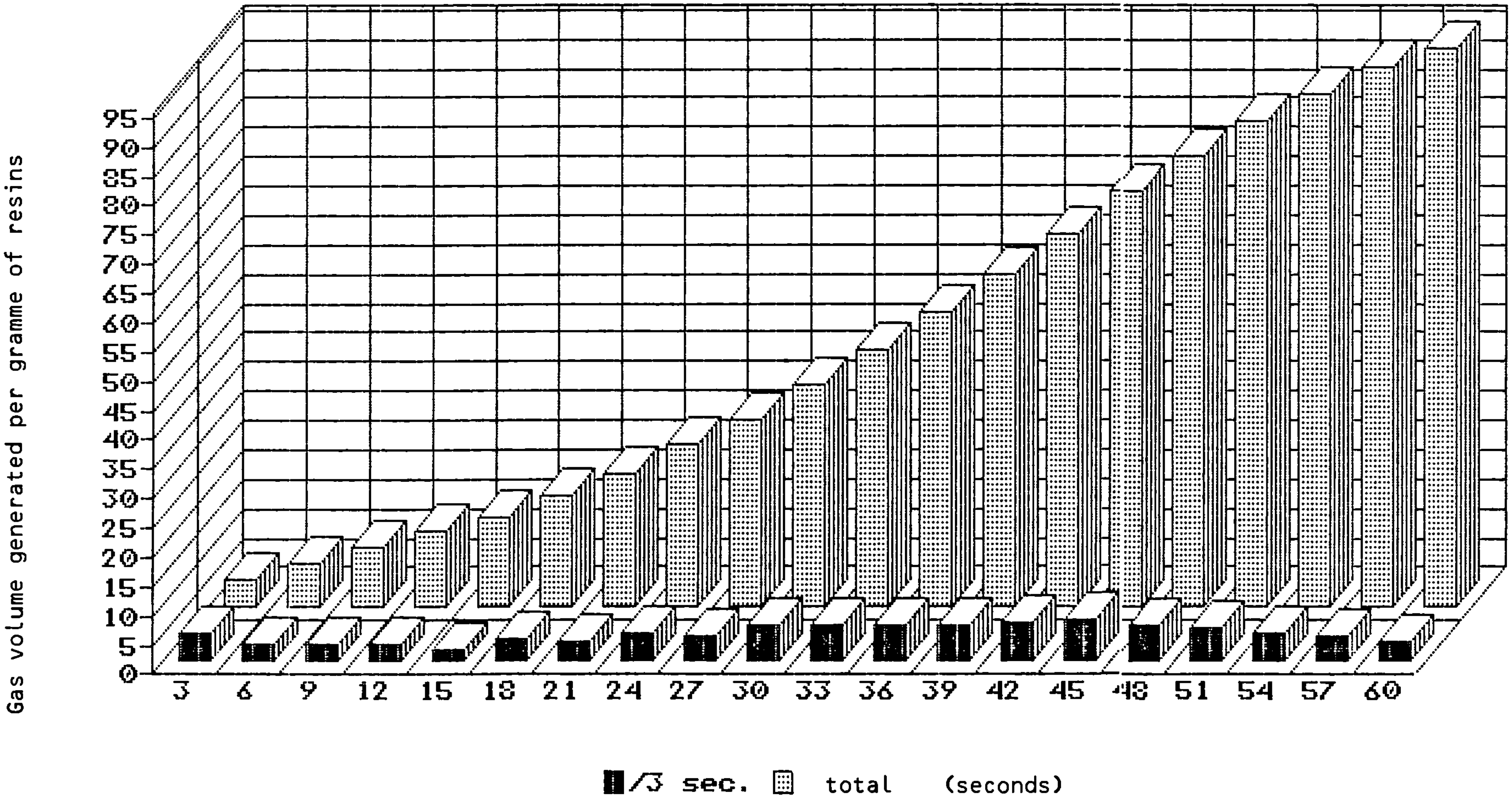


Figure 2: Generation of gas volume at 200° C per 3 sec. and total (sample: p 88 - 7427)

6. PROJECT N°6: ESTIMATION OF THE QUANTITIES OF RADIOACTIVE WASTE ARISING FROM DECOMMISSIONING OF NUCLEAR INSTALLATIONS IN THE COMMUNITY

A. Objective

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

B. Research performed under the 1979-83 programme

The following research work has been performed:

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

C. 1984-88 programme

Research should be performed in the following main areas:

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

D. Programme implementation

Six research contracts relating to Project N°6 were executed in 1988, of which three were completed.

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

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

A. Objectives and Scope

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

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

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

B. Work Programme

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

C. Progress of Work and Obtained Results

The work has been completed, the final report is under publication.

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

Contractor: United Kingdom Atomic Energy Authority, Windscale Nuclear Laboratories, United Kingdom
Contract N°: FI1D-0020
Working Period: December 1984 - June 1988
Project Leader: F.G. Brightman

A. Objectives and Scope

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

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

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

B. Work Programme

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

C. Progress of Work and Obtained Results

Summary

An air-operated remote drilling device has been developed and tested in the laboratory for use in sampling the high-activity loop tubes on WAGR. However, the sampling was postponed into 1989. A review of the 1978 activity inventory, using more modern calculation codes, has shown good general agreement with the original assessments. In a general review for the final report on this contract, simple drilling, abrading and coring methods have been adopted for sampling at a handling cell during the dismantling operations.

Progress and Results

1. Development of drill for loop pressure tube sampling (B.2., B.5.)

An air-operated drill was developed and proved in laboratory trials at AERE, Harwell, for the sampling of WAGR loop tubes - the items of highest specific activity (up to 4 x 10^{13} Bq/gm). During the year, preparations were made for its use in a high pressure loop tube of the Windscale AGR. However, delays in preparation of the safety assessment for the sampling scheme resulted in the postponement of the loop tube sampling into 1989, with revised arrangements for finance. As part of the preparation for this work, the inner 'partition' tube from Large Loop D (one of the loop facilities utilised throughout reactor operation) was successfully removed.

2. Revision of WAGR active inventory, using new calculation codes and new graphite analysis data (B.1., B.3.)

During 1988 a revision of the originally-calculated radioactive inventory for the reactor began. The computer codes used in the 1977/78 assessment (Reference/1/) have been superseded by more widely-applicable methods, including transport codes. The new assessments have taken account of the analyses of steel, concrete and graphite carried out as part of this contract. New graphite analyses were carried out, demonstrating the presence of Tritium in the range 100-300 KBq/gm and Carbon-14 in the range 6-16 KBq/gm. The general conclusions were:

(1) that the original radioactive inventory was adequately assessed; in general recalculations produced results within a factor 2 of the original estimates.

however,

(2) that the wider range of impurity content than expected, especially (for example), for Cobalt and Silver, required that sampling checks would be necessary during dismantling.

The outline inventory, which remains adequate for planning purposes, is shown in Table 1.

3. Design of final sampling/analysis scheme (B.4.)

The use of special methods such as laser microprobe analysis (Reference/2/) has been abandoned as a result of the inadequate sensitivity for determination of Cobalt-59 impurity, especially in the Stainless Steels at WAGR. The final sampling scheme provides for drilled samples from steel components, diamond abrasive paper (rubbing) samples from steel surfaces and cored samples of graphite and concrete to confirm calculated activity inventories. The engineering design of a sampling cell within the WAGR Waste Packaging Building (Reference /3/). Analyses are to be carried out on-site as far as possible.

4. General review for final report (B.2., B.4., B.6., B.7., B.8.)

A review of experience during the contract period (1985-1988) has been made in preparation for the final report. The simplest sampling methods, using drilling, abrading and coring, are to be adopted. Checks in the dominant gamma radioactivity are to be made both on components (such as steel plates) and completely filled disposal boxes using the WESTD code developed as part of the contract. It has been recognised that sampling of the principal materials will be necessary during dismantling, to ensure allocation of wastes to the correct waste classifications within UK definitions.

References

/1/ LUNNING, W H. Decommissioning Nuclear Reactors. ATOM 265, 295-300 (1978).

/2/ BRIGHTMAN F G. WAGR Decommissioning - 5th CEC Progress Report, Project FI.1D.0020 UK(H) (1987).

/3/ ASHCROFT D J. Design Progress. WAGR Decommissioning Newsletter No. 5, Paper 2. (1987).

Table I: Outline activity inventory for WAGR

Material	Mass (t)	Activity (TBq)
Mild steel	761	525
Stainless	89	2194
Graphite	283	65
Concrete	750	3
Total	1883	2787

6.3. Systems for Contamination Measurements on Curved Surfaces

Contractor: Reaktorwartungsdienst und Apparatebau GmbH, Jülich, Germany
Contract N°: FI1D-0021
Working Period: July 1984 - December 1987
Project Leader: B. Hermanns

A. Objectives and Scope

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

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

B. Work Programme

B.1. Development of a basic electronic equipment, suitable for the existing various prototype round and flat detectors with integrated gas supply and analogic part; testing with prototype detectors in the laboratory and under real conditions (KRB-A, Gundremmingen); development of further detectors to complete the range.

B.2. Development of an optimised stationary and portable digitally working unit with background substraction; development of semi-automated or automated measurement systems for irregular surfaces and improvement at laboratory scale.

C. Progress of Work and Obtained Results

The work has been completed, the final report is under publication.

6.4. Optimisation of Measurement Techniques for Very Low-Level Radioactive Material

Contractor: Kraftwerk Union AG, Erlangen, Germany
Contract N°: FI1D-0048
Working Period: September 1986 - March 1989
Project Leader: R. Hoffmann

A. Objectives and Scope

In decommissioning nuclear installations, various types of waste materials which are either free of activity or activated/contaminated have to be released. Unrestricted use of these materials may be permitted if the residual activity concentrations are below limits set by the licensing authority with regard to the radiological risk. In order to prove compliance with these limits, residual activity concentrations have to be measured on every single piece of material, which can be very complicated and time-consuming. The derivation of dependable results is difficult because of the non-ideal conditions usually prevailing and the high degree of precision required.

The aim of this research programme is to assess eligible measuring techniques and to optimise them with respect to accuracy, time and cost.

B. Work Programme

B.1. General basic studies to determine the source-dependent frequency distribution for the nuclide content of radioactive material.
B.2. Compilation of radiologically and metrologically relevant parameters.
B.3. Assessment of parameter importance by measurements on representative geometries using various detectors.
B.4. Procurement/production of representative samples, of volume-related and area-related activity standards and of suitable detectors.
B.5. Experimental determination of detector efficiencies and detection limits for various relevant geometries and nuclides.
B.6. Evaluation of results supported by computation if necessary, in order to set up a guide for selection of the optimum measuring technique accounting for material, measurement time and cost.

C. Progress of Work and Obtained Results

Summary

Determination of detector efficiencies was continued. In order to assess the range of loss of efficiency (and thus the corresponding increase of the limits of detection and/or measurement times), additional geometric arrangements of contaminated surfaces - detectors were introduced. Low energy beta emitters (C 14 and Pm 147) and a pure gamma emitter (Cr 51) were added as well as new types of detectors (plastic scintillation, surface barrier, photodiodes). Depending on the distance between sources and detectors (1-5 cm), a range of efficiency covering 4 orders of magnitude was found.

Alpha self-absorption was determined by spectrometric methods and yielded absorption factors of up to 3 depending on the carrier material.

Progress and Results

1. Fabrication of suitable activity standards (B.4.)

Several new standards were fabricated to be used in different measurements.

1.1. Plane geometry

Solutions of Co 60, Cs 137, Sr 90/Y 90, Am 241, C 14, Pm 147 and Cr 51 were dropped on chromatographic paper of area 20 cm x 20 cm.

1.2. Alpha self-absorption

For alpha-spectrometric measurements, standards of Am 241 on different matrix materials were fabricated, namely
- chromatographic paper
- paper filter
- glass fibre filter
- writing paper
- cotton tissue (lab coats)
- thin layer on steel plates (surface evaporation)

Since the grid ionization chamber to be used for ø 200 mm standards went out of order and could not be repaired in proper time, a second set of standards of ø 50 mm had to be fabricated for use with a conventional surface barrier detector.

1.3. Corner/edge geometry

In order to assess the efficiency loss when measuring activities concentrated in corners or edges, 3 standard sizes in 3 different geometries were fabricated:
(1) Corner area sources with 3 plane standards of area 5 cm x 5 cm in rectangular position
(2) "Point" sources of ø 5 mm placed in corners
(3) "Line" sources of size 20 cm x 0.5 cm placed in edges

Nuclides used were Co 60, Cs 137, Sr 90 and Am 241.

2. Determination of detector efficiencies (B.5.)

For plane geometry, the efficiencies of the following detectors were determined for detector - source distances of 1 cm and 5 cm:
- GM end window counting tubes: Valvo ZP 1401 and Valvo ZP 1430
- Proportional Counters with side entrance windows
  R+A PZ 14 R and R+A PZ 32 F+S

- Plastic scintillation detector: NE 841
- Photodiodes: Hamamatsu S 1790-02
  Hamamatsu S 2551-01
  Hamamatsu S 2575

C 14 cannot be detected by any of these detectors and Cr 51 can only be detected by the proportional counters because of their higher gamma-sensitivity. The Hamamatsu photodiodes, because of their small volume, proved to be sensitive for alpha-particles only.

Determination of alpha (Am 241) self-absorption for different matrix materials was undertaken with Intertechnique/PK 250-100 surface barrier detector; the results shown in figure 1 are a measure of the water absorbent quality of the matrix materials used.

For measurements at corner/edge geometries, the efficiencies of the following detectors were determined:

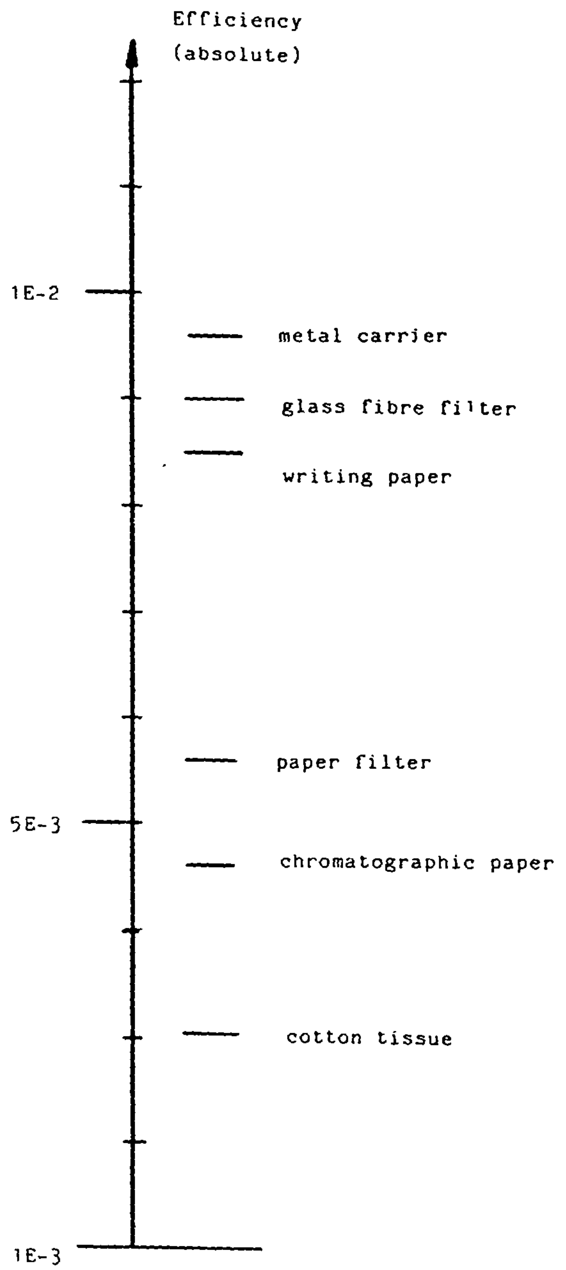
for geometry 1 (area source corner) and 2 (point source corner) loaded with Co 60, Cs 137 and Sr 90:
- Valvo ZP 1430
- Canberra "PIPS" surface barrier, 600 mm^2 active area

at source-detector distances of 1,3 and 5 cm,

for geometry 3 (line source edge) loaded with Am 241:
Hamamatsu diodes, with active areas of
- 10 x 10 mm
- 30 x 3,4 mm
- 20 x 3 mm
- 29,1 x 1,2 mm with and without light-tight metal entrance layer.

Results are summarized in figure 2 and show a range of efficiency of 4 orders of magnitude when normalized to area activity concentration (Bq/cm^2). The corresponding efficiencies normalized to activity (Bq) and the limits of detection are still being evaluated.

These measurements will be supplemented by the inclusion of Pm 147, of minimal source-detector distance ("in contact"), and of detectors Intertechnique, Hamamatsu, Valvo 1401 for geometries (1) and (2), and of K+A, GM, surface barrier detectors for geometry (3).



gure 1: Absolute efficiency for Am-241 on various carrier materials.
(Detector: Intertechnique IPK 250-100, measurement in vacuum).

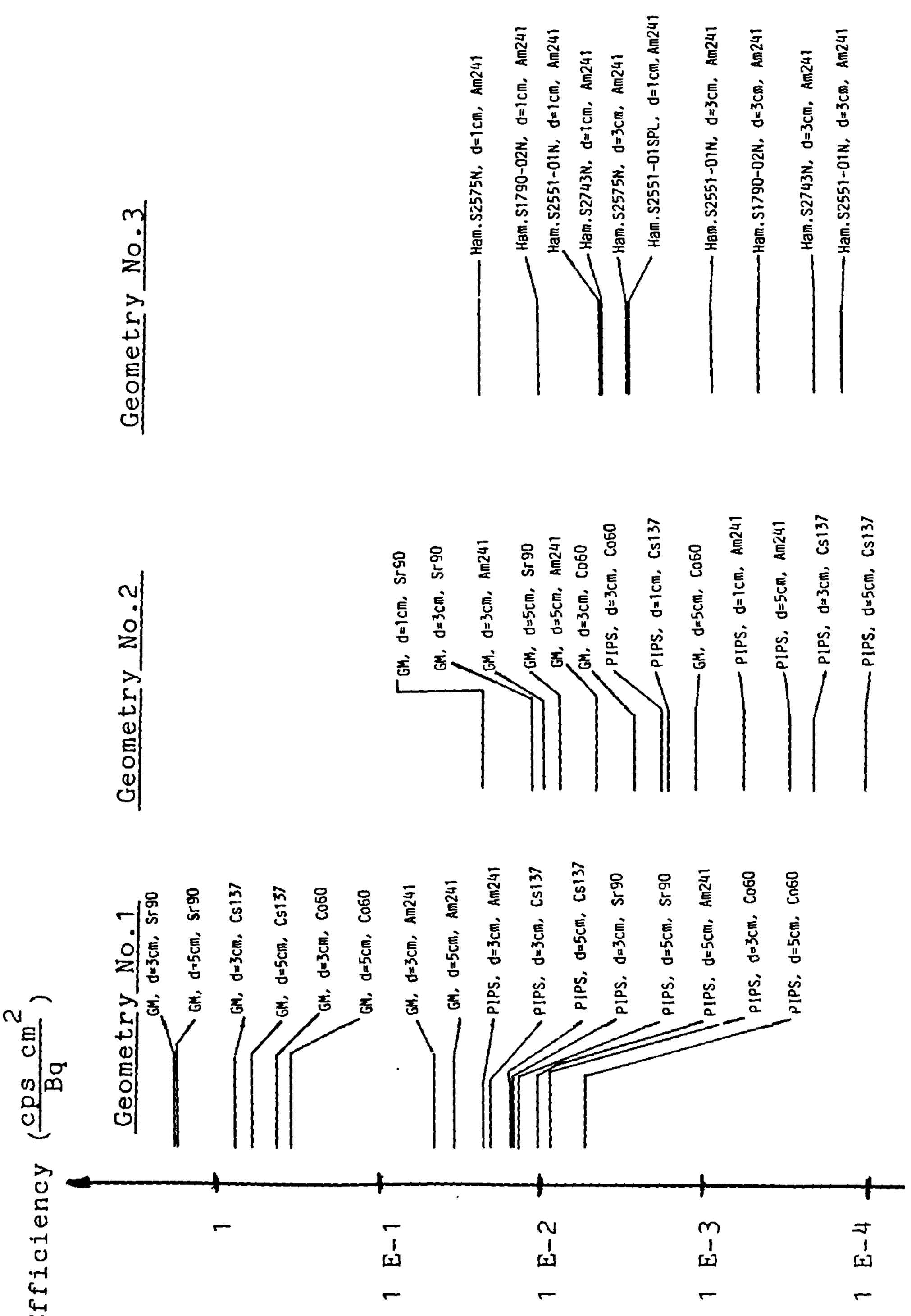


Figure 2: Efficiencies for corner geometries, various detectors and isotopes.

PIPS: Passivated Ion Implanted Silicon detector Canberra SPD-600-40-100-AB/LT.
GM: Geiger-Mueller counting tube Valvo ZP 1430.
Ham.: Hamamatsu photodiodes, various types.

6.5. Monitoring Gamma Radioactivity over Large Land Areas Using Portable Equipment

Contractor: Imperial College of Science and Technology, London, United Kingdom
Contract N°: FI1D-0049
Working Period: May 1986 - December 1988
Project Leader: T.D. MacMahon

A. Objectives and Scope

After a nuclear installation has been decommissioned, the land on which the reactor building and other structures were sited will be available for industrial, residential or agricultural use. Before such a change in use can be accepted, it is essential that the site is monitored to determine whether any residual activity is present in the site material.

Standard sampling techniques that make use of core samples of site material are prohibitively expensive when it comes to detecting localised sources of activity. However, survey techniques, using portable equipment located on the site, can be used to detect localised sources (though only indirectly in the case of alpha and beta emitters).

This research programme is concerned with a survey technique that is used to detect localised sources of gamma emitters. This technique makes use of an adaptive moving array detector system, consisting of an array of detectors, drawn along the surface of the site. Spectra, acquired at periodic intervals, are analysed in real-time to determine the likelihood that a gamma source is present in the region scanned. Scanning is data adaptive - the time spent scanning a region of the site is related to the likelihood that the region contains a gamma source.

The objective of this work programme is to determine the scanning time per unit area for this technique in terms of the intensity of the localised gamma-source, the energy of the emitted gamma-ray, the depth of the source below the site surface, and the composition of the site material.

B. Work Programme

B.1. Determination of the radiation detector system response function in terms of the detector-source geometry, the linear attenuation coefficient of the site material and the distribution of radionuclide activity.

B.2. Construction of stochastic model of the detector system response in terms of the linear attenuation coefficient of the site material, the distribution of radionuclide activity and the stochastic process governing radioactive decay.

B.3. Determination of the linear attenuation coefficient of common site materials as a function of moisture content and gamma-ray energy.

B.4. Development of a computer program to estimate radionuclide activity, with particular attention to the depth of a point source below the site surface.

B.5. Development of a stochastic process for the count rate of a moving detector system, and the construction of a statistic to test the hypothesis that no localised activity source is present in the site.

C. Progress of Work and Obtained Results

Summary

Work carried out during 1988 has enabled the scanning time per unit area for an adaptive moving array detector system to be determined in terms of the source intensity and of the depth of the source below the site surface. A field simulation of the technique has been carried out using a buried ^{137}Cs source and has confirmed detection limit calculations. An improved analysis procedure based on likelihood radio hypothesis testing has been developed and its use demonstrated on computer simulated data.

Progress and Results

1. Activity Detection Limits (B.4) Calculations of activity detection limits of buried radioactive sources have been carried out for two combinations of detector and data reduction technique: firstly, using NaI detectors and integrating all counts in the total spectrum, secondly, using Ge detectors and recording only counts in the photopeak. Results of the calculations showed that the NaI detector total spectrum technique gave lower detection limits for sources buried deeper than 0.35m whilst the Ge detector photopeak technique was superior for depths less than 0.35m. As a practical compromise Ge detectors can be used and both total spectrum and photopeak counts can be recorded simultaneously. Allowing a total scanning time of one week per hectare using five Ge detectors on a boom with 0.3m spacing, with the boom driven forward at 36m h^{-1} and having a transverse oscillation of 0.3m with a period of 15s, the activity detection limits shown in Table I are calculated. Detection limits are inversely proportional to the square-root of the scanning time.

2. Data Analysis (B.5) As the detector system moves across the site being monitored spectra from each of the 5 detectors are recorded in 30s intervals and immediately reduced to both a total spectrum count and a photopeak count. Five such intervals make up one spectral block corresponding to an area of 1.5 x 1.5m^2. The data in each spectral block is then fitted, firstly to a three dimensional Gaussian distribution, and secondly to a uniform planar distribution. The decision as to whether a source is present or not then depends on testing the hypotheses that the data fits the Gaussian distribution or fits the uniform planar distribution. If the hypothesis tests are inconclusive then the spectral block is immediately re-scanned (data adaptive system) and data accumulated until a definite decision can be made.

Two statistical fitting procedures have been investigated: one involving a least squares fitting technique and the second involving maximum likelihood parameter fitting and likelihood ratio hypothesis testing.

In the first case, the hypothesis testing is made by means of the F-test based on the F-distribution: if two statistics χ_1^2 and χ_2^2 are determined, corresponding to the Gaussian and the planar distributions, the ratio of the reduced χ^2 values will be distributed according to an F-distribution. In the statistical literature tables are available which give the probability of exceeding $F(\nu_1, \nu_2)$ when the experiment is repeated (ν_1 and ν_2 are the number of degrees of freedom when fitting the 2 distributions). When the hypothesis test reveals the presence of a source then the Gaussian fit allows estimates to be made of the location (in the x, y plane) and spread of the source. Estimates of the source activity and depth can also be made if both photopeak counts and peak-to-Compton ratios are recorded.

3. Likelihood Ratio Hypothesis Testing The likelihood ratio, λ, is defined as $\lambda = L(P) / L(G)$ where L(P) is the likelihood function for a planar distribution and L(G) that for a Gaussian distribution. It can be shown that

$$-2\ln\lambda = \chi^2(P) - \chi^2(G)$$

so that $-2\ln\lambda$ is expected to have a χ^2 distribution with four degrees of freedom. To demonstrate this a thousand sets of data were generated by computer simulation for 25 grid points assuming a uniform planar distribution with mean 1245 ± 5%, the background level measured in the field simulation. The distribution of $-2\ln\lambda$ values was found to be that expected of a chi-squared distribution with four degrees of freedom. The critical value of $-2\ln\lambda$ above which there is a probability of 5% of rejecting the planar distribution is 9.48.

The same procedure was adopted to study the distribution of $-2\ln\lambda$ values for the Gaussian distribution. The $-2\ln\lambda$ values then follow a non-central χ^2 distribution with the non-centrality dependent on the assumed source activity A_0. For any particular source depth it is possible to vary A_0 until 5% of the $-2\ln\lambda$ values lie below the critical value of 9.48. This is an alternative method for predicting detection limits such that there is a 5% probability of concluding there is no source present when there is one present.

Table I : Activity Detection Limits as a Function of Source Depth

Depth (m)	Detection Limit (Bq)
0.1	1.9×10^{3}
0.5	4.1×10^{5}
1.0	6.7×10^{7}
1.5	1.5×10^{10}
2.0	4.1×10^{12}

6.6. Radioactive Wastes Arising from the Dismantling of a Commercial Fast Breeder Reactor

Contractor: Novatome, Le Plessis Robinson, France
Contract N°: FI1D-0050
Working Period: April 1987 - June 1989
Project Leader: C. Alary

A. Objectives and Scope

The dismantling of a commercial Fast Breeder Reactor (1200-1500 MWe pool type) produces large quantities of radioactive waste differing from that of a commercial Light Water Reactor due to neutron activation and sodium cooling. Material and radioactivity inventories have been determined for commercial LWRs, but not yet for a commercial FBR.

The aim of this research is the establishment of a detailed inventory of the radioactive waste from the dismantling of a French FBR (SPX1), with particular view to the 3500t of sodium and including the primary argon circuit, the secondary argon circuit as well as the auxiliary systems for fuel element handling.

B. Work Programme

B.1. Detailed inventory of the various relevant components to be dismantled, with respect to a large Fast Breeder Reactor (1200-1500 MWe).

B.2. Literature study to obtain waste classification criteria according to decontamination procedures, conditioning, transport and disposal.

B.3. Establishment of a waste classification table to determine appropriate procedures for conditioning, transport and disposal.

B.4. Collection of available data related to the neutron reaction coefficient.

B.5. Determination of the activity levels of each component and corresponding classification, with particular regard to stellite charged parts (high cobalt content). Proposal of a cutting programme for large components.

B.6. Collection of available data on radioactive contamination inside the reactor.

B.7. Determination of the contamination of each component and corresponding classification.

B.8. Balance of the waste according to classification and corresponding range of conditioning options.

B.9. Evaluation of the effect of conceptual options on the waste balance.

B.10 Evaluation of sodium specific criteria for conditioning, decontamination, transport and storage/reuse of components which have worked in sodium.

C. Progress of Work and Obtained Results

Summary

The tasks carried out during the year 1988 pertain to the B.4 and B.5 working steps as described in the work programme.

In the frame of the B.4 working step, the nuclear data required by the calculation of the activation sources were collected. The 113 neutron fluxes were computed throughout the reactor pit, the reaction rates were of interest too.

In the B.5 working step, the calculation of the specific and of the total activities of the main components and structures has been undertaken on the basis of the data collected in the B.1 step: materials, isotopic breakdown and so on.

Progress and results

1. Calculation of the activation rates (B.4)

In the prospect of Fast Breeder Reactor dismantling, because of the delay between the reactor shutdown and the start-up of the dismantling, the only radioactive products of concern are the long lived radioactive products. In this survey more than forty neutron capture reactions were identified as producing such long lived radiation products, i.e. with half lives longer than one year. This inventory is based on the isotopic composition (impurities included) of the materials contained in the reactor pit.

The nuclear data: cross sections, half lives, required by further calculations were collected too, see table 1. Most cross sections of interest do not exist in the multigroup libraries in use in nuclear engineering. So our calculations rely on references 1, 2 and 3.

The multigroup neutron fluxes were calculated by solving the neutron transport equation by means of the computer code CANISN in connection to the neutron cross section library BABEL (113 groups).

2. Calculation of the component and structure activities (B.5)

The calculation of activities is underway. Owing to their size, the steep neutron flux variations, components and structures are divided into lumps of suitable dimensions to limit the range of the specific activity over their volume. At the moment, the results which have been worked out, regard the following items:

- above core structure
- diagrid
- intermediate heat exchanger
- control rod mechanism

see for instance table II.

References

/1/ Gerhard Erdtmann - Neutron activation tables

/2/ BNL 325 second edition - Neutron cross sections - May 1984

/3/ R. PANNETIER - Vademecum de technicien nucléaire - tome II - table des caractéristiques - seconde édition.

Table I: NUCLEAR REACTIONS OF ACTIVATION

(HALF-LIVES OF THE PRODUCTS > 1YEAR)

REACTION	TARGET	ABUNDANCE	σ	PRODUCT	HALF-LIFE
		(%)	I		
(nth, γ)	C 13	1.11	1.1mb	C 14	5736a
			1.5mb		
(nth, p)	N 14	99,635	1,8b	C 14	5736a
(nth, α)	O 17	0.039	235mb	C 14	5736a
			105mb		
(nth, γ)	Cl 35	75.77	43b	Cl 36	3 E5a
			17b		
(n,2n)	Cl 37	24,23	0,120mb	Cl 36	3 E5a
(nth, γ)	K 39	93.3	1.96b	K 40	1 28 E9a
(nth, α)	"	"	4,3mb	Cl 36	3 E5a
(n,p)	"	"	20mb	A 39	269a
(n, α)	"	"	13mb	Cl 36	3 E5a
(n,2n)	K 41	6,7	0,160mb	K 40	1.28 E9a
(n,p)	Ca 40	96,94	77mb	K 40	1.28 E9a
(nth, γ)			0,4b	Ca 41	1.3 E5a
			0,18b		
(n, α)	Ca 42	0,65	2,7mb	A 39	269a
(n,2n)	"	"	0,04mb	Ca 41	1.3 E5a
(nth, γ)	Fe 54	5,8	2,25b	Fe 55	2,7a
			1.2b		
(n,2n)	Fe 56	91,7	0,068mb	Fe 55	2.7a
(nth, γ)	Co 59	100	37b	Co 60	5,27a
			77b		
(n,p)	Ni 60	26,23	4,4mb	Co 60	5.27a
(n,2n)	"	"	0,060mb	Ni 59	7.5 E4a
(nth, γ)	Ni 62	3,66	14,5b	Ni 63	100a
			7b		
(n,2n)	Ni 64	1,08	0.36mb	Ni 63	100a
(n, α)	Cu 63	69.1	0.56mb	Co 60	5,27a
(nth, γ)	Se 78	23,5	430mb	Se 79	65000a
			4.7b		
(n,p)	"	"	< 1mb	As 78	17,5a
(n,2n)	Se 80	50	0,340mb	Se 79	65000a
(nth, γ)	Nb 93	100	1.2b	Nb 94	2 E4a
			9b		
(n,2n)	"	"	1,1mb	Nb 92	2 E7a

Table1: NUCLEAR REACTIONS OF ACTIVATION

(HALF-LIVES OF THE PRODUCTS > 1YEAR)

continued

REACTION	TARGET	ABUNDANCE	σ	PRODUCT	HALF-LIFE
		(%)	|		
(n,n')	Nb 93	100	<100mb	Nb 93m	13,6a
(nth, γ)	Ag 107	51,83	4,5b	Ag 108m	127a
(nth, γ)	Ba 132	0,095	8,5b	Ba 133	10,4a
			100b		
(n,p)	Ba 134	2,42	0,084mb	Cs 134	2,06a
(n,p)	Ba 135	6,59	0,095mb	Cs 135	2.3 E6a
(n,p)	Ba 137	11,32	0,022mb	Cs 137	30,1a
				Ba 137m	
(nth, γ)	Eu 151	47,8	5900b	Eu 152	12,4a
			5564b		
(nth, γ)	Eu 153	52,2	390b	Eu 154	8,5a
			1635b		
(nth, γ)	Eu 154	artificiel	1500b	Eu 155	4,65a
			1240b		
(n,p)	Eu 151	47,8	0,049mb	Sm 151	93a
(n,2n)	"	"	3,7mb	Eu 150m	35a
(n,2n)	Eu 153	52,2	2,1mb	Eu 152m	9,3h
				Eu 152	12,4a
(n,2n)	Ta 180	0,012	17mb	Ta 179	1,65a
(n,p)	Pb 204	1,4	0,019mb	Tl 204	3,78a
(n,2n)	Pb 206	24,1	4,1mb	Pb 205	1.4 E7a
(nth, γ)	Ni 58	67,88	4.9b	Ni59	7.5E4a
			2.4b		

Table II : Maximum ponctual activity of some components and structures

Component or Structure	Material	Maximum ponctual activity Bq/cm^3
spherical seat	stellite	0.3348 E+13
diagrid support	stellite	0.8418 E+09
	Z2 CN 18-10	0.1550 E+08
anti-convection device	Z2 CN 18-10	0.5149 E+06
internal vessel	Z2 CND 18-12	0.1527 E+06
primary pump	Z2 CND 18-12	0.2327 E+04
primary pump bolts and nuts	Z6 NCTDV 25-15	0.2434 E+04
pump and diagrid interconnection pipe	Z2 CN 18-10	0.1481 E+11
lateral neutron shield support	Z2 CN 18-10	0.2205 E+11
integrated purification system	Z2 CND 17-12	0.1588 E+06
concrete under vessel	concrete	0.1348 E+00
main vessel	Z2 CND 18-12	0.6782 E+03
safety vessel	Z2 CN 18-10	0.2661 E+03
lower diagrid plate	Z2 CN 18-10	0.2283 E+08
upper diagrid plate	Z2 CN 18-10	0.1446 E+11
core catcher	Z2 CND 18-12	0.1370 E+06
core cover plug structures in Na	Z2 CND 18-12	0.5977 E+08
I.H.X.	Z2 CND 18-12	0.2327 E+04
I.H.X bolts and nuts	Z6 NCTDV 25-15	0.2434 E+04
emergency residual heat exchanger	Z2 CN 18-10	0.2751 E+06

6.7. Methodology for Assessing Suitable Systems for Management of Reactor Decommissioning Wastes

Contractor: National Radiological Protection Board, Chilton, Didcot, United Kingdom
Contract N°: FI1D-0051
Working Period: January 1987 - April 1989
Project Leader: J. Davis

A. Objectives and Scope

The objective of this research is to conduct a broadly-based study in which a methodology is demonstrated for assessing as many as possible of the factors relevant to decisions on the management of all the major types of waste arising in Stages 2 and 3 of decommissioning. The methodology will include the use of decision-aiding techniques and will show how quantitative and qualitative factors can be weighed against each other, in order to provide guidance to those responsible for taking decisions. The methodology will be illustrated by examples based on several different types of reactor.

B. Work Programme

B.1. Definition and description of factors relevant to waste management decisions including radiological hazard, socio-political factors and cost effectiveness.
B.2. Definition of all radioactive wastes arising from Stage 2 and 3 decommissioning of typical PWRs, AGRs and Magnox reactors.
B.3. Definition of suitable management systems for immediate and delayed dismantling and for various disposal and recycling/reuse routes.
B.4. Assessment of the radiological impact of each management system to workers and members of the public.
B.5. Assessment of other environmental factors associated with each management system, including non-radioactive pollutants, transport, conventional safety and effects on resources.
B.6. Assessment of social and political factors associated with each management system, such as public acceptability and international concerns .
B.7. Assessment of the cost of waste treatment, transport and disposal, associated with each management system.
B.8. Demonstration of the methodology on the three reactor types to identify the dominating decision-aiding factors and the priorities for further work.

C. Progress of Work and Obtained Results

Summary

During this period waste management systems have been defined according to sequential steps in the decommissioning process. The radiological impact on workers and public at each step has been assessed, partly from information given in other studies, partly by predictive modelling of the long term consequences of radioactive waste disposal. Environmental, social and political factors have been considered and included where possible for demonstration purposes. The costs of each step have been assessed, based on information given in the literature. A multi-attribute utility analysis program is being used as a decision-aiding tool to demonstrate how quantitative and qualitative factors can be included in the decision-making process.

Progress and Results

1. Waste management systems (B.3.)

For this study an overall waste management system is assumed to follow the following steps sequentially: dismantling, storage, recycling, conditioning/packaging, transport and disposal. Options exist at each step giving a great number of possible permutations. These have been simplied as follows: dismantling may occur immediately or after some delay period; interim storage may vary only according to its duration; recycle and re-use may be included for steels with fixed assumptions on partitioning of activity and exposure of public on re-use; minimal conditioning and a single broadly-representative package design for all wastes arising at stages 2 and 3 of decommissioning are assumed; the options for disposal are shallow-land burial in a fully-engineered facility, deep geological disposal in clay, geological disposal offshore which is accessed by a coastal tunnel and deep-ocean disposal.

2. Radiological impact assessment (B.4.)

The radiological impact arising at each step in the management of decommissioning wastes has been assessed. Radiological impact is taken here to include maximum annual individual doses and risks and the collective dose integrated over various time periods. Doses to workers during decommissioning operations could be derived from the external dose rate at a range of times and at various locations around the reactor. However, shielding and remote systems are expected to be used to reduce worker doses particularly when short delay periods are considered. If individual dose limits are satisfied, the collective dose to workers may not be reduced if the cost of the equipment needed to reduce it outweighs the detriment avoided. This is an ALARA decision, the outcome of which cannot be foreseen before the event. Given this unknown, the collective dose to workers is assumed to correspond to a site target of 2 man Sv y^{-1} for immediate dismantling, and to be directly related to the fall in external dose rate at 100 years. Off-site worker doses are assumed to comply with IAEA transport regulations and have been taken from published estimates where possible.

Doses to the public during operations are on the whole predicted to be very low indeed. The radiological impact on the public in the long-term after disposal of wastes has been calculated for two release mechanisms: 1) The waste is gradually degraded by natural processes and radionuclides dissolve and gradually disperse back into man's environment. Rates of release from packages in different disposal environments were derived from data on the corrosion rate of steel and the rate of disintegration of concrete. Linear compartment models were then used to predict the behaviour of radioactivity in the environment; 2) Direct human intrusion into a disposal facility was considered and the risk to the intruder from exposure to radioactive material assessed, taking into account the low probability of occurrence..

ICRP Publication 30 dosimetry, as modified by NRPB GS7, has been adopted throughout.

3. Assessment of environmental, social and political factors (B.5.,B.6.)

Environmental, social and political factors have been considered qualitatively, using information provided in recent studies. A comprehensive assessment would include features which will vary depending on the particular site in question. Nevertheless, several key environmental and social factors are being included quantitatively, as far as possible, for demonstration purposes. Political factors are less predictable and can only be included speculatively. A range of political impacts is being included so that the nature and extent of their influence on the decision can be demonstrated.

4. Assessment of costs (B.7.)

Costs associated with the various steps in stage 2 and 3 decommissioning have been assessed from information provided in the literature. The key issues surrounding the estimation of costs of immediate versus delayed decommissioning are discounting and surveillance costs. Savings arising from the reduced need for expensive remote systems at long delay periods are difficult to account for because of the implicit uncertainty in cost versus worker dose decisions. For this demonstration assessment a choice of either no discounting or discounting at a typical rate is assumed. Site surveillance costs have been included for the 90 year intervening period.

5. Demonstration of methodology (B.8.)

A multi-attribute utility analysis program is being used as a decision-aiding tool. This allows comparisions to be made between a number of waste management options, over a wide range of impacts, i.e. radiological impacts, costs, environmental, social and political impacts. Several waste management systems are being examined incorporating data from B.5, B.6 and B.7 and scaling up the unit radiological impact results (B.4) according to inventories from several reactor types (B.2). This study concentrates on the PWR as the reactor most widely in use. However, comparison of waste management options for Magnox and AGR reactors will also be demonstrated.

6.8. Radiological Evaluation of Releasing Very Low-Level Radioactive Copper and Aluminium

Contractor: Commissariat à l'Energie Atomique, CEN Fontenay-aux-Roses, France
Contract N°: FI1D-0052
Working Period: June 1986 -December 1988
Project Leader: H. Garbay

A. Objectives and Scope

"De minimis" limits are being established in various countries (Germany, Italy, France, UK) and by the CEC for the recycling of steel, and by the IAEA for disposal/incineration of waste. Taking into account the important exchanges of metal between EC countries, it seems necessary to obtain common "de minimis" values also for other materials arising in large quantities in the dismantling of nuclear installations, particularly for copper and aluminium.

This study comprises all possible recycling modes, as well as the discharge to the environment, of very low-level copper and aluminium coming from decommissioning and refurbishing of nuclear installations.

B. Work Programme

B.1. Study and definition of relevant basic data relating to the recycling of copper and aluminium, including industrial use, transformation, work place characteristics, type and quantities of impurities.

B.2. Compilation and synthesis of the reviewed data and calculation of the radiological consequences due to recycling, reuse and refuse disposal.

B.3. Determination of activity limits applicable to copper and aluminium and comparison with limits under definition for steel, concrete and technological waste.

B.4. Evaluation and comparison of the costs of the two management modes, i.e., first, conditioning, transport and storage of radioactive waste, and second, conditioning, transport and recycling of non-radioactive waste.

C. Progress of Work and Obtained Results

Summary

The work carried out during the year 1988 was mainly focused on experiments of activity dispersion by metal fusion in semi-industrial scale, and on experiments of aluminium recycling and copper recycling in industrial scale (technology, products and by-products, masses, partition of chemical elements).

Collection of data on quantities of low radioactive metal scrap likely to arise on the metal's market were also conducted.

Progress and Results

The data collected concern quantities of electrical cables in reactor plants. French and German data have been compared. The quantities of cables may vary a lot according to the country and to the different plants. One reason is that the manufacture of cables can be different from one country to another and standards are different.

However, a rough evaluation would give between 450 tonnes and 650 tonnes of electrical cables in the reactor building, the auxiliary building and the storage building in French 1300 MWe PWR plants ; considering the probability of occurence of contamination hitches during the reactor's life it seems reasonnable to consider that up to 300 to 400 t. of low-contaminated cables would occur at the dismantling period.

In German 1300 MWe PWR plants, the evaluation of total cables of reactor building and auxiliary building is between 280 tonnes and 600 tonnes.

The main work performed this year consisted in experiments to evaluate the partitioning of elements during fusion of copper and aluminium.

Two experiments have been realised in industrial plants.

The total dust concentration during melting and the fraction of inhaled particles (13 µm) were quantified ; the results are shown in the following table.

The partition of metal and by-products was also quantified ; production of by-products is presented in the following table for one tonne of metal produced.

The partition of some elements into product and by-products was also measured to allow the calculations of radiological consequences in the industrial plants.

Another experiment on semi-industrial scale was realised with enriched uranium contaminated aluminium alloy. This experience has shown that partition of metal and by-products was different than on industrial scale.

Six melt experiments were realised with scraps at different levels of specific activity comprised between 2.7 $Bq.g^{-1}$ and 60 $Bq.g^{-1}$.

While melting the two high specific activity lots, we found that uranium arised mainly into slags.

Whatever is the initial contamination, the specific activity in the ingot seems to be constant at a level of 2 $Bq.g^{-1}$.

Conclusion

On the base of these results and by evaluating other parameters, radiological consequences will be calculated for the recycling of aluminium and copper and their reuses and will be presented in the final report.

Table I: Dust concentrations during melting and casting.

		Aluminium	Copper
Melting	Total mass concentration ($mg.m^{-3}$)	4.5	6.85
	Fraction of inhaled particles	0.85	0.4
Casting	Total mass concentration ($mg.m^{-3}$)	1.6	2.4
	Fraction of inhaled particles	0.6	0.8

Table II: Production of by-products for one tonne of metal produced.

	Aluminium	Copper
Slags (kg) for 1 t of metal	300	800
Dust (kg) for 1 t of metal	3	17

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

A. Objective

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

B. Research performed under the 1979-83 programme

Activities on the following subjects are in progress:

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

C. 1984-88 programme

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

D. Programme implementation

Five research contracts relating to Project N°7 were executed in 1988, of which two were completed.

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

Contractor: ENEA/Trisaia Energy Research Centre, Policoro, Italy
Contract N°: FI1D-0022
Working Period: July 1985 - June 1989
Project Leader: T. Candelieri

A. Objectives and Scope

The ITREC plant was originally conceived and built as an integrated unit for reprocessing and refabrication of fuel elements. Fuel elements containing uranium and thorium are processed without separation of the fission products. Moreover, the processed material contains Th-228, a strong gamma emitter. The refabrication is, therefore, carried out in a cell fitted with adequate shielding, using remote-operated equipment and techniques. All equipment belonging to the main chemical process is installed in modular units, which provide for remote-controlled removal after appropriate decontamination of the individual unit (rack) for maintenance and modification of equipment (Rack Removal System). This system allows the remote transfer of process equipment from the hot cell to the decontamination cell and its decontamination to levels low enough to permit safe access for the workers of maintenance operations.
The ITREC plant has been operated under hot conditions from 1975 to 1979.

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

B. Work Programme

B.1. Design and construction of cutting equipment for dismantling the centrifugal contactors of Rack 6 bis in the ITREC plant.

B.2. External and internal decontamination of Rack 6 or 6 bis, with a first operation in the hot cell, followed by complete cleaning in the decontamination cell.

B.3. Testing of dismantling by remote cutting of the centrifugal contactors with the highest contamination.

B.4. Design and construction of a storage container for the conditioned dismantled centrifugal contactors.

C. Progress of Work and Obtained Results

Summary

The definite design of a dismantling device has been developed on the basis of a previous feasibility study. The work task has been finalized and applied to remove the centrifugal contactor from Rack 6 bis in order to reduce the radiation exposure of the plant maintenance staff.
The definite design is adequate to accomplish operations remotely.
Cutting of the connection pipe between centrifugal contactors will be facilitated because the particular outline module is a simple repetitive geometrical design.

In 1986, during the preliminary operations for the plant restart with the centrifugal contactors (Rack 6 bis), the decontamination and transfer of Rack 6 were performed. The decontaminating solutions used are stored in the High Level Waste and/or Low Level Waste tanks of the plant. After decontamination, Rack 6 was stored in the corridor area, while Rack 6 bis, equipped with centrifugal contactors, was installed in its place.

Progress and Results

1. Dismantling device (B.1.)

The definite design of the dismantling device has been completed in May 1986.

The project has been realized with the collaboration of SNIA TECHINT SpA, Rome. Laboratory tests were performed with two cutting methods: shears and circular saw; the results pointed out more reliability for shears. Therefore, the final design has two shears, i.e. one to cut horizontal tubes and the second to cut the vertical ones. Each shear is able to cut tubes of any dimension installed on the Rack 6 bis.

2. Decontamination of Rack 6 and 6 bis (B.2.)

The Rack 6 and 6 bis including the evaporator and the centrifugal contactor of the battery have been decontaminated.

3. Construction of dismantling machine (B.3.)

The machine construction has been completed in December 1988 and transferred in the ITREC plant of Trisaia Center.

The Figure 1 shows the frame structure (a) along with the guiding rail (b) supporting the cutting unit (c) which is moved to face each centrifugal contactor of the battery.

In Figure 2 is shown the cutting unit during the cold tests operations where is evidenced:
- hydraulic pistons driving horizontally and vertically the shears (d);
- pneumatic screwers (e) for bolts removing;
- electrical motor reducer (f) for cutting unit motion on the rail;
- full scale mock-up structure (g);
- lowest TV camera (h)

The cutting unit is remotely operated by a control console (figure 1). located outside the decontamination cell close to the shielded window through which the operators follow directly the work.
Television is used as an auxiliary viewing system to help performance handling tasks in hard-to-see places. The television system consists of two cameras with zoom installed on the dismantling device and monitor (m) located on the control consolle. With this system, the capability is available for doing close alignments work as well as general surveillance.

The full scale mock-up allows the cold tests of the dismantling facility. In order to simplify the test, the mock-up is provided only with the contactors most difficult to remove, since the most critical positions for dismantling operations, due to the outline of the cell and handling facilities installed in it, are those located at the extremities of the supporting (the lowest and highest).

The dismantled equipment is transferred to a waste container and pneumatically released from a graple purposely designed and constructed.

The mock up will allow testing of the whole operation, including the handling system of the cell, training operations and maintenance personnel.

4. Remote cutting tests of Rack 6 bis (B.3.)

This step will be carried out in May 1989.

5. Design and construction of storage container for the conditioned dismantled centrifugal contactors (B.4.)

The final design has been completed according to safety rules of ENEA Security Department and the storage container is under construction.

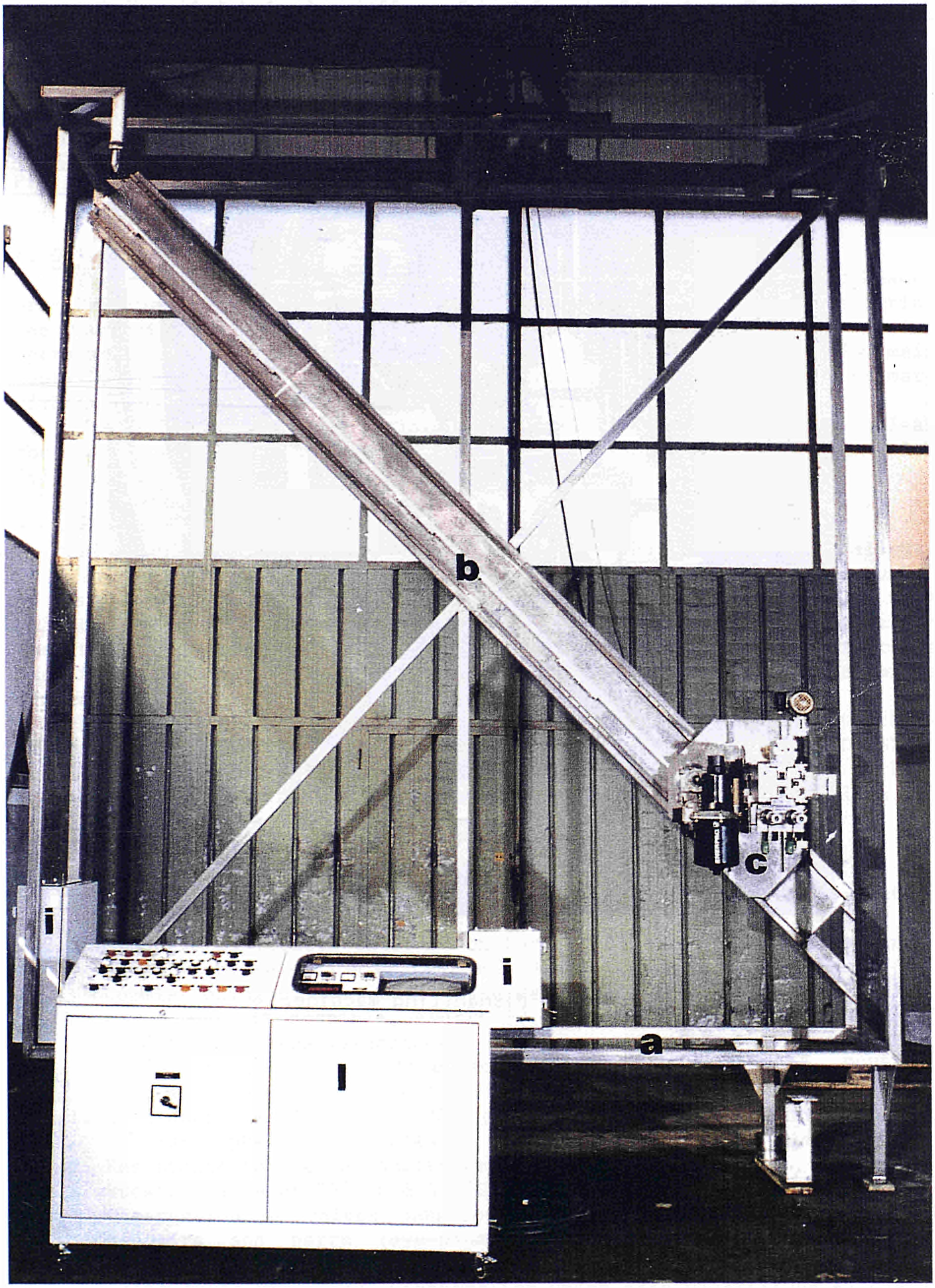


Figure 1: Global view of dismantling machine.

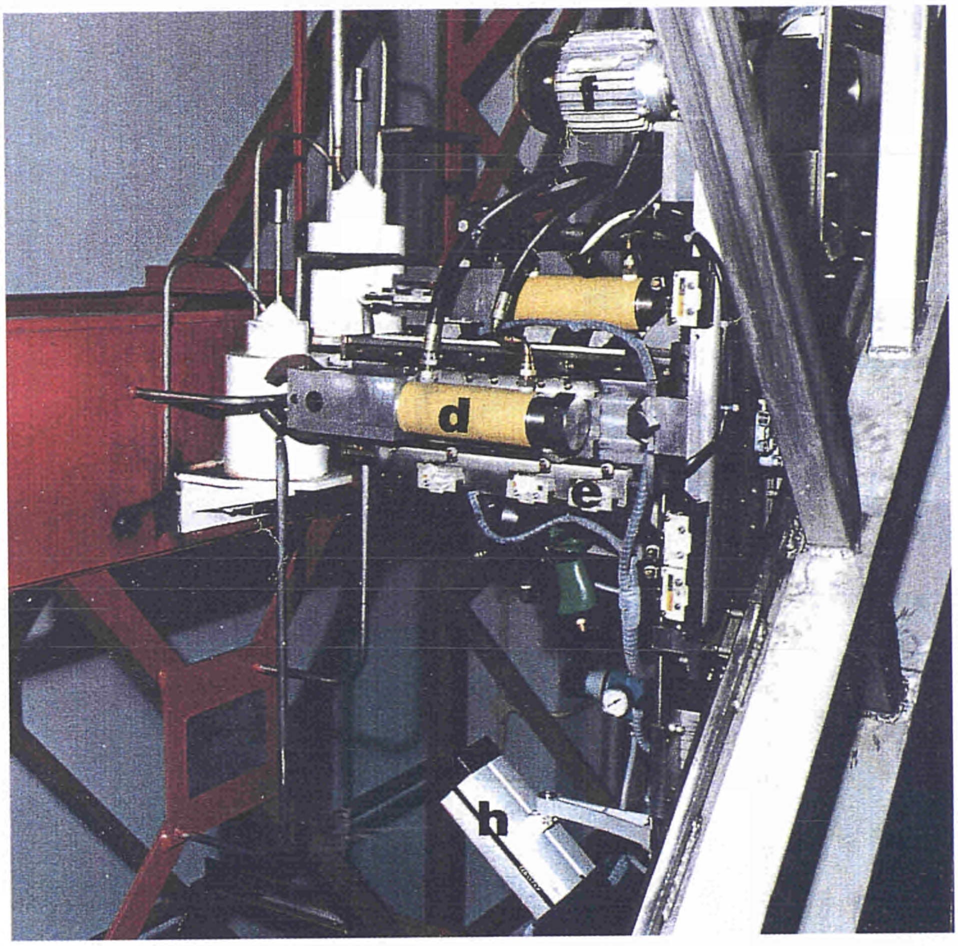


Figure 2: Details of dismantling machine.

7.2. Testing of Cobalt-free Alloys for Valve Applications Using a Special Test Loop

Contractors: Framatome & Cie, Paris and Commissariat à l'Energie Atomique, CEN Saclay, France
Contract N°: FI1D-0053
Working Period: July 1986 - June 1989
Project Leader: C. Benhamou

A. Objectives and Scope

The radiation level around the components of Pressurised Water Reactors (PWR) particularly governs the radiation exposure of the workers during the periodic maintenance operations, as well as during decommissioning operations. Since the activation product cobalt-60 is one of the main contributions to this exposure, the use of cobalt alloys in the primary circuit should be avoided as far as possible.

The alloys likely to replace cobalt alloys mainly used in nuclear cocks and valves, e.g. Stellite Grade 6 and Grade 12, must comply with following criteria:

- good weldability;
- hardness equivalent to that of cobalt alloys;
- resistance to friction and wear equivalent to that of the cobalt alloys.

In the past few years, Framatome, jointly with CEA, assessed a number of hard cobalt-free alloys considered as promising; two of them were selected: Cenium Z 20 and Colmonoy 5. A third alloy will be considered: Everit 50, selected as a result of the first Community research programme (see final report EUR 9865), if information necessary to its realisation is available.

This research aims at establishing the performances of these three alloys, comparatively to Stellite Grade 6, on valves mounted on DOUBLEAU loop of CEA, operated in conditions as close as possible to PWR working conditions. The selected valves are globe-valves and swing check-valves.

The research is led by Framatome.

B. Work Programme

B.1. Basic study including design and specifications of the selected valves (Framatome).

B.2. Synopsis of results obtained on hard cobalt-free alloys in order to justify the selection of Cenium Z 20 and Colmonoy 5 (Framatome).

B.3. Commissioning of the valves with deposits of Cenium Z 20 and Colmonoy 5 and, if possible, Everit 50, compared with Stellite Grade 6 hard-faced valves (Framatome).

B.4. Implementation of the selected hard-faced valves in the DOUBLEAU loop and deposits testing (CEA):

B.4.1. Endurance tests under PWR primary circuit conditions (320°C, 160 bars, pH=7, 1500 cycles).

B.4.2. Resistance to thermal shocks tests (100°C and 250°C).

B.4.3. Erosion tests at 70°C and 320°C during 10 minutes.

B.5. Observation of valves behaviour during tests and examination of deposits and parts (dye-penetrant testing, internal tightness, surface state) after each series of tests (CEA).

B.6. Conclusions and recommendations for using hard cobalt-free alloys as deposit in the valves (Framatome).

C. Progress of work and obtained results :

Summary

Framatome has ordered three series of three different valves which are globe valves, valve body bith bonnet and swing check valve; Alloys used for hardfaced surfaces are Stellite (Cobalt based alloy) for the first series and Colmonoy (Nickel based alloy) and Cenium (Ni Cr Fe based alloy) as cobalt free alloy respectively for the second and third series of valves.

The last four valves identified as to be manufactured at the end of 1987 have been now delivered to the loop operator CEA : the first two with Colmonoy at the end of first semester 1988 and the last two with Cenium at the end of year 1988; referring to problems met in performing the hardfacing deposits, special investigation of welding procedures was necessary tc be completed by the valve supplier for manufacturing of the last two valves.

With respect to PWR working conditions the performance of Cobalt free alloys has to be evaluated comparatively to Cobalt based alloys by testing successively the valves with Stellite, the valves with Colmonoy and the valves with Cenium. FRAMATOME assessed in a document /1/ the principles of such a test program on valves mounted on CEA DOUBLEAU loop; the detailed loop tests to be performed by CEA includes for each series of valves (same hard alloy) Endurance tests, Thermal Shock tests and Erosion tests (Task B.4.); the instructions for the driving of loop tests as well as for the associated non destructive examinations of valves have been reported by CEA on a test procedure /2/.

The full loop test program has been achieved on the branch fitted with Stellite valves during year 1988 /3/; six months instead of four were necessary for driving these tests; these two months delay was due to problem met by CEA in exercizing the globe valve after 1000 cycles during Endurance Tests and in controlling the globe valve leakage after the thermal shock tests; the valve has been dismantled for examination of hardfaced surfaces.

Endurance tests have been completed on the branch with Colmonoy valves.

Due to delay related to valve supply and to a lesser extent in loop tests driving June 1989 instead of December 1988 was proposed and accepted by CEC as the final issue date of the contract.

Progress and Results

1.Valve manufacturer supply (B.3)

At present all the valves planned to be tested have been delivered to loop operator CEA; Table 1 summarizes for each test branch the situation of valves with respect to hard alloy and welding process choices.

2.Implementation of valves in the DOUBLEAU loop (B.4.)

As described previously /4/ the three valves with Stellite and three valves with Colmonoy have been assembled for loop testing on two different branches.

Begun in December 1987 on Stellite branch, the loop tests have been achieved according to the initial program /2/.

As regards the behaviour of valves during the loop tests non destructive examinations were performed for evaluation :

- hydrotest of the branch before endurance, thermal shock and erosion tests,
- visual and dye penetrant examination of hardfaced surfaces after each category of tests,
- control of strength transmitted to the stem by the actuator every 100

cycles during endurance tests,
- water-tightness control of valves after 0, 500, 1000, 1500 cycles for endurance tests, and after 0, 20 cycles $\Delta\theta$ = 100°C and 20 cycles $\Delta\theta$ = 250°C for thermal shock tests,
- flow rate control during erosion tests and leakage control of valves before and after erosion tests at 70 and 320°C.

Figure 1 gives the results for hydrotest and water tightness control of valves with Stellite at different steps of loop tests performance. During endurance tests, galling of the nut acting the stem was identified as responsible of an important leakage observed on the globe valve after 1300 cycles. The endurance tests have been correctly ended after the replacement of the corresponding nut. Visual and dye penetrant examinations subsequently carried out on hardfacing surfaces of the different valves did not reveal any indication.

The thermal shock tests were driven without any problem. A significant leakage was measured on the globe valve after thermal shocks corresponding to high differential temperature ($\Delta\theta$ = 250°C) : 400 cm^3/h instead of 0.5 cm^3/h. Visual and dye penetrant examinations showed a thin crack located clearly in the hardfacing deposit of the glove valve seat.

The globe valve leakage 0.4 l/h being relatively low when compared with that planned for erosion tests 13-20 l/h it was decided to proceed with the last series of loop tests. Special disk with calibrated orifice were used for erosion tests on the globe valves; the leakage measured before erosion tests was more important than aimed : 37 l/h including the supplementary leakage of 0.4 l/h due to the crack observed on the seat instead of 13-20 l/h.

Nevertheless erosion tests have been finally achieved without any problem at 70 and 320°C with a slight increase of globe valve leakage : 40 l/h at 70°C and 46 l/h at 320°C.

Visual and dye penetrant examinations performed afterward on concerned hardfaced surfaces did not show any significant extension of the globe valve seat crack; no indication of defect has been found on the corresponding disk.

Detailed description of the above results has been reported by CEA on the document referenced /3/.

Destructive examination of the hardfaced seat is in progress in order to determine the origin of leakage increase : erosion of the seat crack or erosion of the disk orifice or scattering due to the method used for leakage measurements.

Endurance tests have been completely driven on valves with Colmonoy; no significant difference behaviour was observed between these valves and those with stellite.

References

/1/ COHEN S. Framatome Report IM HT DC 0024 rév.A (1988)
Programme de principe des essais sur Boucle DOUBLEAU : Robinets sans Cobalt

/2/ SAUVADON S. CEA Report DEMT/87.350 rév.1 (1987)
Etude de revêtements durs. Procédure d'essai.

/3/ F.RUFFET, C.DHERVILLERS CEA Report DEMT/88 361 (1988)
Essai de revêtements durs sans cobalt : Essais de référence avec revêtement stellite.

/4/ The Community's research and development program on decommissioning of nuclear installations. Third annual progress report (year 1987). EUR 11715 EN.

Table I : Composition of the loop test branches with Cobalt based alloys and cobalt free alloys

. Stellite Branch (Reference)

VALVE	WELDING PROCESS	HARDFACING DEPOSIT ON BODY SEAT	HARDFACING DEPOSIT ON DISK	INTEGRAL RING
2" BODY WITH BONNET	OX - AC	STELLITE GRADE 6	-	-
2" GLOBE VALVE	OX - AC	STELLITE GRADE 6	STELLITE GRADE 12	VIRIUM 16 (STELLITE GR.6)
3" SWING CHECK VALVE	PLASMA	STELLITE GRADE 6	STELLITE GRADE 6	STELLITE GRADE 6

. Colmonoy Branch

VALVE	WELDING PROCESS	HARDFACING DEPOSIT ON BODY SEAT	HARDFACING DEPOSIT ON DISK	INTEGRAL RING
2" BODY WITH BONNET	TIG	COLMONOY 4	-	-
2" GLOBE VALVE	TIG	COLMONOY 4	COLMONOY 5	NITRONIC 60
3" SWING CHECK VALVE	PLASMA	COLMONOY 4.26	COLMONOY 4.26	NITRONIC 60

. Cenium Branch

VALVE	WELDING PROCESS	HARDFACING DEPOSIT ON BODY SEAT	HARDFACING DEPOSIT ON DISK	INTEGRAL RING
2" BODY WITH BONNET	OX - AC	CENIUM 36	-	-
2" GLOBE VALVE	OX - AC TIG	CENIUM 36 -	- CENIUM Z 20/698	CENIUM Z 20/698
3" SWING CHECK VALVE	PLASMA	CENIUM Z 20	CENIUM Z 20	CENIUM Z 20/698

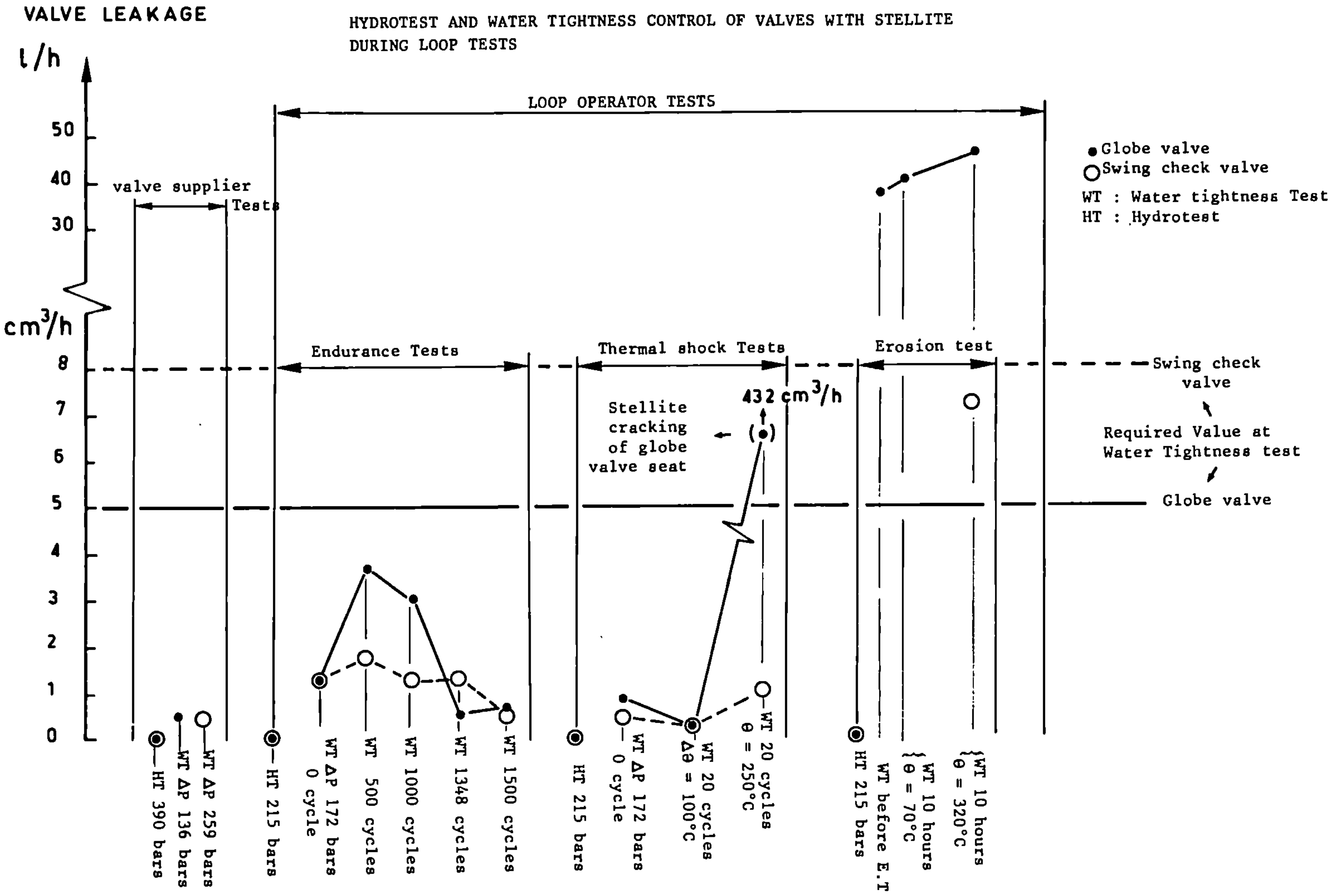


Figure 1: Hydrotest and water-tightness control of valves with stellite.

7.3. Pre-stressed Concrete Reactor Vessel with Built-in Planes of Weakness

Contractor: Taylor Woodrow Construction Ltd, Southall, United Kingdom
Contract N°: FI1D-0054
Working Period: October 1986 - December 1988
Project Leader: P. Dawson

A. Objectives and Scope

In the Community's first five-year (1979-83) programme of research on decommissioning of nuclear power plants, work was carried out to identify features which could be introduced to future plants to facilitate their eventual decommissioning and dismantling.

Among the features identified was the possibility of constructing the activated region of a pre-stressed concrete reactor vessel (PCRV) or a biological shield in blockwork which could later be removed easily in predetermined regularly shaped pieces. Although this feature would give rise to a number of planes of weakness in the inner regions of the structure, preliminary analyses showed that such planes of weakness do not significantly affect the overall performance or integrity of the structure.

The objective of the present research is to carry out supplementary, more detailed analyses focusing as necessary on problem areas, and then to verify the analytical results by means of simple, small-scale models of PCRVs with built-in planes of weakness.

The research will be directed mainly at PCRVs typically as used in the current Advanced Gas-cooled Reactor systems. The analyses will be confined to such structures and the models will also relate to them. However, it is considered that the results of the research will be equally applicable to the concrete biological shields used in current Light Water Reactor systems.

B. Work Programme

B.1. Review of the previous analyses to identify regions of the PCRV which may be particularly sensitive to the introduction of planes of weakness.

B.2. Construction of a computer model, of vessel models to take account of any significant indications derived from the review of the previous analyses.

B.3. Analyses of model vessels with and without planes of weakness, for a selection of load cases.

B.4. Construction of two concrete models of single cavity PCRV structures at scale 1/20, with and without planes of weakness.

B.5. Pressure testing of these structural models.

B.6. Conclusive assessment of results in relation both to PCRVs and Light Water Reactor system biological shields.

C. Progress of Work and Obtained Results

Summary

During 1988, the work carried out comprised analyses, construction and testing of the two models, one without and one with planes of weakness.

The analyses were carried out using the computer program ADINA (Automatic Dynamic Incremental Non-Linear Analysis) 1981 version, and the preliminary results suggested that some minor refinements to the initial design of the models would be advantageous. These were effected prior to the comprehensive analyses and subsequent construction of the models.

Construction of the models followed closely behind the analytical work, construction of the model without planes of weakness preceding that with planes of weakness by approximately two weeks. Steps were taken to ensure that, within practical limitations, this time interval was maintained between key events in the construction and testing of the two models so that when tested each would be essentially the same age and in the same condition.

Approximately two months after completion of construction, each model was pressure tested using water as the pressurising medium. Following a preliminary partial pressurisation to check instruments, each model was subject to three pressure cycles to design pressure and then one cycle to an overload pressure of 1.5 x design pressure. During each pressure cycle internal strains were recorded at 24 points and external surface deflections at 22 positions.

Progress and Results

1. Conclusions of the Review of Previous Analyses (B.1.)

This part of the study was completed during 1987, and is reported in the Annual Progress Report for that year.

2. Analyses (B.2., B.3.)

Analyses of the models have been carried out using the computer program ADINA, 1981 version. This is a computer program for the static and dynamic displacement and stress analysis of solids, structures and fluid structure systems. The program is such that it can perform a linear analysis and then follow this with a non-linear analysis requiring only a relatively few input changes. Each model was loaded incrementally with the same load history and the respective behaviour, including deformation, crack propagation and damage, monitored and compared as the loading increased.

The analytical grid used for the analyses is shown in Figure 1.

The type of analytical output obtained is shown in Figure 2 which shows the vertical deflection at the centre of the top cap for increasing pressure. This suggests that both models behave very similarly, with linear behaviour up to approximately 2.50 MN/sq.m (1.32 x design pressure) and an ultimate load capacity of approximately 4.75 MN/sq.m (2.50 x design pressure).

The locations of the nodes and sections at which analytical results have been plotted are shown in Figure 1.

3. Construction of the Models (B.4.)

Each model was constructed in three pours, as indicated in Figure 3.

The planes of weakness blocks were formed from 75 mm square electrical conduit cut into 37.5 mm lengths for the base blocks and 75 mm lengths for the wall blocks.

Each model incorporated instrumentation consisting of internal strain gauges, fixed to the reinforcement, and external deflection transducers. The respective locations of the instruments and of the points at which the analytical output was plotted, as shown in Figure 1, were chosen to be compatible so that these plots could be used directly for monitoring the behaviour of the models during their pressure tests.

4. Pressure Testing of the Models (B.5.)

The pressure test on each model consisted of three pressure cycles to design pressure (1.9 MN/sq.m) followed by one cycle to 1.5 x design pressure (2.85 MN/sq.m), using water as the pressurising medium. During each cycle to design pressure the instrumentation was read at 0.50, 0.86, 1.20, 1.55 and 1.9 MN/sq.m, and, for the overload test, again at 1.9, 2.24, 2.59 and 2.85 MN/sq.m. During the tests, the data from a selected few instruments were plotted to monitor structural behaviour as the tests progressed.

5. Conclusive Assessment (B.5.)

On the basis of the results obtained, the following preliminary conclusions may be drawn:

(a) The analyses indicate that the planes of weakness configuration assumed in the study has no significant effect on the structural behaviour of the model PCRV.

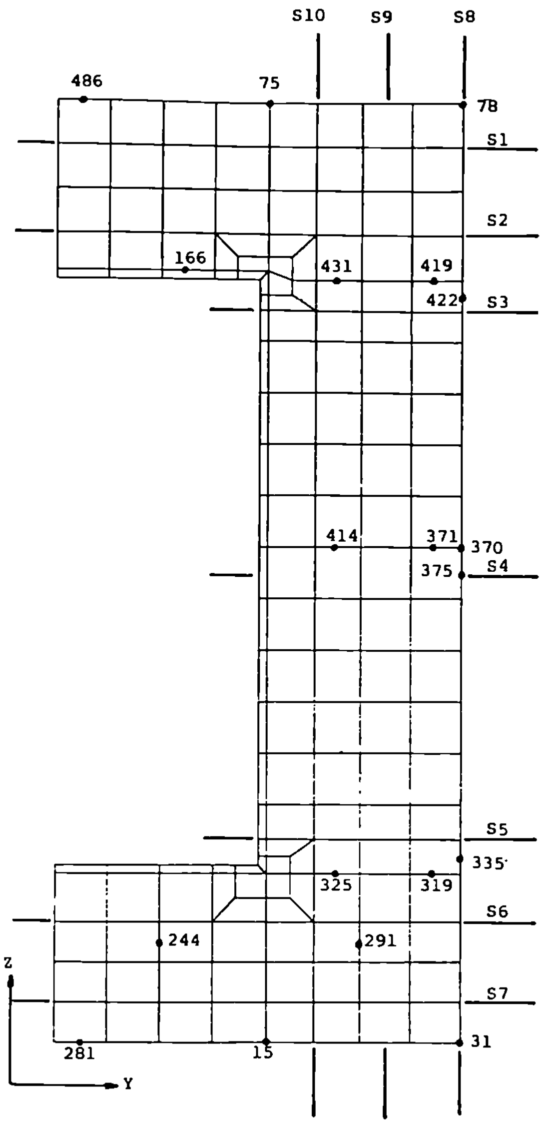
(b) The top cap of the model with planes of weakness agreed with analytical predictions. Figure 4 shows the plots of the predicted and measured vertical deflection at the centre of the top cap of this model.

(c) The barrel regions of both PCRV models behaved similarly and essentially as predicted by the analyses.

(d) Whilst the bottom cap of the model without planes of weakness behaved as predicted by the analyses, the bottom cap of the model with planes of weakness exhibited higher strains and deflections than predicted. It is believed that this behaviour may have been due to a mismatch between the physical and analytical modelling of the material used to form the weakness planes, and reflects the non-representative scaling of the thickness of the material which would be used in a full size structure. The effects of such a mismatch would be expected to be more evident when located in regions subject to bending, such as occurs in the cap.

The overall conclusion of the study is, firstly, that the analyses confirm the findings of the previous study in that they indicate that the planes of weakness have no significant effect on the behaviour or integrity of the PCRV. Secondly, it may be said that the common regions of the models unaffected by the suspected mismatch arising from the scaling effect of the weakness plane material, behaved in an essentially similar manner. To this extent, therefore, the model tests would appear to give support to the analytical results.

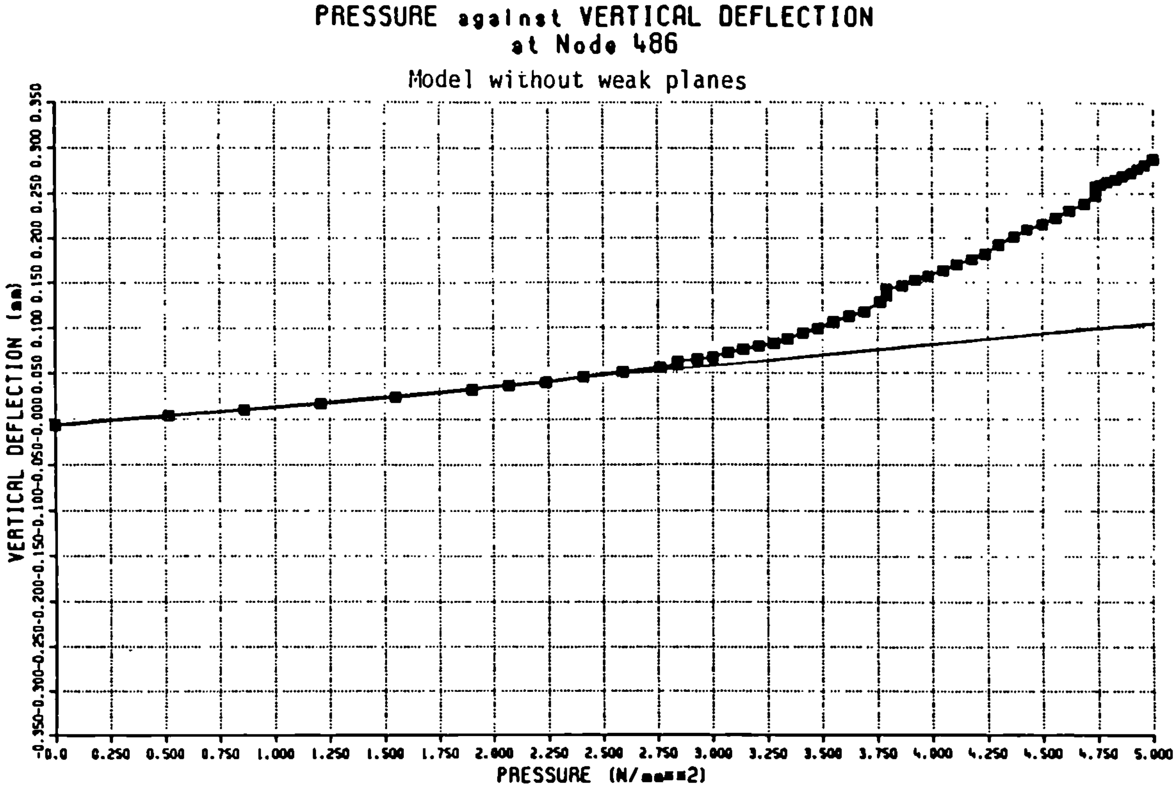
The final report on the study is now under preparation.



CO-ORDINATES OF NODES AND SECTIONS

Node or Section	Y co-ordinate (mm)	Z co-ordinate (mm)
486	42.25	1460
75	338	1460
78	645	1460
166	211.25	1198
431	449	1185
419	605	1185
422	645	1161.25
414	449	770
371	605	770
370	645	770
375	645	730
335	645	289
325	449	262
319	605	262
244	169	155.83
291	485	155.83
281	42.25	0
15	338	0
31	645	0
S1	-	1392
S2	-	1256
S3	-	1137.50
S4	-	730
S5	-	316
S6	-	187
S7	-	62.33
S8	645	-
S9	525	-
S10	413	-

Figure 1: Analytical grid showing location of points at which analytical output was plotted.



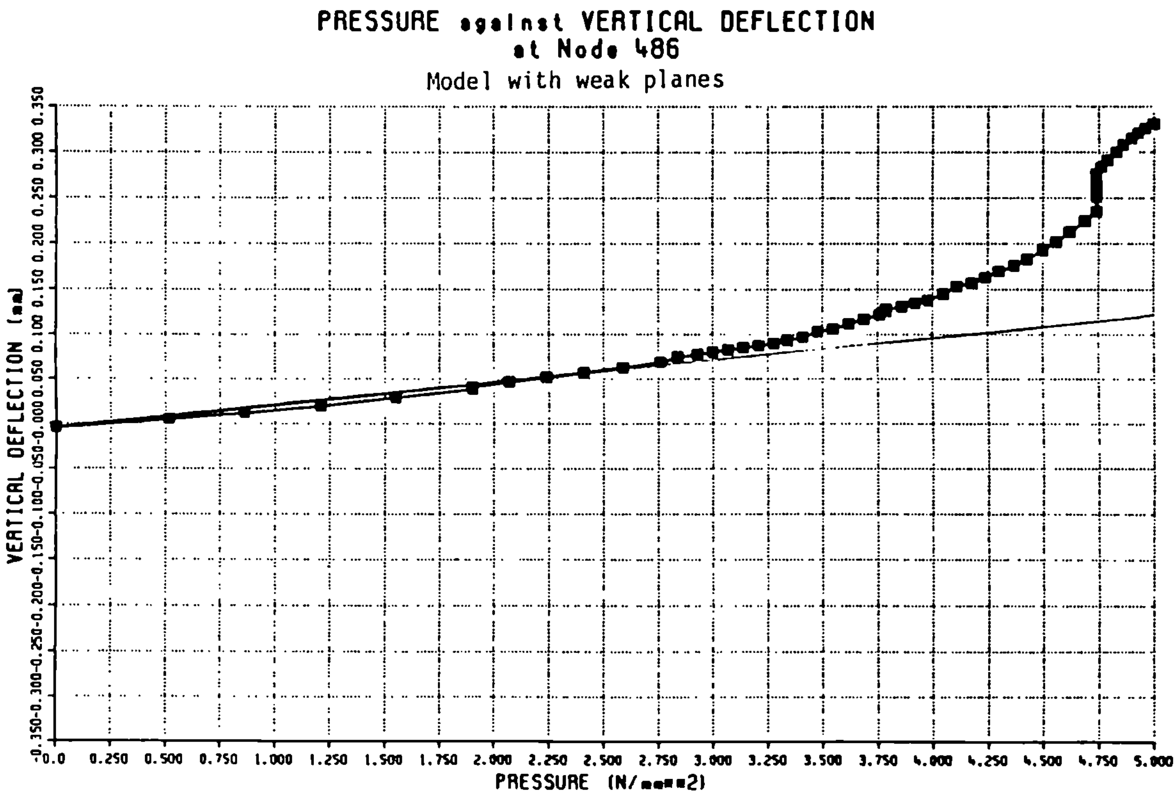


Figure 2: Typical analytical output plot - vertical deflection at centre of top cap.

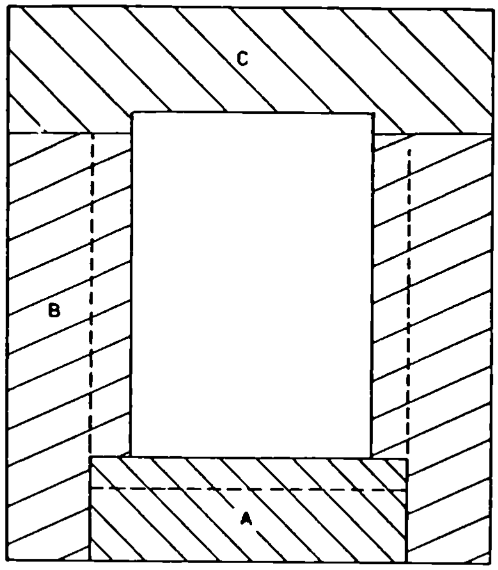


Figure 3: Concreting sequence for the models

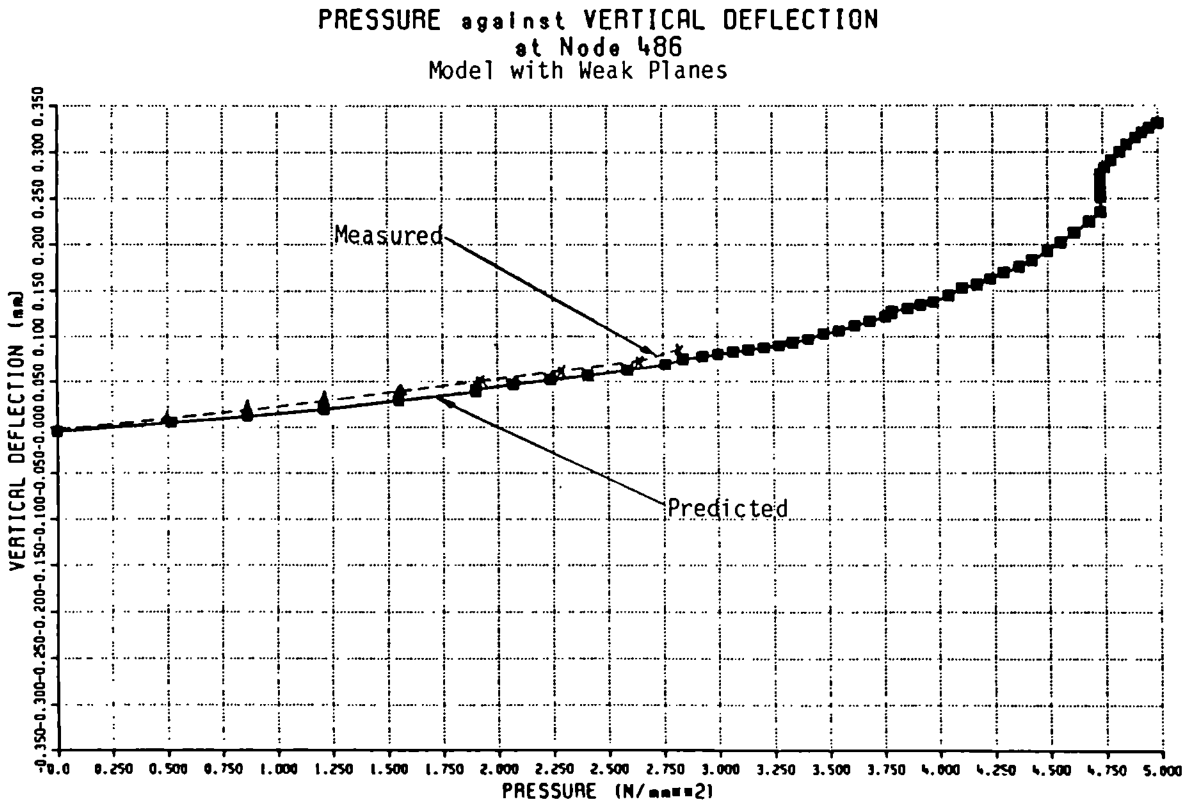


Figure 4: Plots of predicted and measured vertical deflection at centre of top cap.

7.4. In-situ Sealing of Concrete Surface by Organic Impregnation and Polymerisation

Contractor: Snia Techint Spa, Roma, Italy
Contract N°: FI1D-0055
Working Period: October 1986 - December 1988
Project Leader: V. Pellecchia

A. Objectives and Scope

The impregnation by resins of concrete structure is a process known as PIC (Polymer Impregnated Concrete). This process consists of dehydration of concrete, injection of monomer and thermopolymerisation of the resin. The PIC process is utilised to improve the chemical and physical behaviour of concrete structures, in order to extend the lifetime of bridges and viaducts under heavy traffic and severe atmospheric conditions. In the nuclear field the PIC process is being developed for immobilisation of radioactive wastes.

The objective of this research is to optimise the PIC technique for horizontal, vertical and subvertical concrete surfaces.

In a nuclear facility the impregnation of concrete structure is expected to give the following advantages:

- increase of mechanical resistance to impact loads, wear and abrasion;
- increase of leach resistance;
- increase of the mechanical restraints load capability;
- no maintenance required during plant operating lifetime;
- long-term integrity after final shutdown of the plant;
- very low capability to absorb contaminants because of full occlusion of all porosities of concrete structure.

The research is mainly directed to verify the above-mentioned points by designing, manufacturing and testing prototype equipment. The research programme will be jointly carried out with ITALCEMENTI Spa.

B. Work Programme

B.1. Design, manufacturing and implementation of special prototype device for impregnation by resins of concrete structures.

B.2. Pre-operational tests on concrete structures having different surfaces, in order to verify capability of experimental equipment to perform the injection of the monomer in all directions.

B.3. Optimisation of the PIC process parameters (temperature, vacuum dehydration, monomer pressure injection, etc.)

B.4. Qualification of the PIC process including comparison of the properties of the concrete matrix before and after the PIC treatment (mechanical and chemical tests, porosity measurement, etc.).

C. Progress of work and obtained results

Summary

Referring to points B.3 and B.4 of work schedule, the activities completed on 1988 have been:

- B.3.1 project and erection of a concrete walling unit having a size such as to be representative of a real scale testing structure.
- B.3.2 characterization of material used for the execution of concrete walling unit.
- B.3.3 impregnation and polymerization of the concrete structure.
- B.4 characterization of the concrete before and after the impregnation treatment.

Erection of the concrete unit was done with a four months of delay for troubles met in achieving fabrication permit from local Authorities in Bergamo. This unforeseen event has delayed all above mentioned activities, in particular activities B.3.3 and B.4 were carried out only in part, but will be completed during first quarter of 1989 (three months of delay on timetable).

Progress and results

1. Concrete structure (B.3.)

For the purposes of the investigation campaign a special concrete structure has been designed and implemented, whose dimensions are such as to be representative of a typical nuclear installation (i.e. a shielding wall of a medium activity hot cell). An outline of the structure, consisting of vertical and horizontal elements is illustrated in Fig.1. It had been decided that half of structure should be made of normal concrete and the second half of high density baritic one.
During the casting process, some material was collected for the purpose of preparing specimens for lab testing of both types of concrete. The values of physical and mechanical characteristics on both normal and impregnated specimens are shown on Tab.I and Tab. II.

2. Impregnation of concrete (B.3.)

Impregnation tests were started after some maintenance on prototype machine, since adverse environmental circumstances (the temperature had been below 0 °C for a large part of the day during work) made it necessary to replace plastic pipings and packings with others able to withstand low temperatures without becoming brittle.
In a first stage, involving six sectors of structure, the following (short) cycle of treatment was adopted, that took account of preliminary tests performed at the Italcementi Test Laboratory in Colleferro near Rome:

- 10 hours heating, the first four of wich at a temperature set at 105 °C and the subsequent six at top temperature (160 °C).
- 14 hours cooling.
- 1 hour for concrete degassing.
- 5 hours impregnation under slight overpressure (20 kPa).
- 4 hours polymerization with water at 80 °C.

In relation to results achieved with above proceeding cycle, the treatment cycle was modified as follows:

- 8 hours heating with temperature setting preset at 110 °C.

- 8 hours heating at 160 °C.
- 12 hours cooling.
- 1 hour degassing.
- 10 hours impregnation under overpressure.
- 6 hours polymerization with water at 80 °C.

Three holes to a depth of 3.5, 5.5 and 10 cm respectively were drilled into central area on the test surface area of the concrete, corresponding to the central area of the heating plate. Another 10 cm deep hole was drilled in the external area at about 20 cm far from the edge of the plate. For the purpose of temperature measurement, thermoresistances Pt-100 connected with a precision multi-meter and a microprocessor were placed within the holes. Temperature trends in function of time are shown in figs. 2, 3, 4 and 5. The first two concern the "short" treatment cycle, the last two the "extended" treatment. Temperature trends versus time measured on the portions of structure that were built out of baritic concrete are considerably different from those obtained on normal concrete as consequence of specific thermal conductivity of both concrete matrixes.
For hot water polymerization two procedures were tested. In the first water was heated in the boiler and circulated via the pump. In the second water was heated inside the impregnation chamber via the resistor fitted into the plate, starting off with hot water and without resorting to circulation.

3. Impregnated concrete characteristics (B.3.)

Data available on 31st Dec 1988 are only partial and relevant to preliminary test results. After polymerization, the following samples were collected from each of the treated segments of structure by means of core boring: 3 cores measuring 5 cm in diameter, 1 core with a 10 cm diameter and 1 core with a 15 cm diameter. Sampling and tests on the concretes of the sections treated with the "extended" cycle are currently in progress.
The cores with a Ø5 cm were cut, by means of a diamond-bladed saw, into 1 cm thick slices and from each of these the samples to be used for differential thermal analysis and pore-size analysis were cut out. Two cylindrical 2 cm specimens were sliced from the 10 cm diameter core. One was cut on the top surface and the other at a core depth of about 20 cm to measure water absorption by capillarity upstarting and for the determination of porosity accessible to water. Specimens were collected from the cores with Ø15 cm diameter for the purpose of evaluating their resistance to sliding friction by means of an Amsler tribometer.

4. Results

The differential thermal analysis gave the polymer content at different depths. The results are shown on figs. 6 and 7. Limitedly to specimens collected from "short" treatment cycle, the trends clearly reveal that the thickness of the concrete involved in the impregnation is about 30 mm, and no traces of polymer are found at a depth of 45 mm. The results of porosity analyses effected via mercury porosimeter are shown on the diagrams of figs. 8 and 9. In order to obtain significant data, the investigation had to be carried out on a large number of specimens. The curves shown on these figures stress that the porosity of samples collected on the external surface is around half that of samples collected internally. The increase of porosity in relation to the figure ranging between 50 and 70 mm is however significant for both types of concrete, but more markedly for

baritic concrete. This effect is most likely due to the heat treatment on an area of concrete that is not involved in impregnation-polymerization.
The determination of capillary absorptions, the results of which are summarized on the diagrams of figs. 10 and 11, stress the effect of impregnation. Impregnated concrete absorbs by capillarity at a much slower rate than normal concrete: after round 24 hours, the latter achieves a state equilibrium, while PIC doesn't do so even after 320 hours. Table III shows the results of the water absorption test under vacuum, according to UNI 6394. Each figure is the mean of two determinations. The volume mass of concrete is unaffected by impregnation; porosity accessible to water matches the results of the pore-size analysis. Figures relating to resistance to wear, whose average results are shown on Table II, reveal that impregnation has a weak effect on concrete made of silico-calcareous aggregates, while it boosts resistance to wear of barytic concrete by around 15% .

5. Conclusions

Since only about 50% of the test structure has been treated at this stage and some basic research is still in progress (i.e. leaching tests), it is too early to draw conclusions. Quite likely the treatment cycle will have to be modified further: some empirical investigations have revealed that monomer penetrates well beyond the 30 mm thickness within which the presence of polymer was noticed, and it will therefore be necessary to extend the treatment phase of thermal polymerization.
The complete characterization of PIC material, according to the aims of the project, will be available upon completion of all lab tests, in particular the leaching tests actually in progress according to procedure outlined in UNI 8798.

TABLE I - Physical and mechanical characteristics of concrete made of baritic aggregates - Not impregnated and impregnated.

	Lab. mixture		Yard mixture	Lab. impregnation
Cement amount daN/m3	320	360	320	-
Water/cement ratio	0.50	0.43	0.50	-
Vol. mass daN/m3	3490	3568	3475	-
Bending strength				-
7 days daN/cm2		44.6	39.4	-
28 days "		50.2	45.8	-
Compress. strength				
7 days	332	361	332	-
28 " "	391	461	436	-
90 " "			561	712
Mod.of elasticity daN/m3			294800 (308680)	314200 (316200)
Mod. of Poisson			0.24 (0.20)	0.23 (0.22)
Coeff. of thermal conductivity W/m °C			1.42	1.27
Thermal expans. µm/m °C			18.01	17.70
Polymer content % in wt.				2.55 ÷ 1.89

TABLE II - Physical and mechanical characteristics of concrete made of siliceous aggregates - Not impregnated and impregnated.

	Lab. mixture	Yard mixture	Lab. Impregnation
Cement amount daN/m3	320	354	-
Water/cement ratio	0.60	0.50	-
Vol. mass daN/m3		2370	-
Bending strength			
7 days daN/cm2		52.1	-
28 " "	48.3	53.8	
Compress. strength			
7 days		379 - 336	
28 " "		440 - 415	789
90 " "		531	874
Mod.of elasticity daN/m3		364.900 (396.400)	400.000 (403.900)
Mod. of Poisson		0.20 (0.16)	(0.12)
Coeff. of thermal conductivity W/m °C		2.13	2.35
Thermal expans. µm/m °C		12.45	12.25
Polymer content % in wt.		-	3.06 ÷ 2.38

TABLE III - Volume mass, porosity accessible to water and wear effect.

	Volum. mass daN/m3	Porosity % in volume	Wear mm/1000 m
Baritic concrete:			
ext. surface	3464	6.93	10.60
at 16 cm depth	3491	10.20	14.56
Normal concrete:			
ext. surface	2358	7.87	6.25
	2316	10.02	6.67

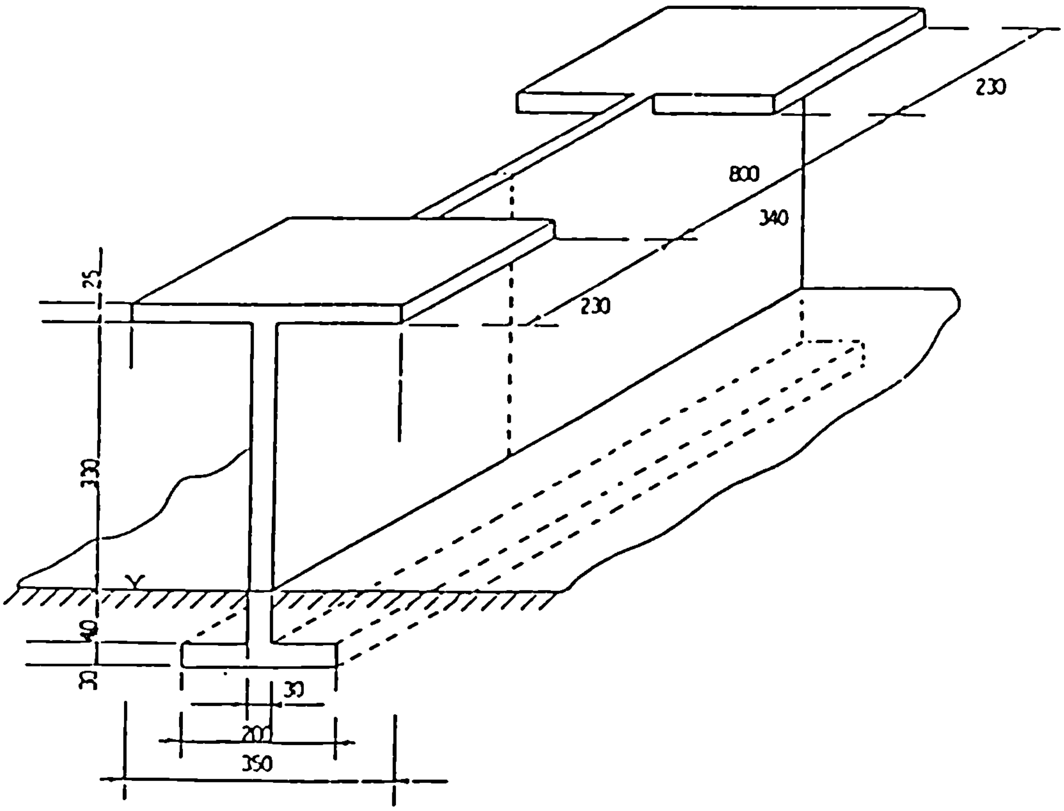


Fig. 1 - Axonometric draft of special concrete structur
implemented for "in situ" trial tests.

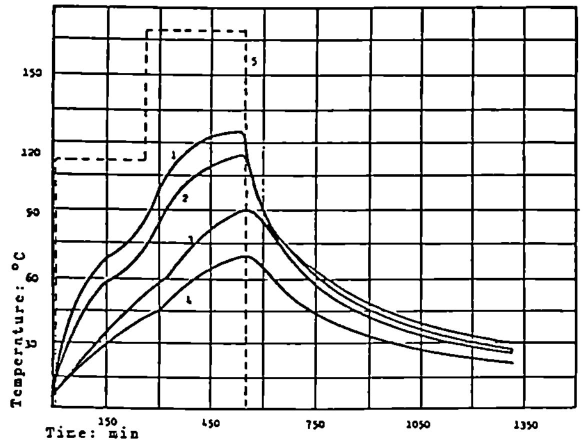


Fig.2 - Temperature/time diagram during dehydration of baritic concrete.

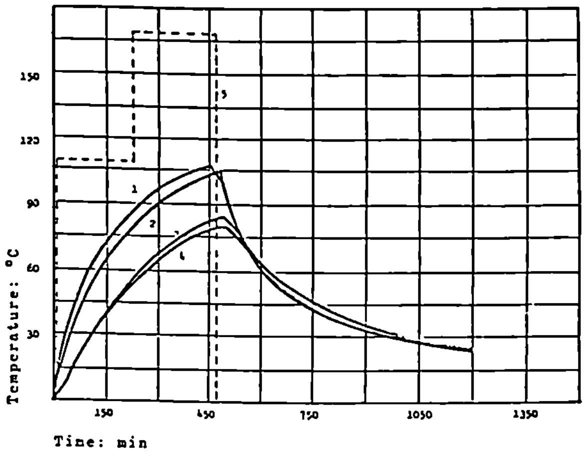


Fig.3 - Temperature/time diagram during dehydration of normal concrete.

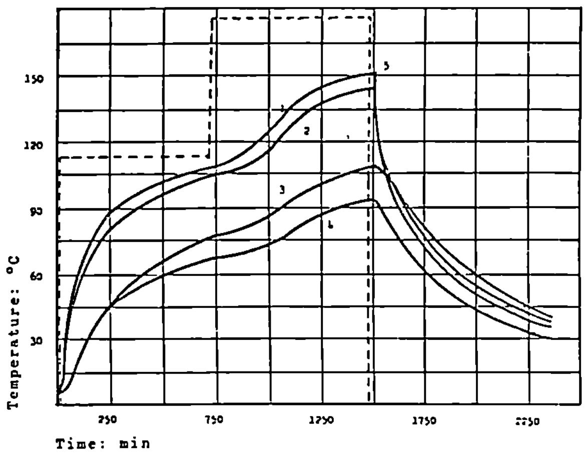


Fig.4 - Temperature/time diagram during dehydration of baritic concrete.

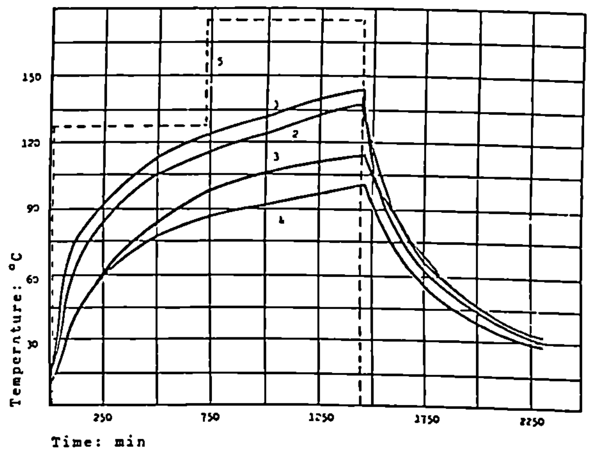


Fig.5 - Temperature/time diagram during dehydration of normal concrete.

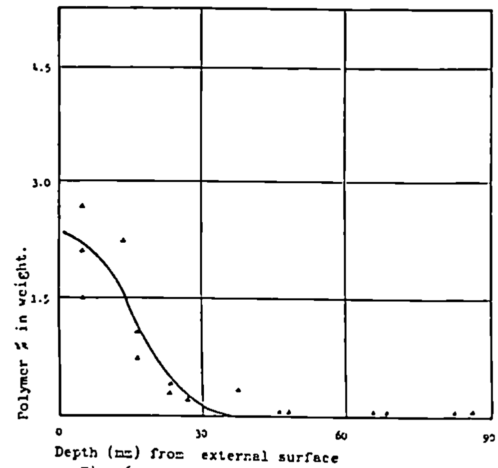


Fig. 6 - Baritic concrete: polymer content (% in weight) at different depths.

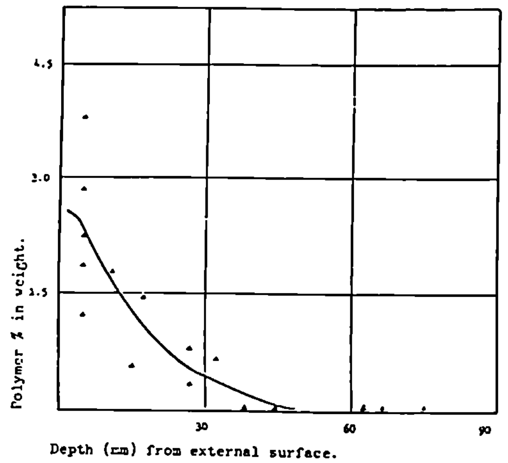


Fig. 7 - Normal concrete: polymer content (% in weight) at different depths.

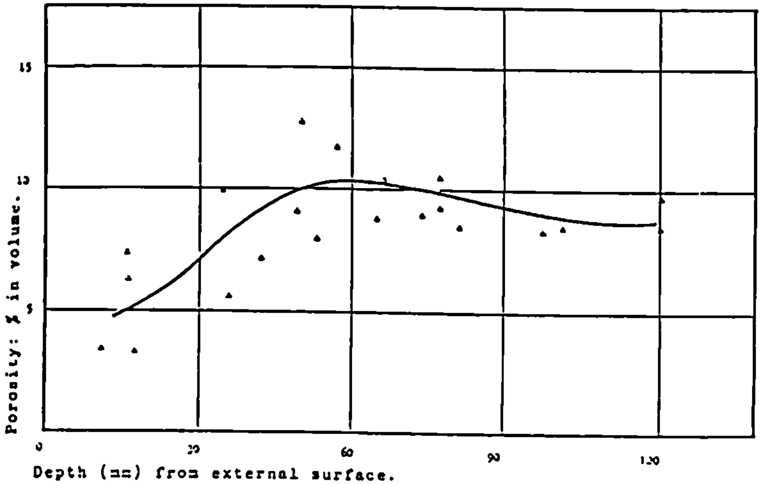


Fig.8 - Baritic concrete: porosity (% in volume) versus depth.

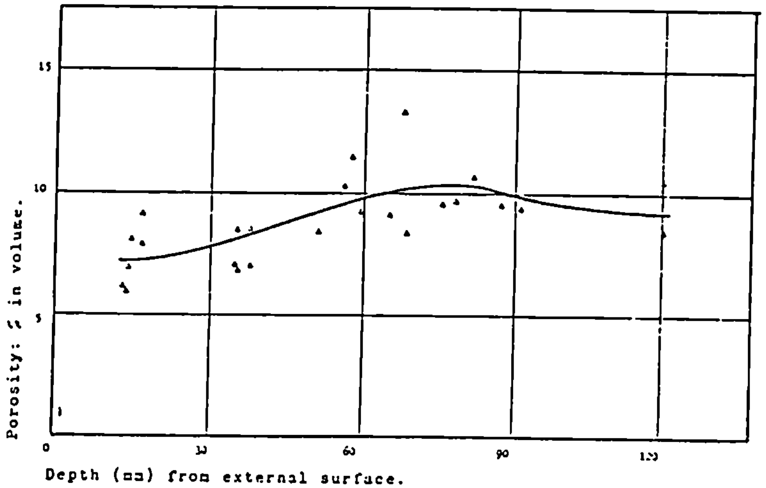


Fig.9 - Normal concrete: porosity (% in volume) versus depth.

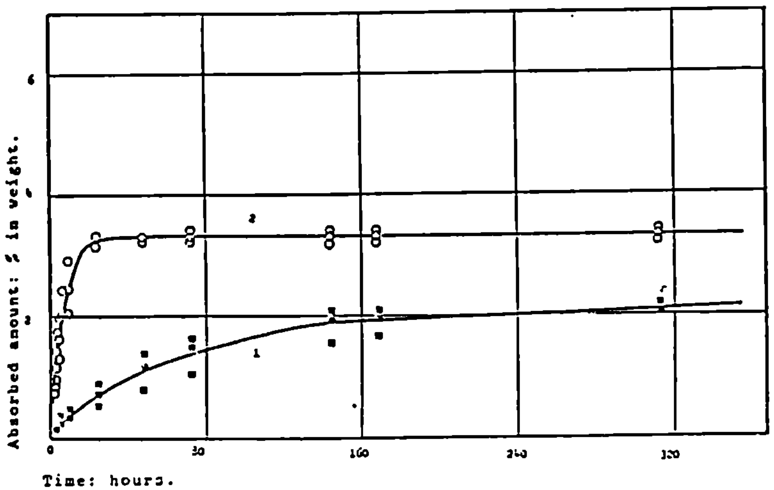


Fig.10 - Water quantity absorbed by capillarity in concrete specimens:
1-impregnated concrete;2- not impregnated.

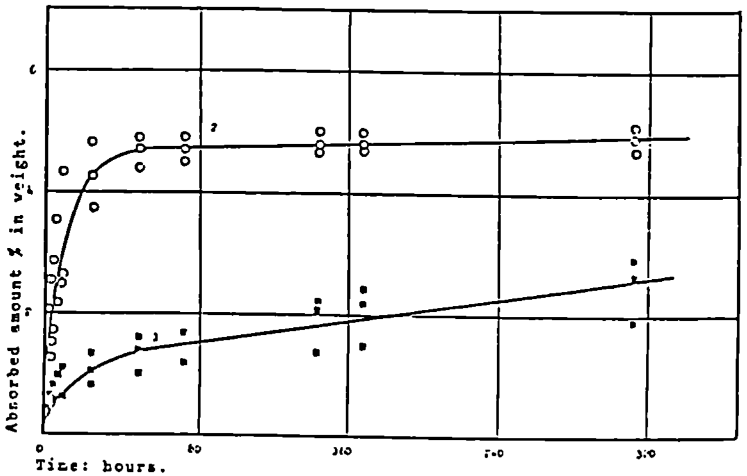


Fig.11 - Water quantity absorbed by capillarity in normal concrete specimens:
1-impregnated concrete;2- not impregnated.

7.5. Influence of Design Features on Decommissioning of a Large Fast Breeder Reactor

Contractor: Novatome, Le Plessis Robinson, France
Contract N°: FI1D-0056
Working Period: October 1986 - April 1989
Project Leader: C. Alary

A. Objectives and Scope

The objective of this research is the identification of the design and construction rules which should reasonably be brought into operation for the projected Fast Breeder Reactors (FBRs) in order to facilitate their dismantling.

A pool-type sodium-cooled FBR with a generating power larger than 1000 MWe will be taken as a reference for this study. Priority will be given to design features involving low equipment costs and little developments. Other features will only be mentioned.

B. Work Programme

B.1. Compilation and analysis of the design features of a FBR (> 1000 MWe) with regard to decommissioning (main dimensions and quantities, nature and localisation of contamination and activation, capacities of auxiliary systems, etc.)

B.2. Identification of the main features determinant to decommissioning: calculation of activation and dose rates of major components, assessment of problems posed by contamination, etc.

B.3. Study of various stages of dismantling, including assessment of remote dismantling and deferred dismantling.

B.4. Identification and evaluation of cost-effective design features facilitating decommissioning, considering low-cobalt steels, coatings, primary circuit draining and rinsing, decontamination of reactor internals, remote dismantling, etc.

C. Progress of work and obtained results

Summary

After a first calculation of activation made during the year 1987, we have realized complementary calculations taking into account the activation and contamination of SPX1 / SPX2 components. The decay time influence has been studied for some representative structures of reactor block. We note that the main radio-nuclides are (in activity) for steels:

Fe 55 and Co 60 for short time decay (activation),

Ni 63 , Nb94 and Ni59 for long time decay (activation),

Cs 137 and Na 22 for contamination.

From these elements, we have estimated the phases of final shut down and decommissioning to be followed according to the differents AIEA levels. For that, we have defined the delay time from the known technical criterias.

The different scenarios analysis and the activity calculations allow us to precize the different types of measure to facilitate the decommissioning (in the case of SPX2):

- reduction of cobalt content of steels and alloys:
- coating choice: The substitution of cobalt coatings by low grade cobalt coating gives the best benefits on activity reduction but a qualification of one of these coatings has to be done.
- bloc reactor draining, rinsing and decontamination arrangement: different measures, for example draining holes, must be taken .
- internal structures concept: It's difficult to modify the concept of these activated structures without heavy thermohydraulics studies, particulary for removing easily the diagrid. Nevertheless, for the internal structures and for slab structure, we give few measures to facilitate decommissioning.
- general layout and environment: measures for the crane, cells, pool, etc... .

Progress and results

1. Determination of main features for decommissioning (B. 2.)

Complementary calculations have been done for decay time between 0 and 200 years , and they take into account the long half-time radionuclides. The main contributors are the Co60 for gamma radiation source (first 80 years) , Ni 63, Nb 94 and Ni 59 for long time decay. The figures 1, 2 and 3 show us results for the diagrid , inner vessel and main vessel (lower part). Activation levels decrease very much from the most activated structure, tbe diagrid to the external structures (vessels, slab, concrete) which are protected during the 40 years in operation by neutron shield and sodium surrounding the core. The figure 4 shows that the concrete of the pit is not activated: A pool-type fast breeder reactor is not producing activated concrete wastes.

2. Various stages of decommissioning (B . 3.)

In the 87's annual report, we proposed stages of decommissioning but delay time could be more justified . When we describe the typical shape of activation decay for internal structures, we show characteristic inflexion points: the first for 20 to 25 years when Co60 activation becomes lower than Ni63 activation. The second for 70 to 100 years when Co60 γ activation becomes lower than Nb94. The total $\Sigma act_{\beta+\gamma}$ curve or Σact_{γ} curve are given in figure 7 . We can choose these 2 points as decay time for decomissionning at the stage 3 after a stage 1 or 2 and corresponding decay.

3. Identification of design features facilitating decommissioning (B . 4.)

3. 1 Reduction of cobalt content

The cobalt content in Stainless-Steels(SS) depends firstly upon the cobalt content of the nickel iron used in SS, then on smelted alloys percentage, on scrap and finaly on the hearth furnace type. From an initial specification of 0,25%, the cost increases of the Co content reduction (on approvisionnals of SS) becomes higher than 10% , for a specification lower than 0,05%. Technical benefits of this Co content reduction are interresting for high activated structures: reduction of dose rate and activity, reduction of wastes,... The examination of the benefits and possibility to reduce the cobalt content shows that it would facilitate decommissioning, but also the maintenance of the removable components.

3. 2 Coating choice

The base cobalt hard-facing ("stellite") on diagrid sockets, is the main activation source of the reactor block (see Figure 5). Selection tests shows that some substitute hard-facing (nitrure coating) or shoot-peened materials seems to have good friction caracteristics (see ref [2] et [3]). This hard-facing activation has been calculated (Figure 6) : it's equivalent to the SS base metal. The essential measure to take into account is the "stellite" suppression by a substituate hard-facing qualification.

3. 3 Bloc reactor draining, rinsing and decontamination

The measures which must facilitate both the sodium draining and reactor block and internals cleaning are converging: all which improve the gravity draining and the downing flow of sodium make easier the penetration of cleaning products. These measures concern first of all the diagrid and the core support structure, where the taking into account of simple technicals improvements allow to solve important difficulties: for example to foresee the drilling and the hollow prop needed for the drain device.

3. 4 Internal structures and slab structures concept

The features connected with the dismantling of these very activated internals structures are ticklish to treat, because they are connected to their design. But we propose few measures , such as the weight reduction, replacement of concrete biological shields in plugs by removable shields, holes,...

3. 5 General layout

Several improvements connected to the handling means, the packing and the area of conditionning the reactor block structures, are proposed: particulary it's necessary to have an handling multipurpose device available for decommissioning, and to foresee the possibility of the future use of some operation equipements (cells and pool storage fuel) by fitting-up their access.

References

[1] A. CREGUT et al: OCDE Meeting 17 au 19 mars 1980 Paris p. 15-21

[2] A. PIREAU Rapport Belgonucléaire RAP-049- for the CEC (GT "codes et normes")

[3] J. DAVIOT et al : LIMET 88 Avignon ; paper 503

[4] A. CREGUT et al: IAEA SM 243/33 13 au 17 novembre 1978 Vienne p. 187-202

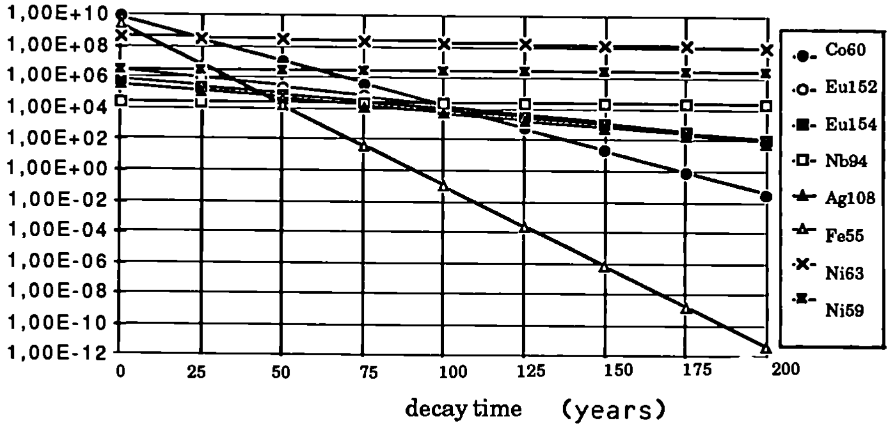


Figure 1: Diagrid activity (Bq/cm^3)

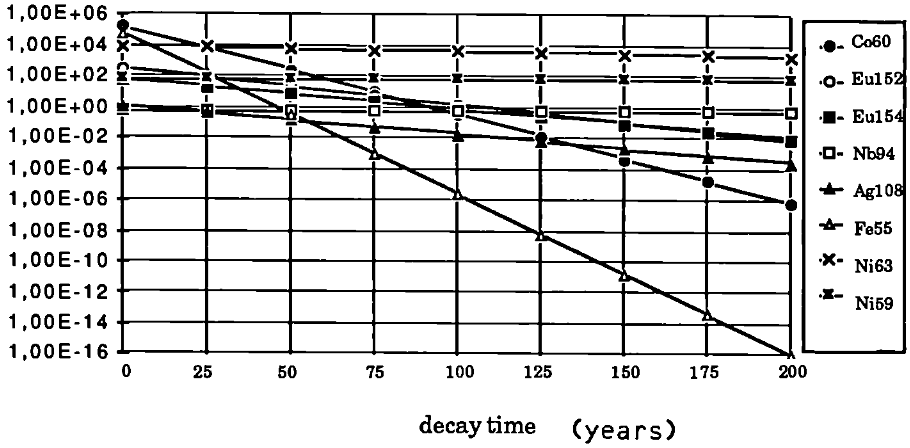


Figure 2: Inner vessel activity (Bq/cm^3)

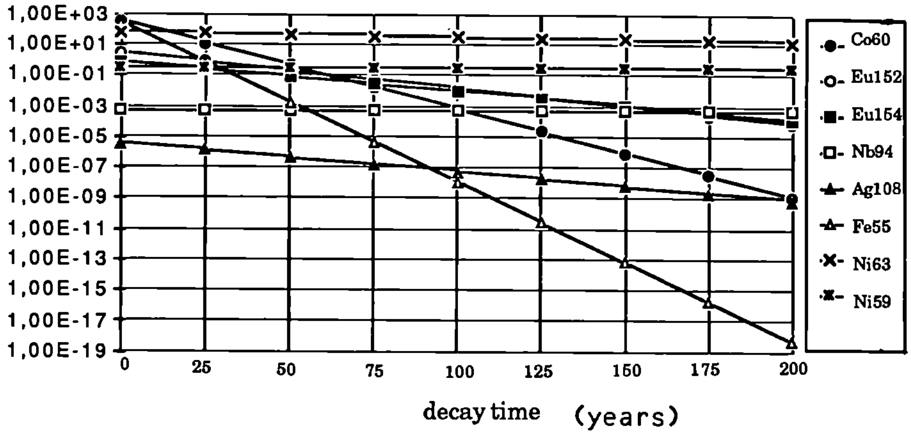


Figure 3: Main vessel activity (Bq/cm^3)

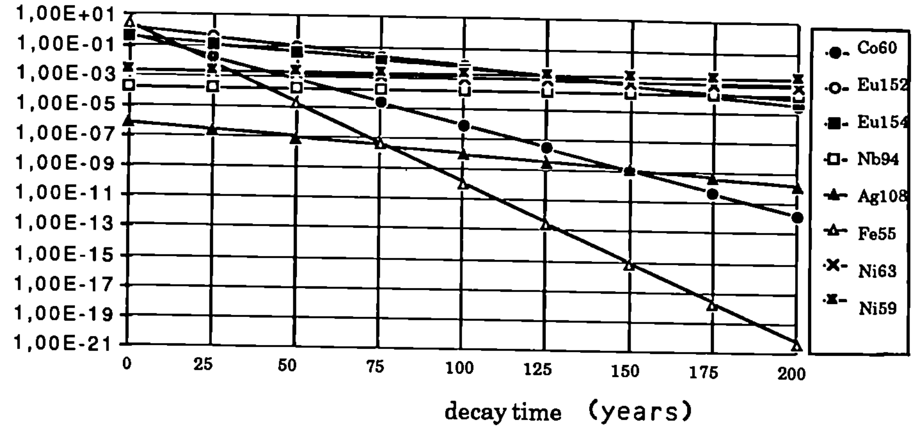


Figure 4: Pit concrete activity (Bq/cm^3)

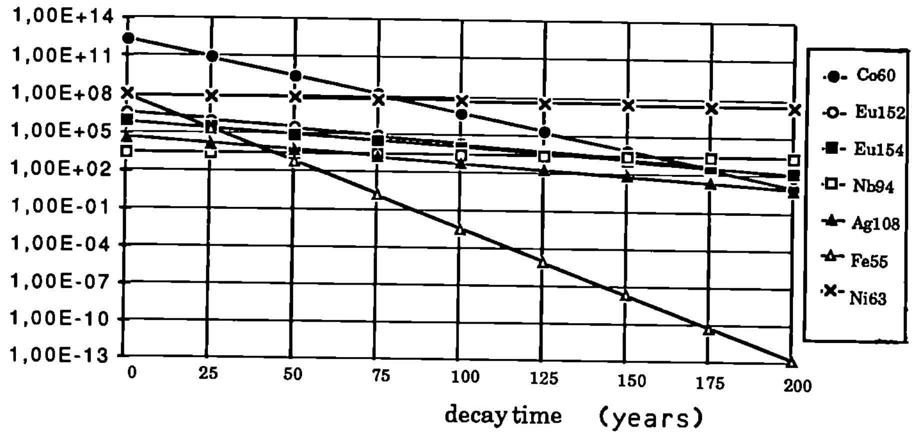


Figure 5: Stellite activity (Bq/cm^3)

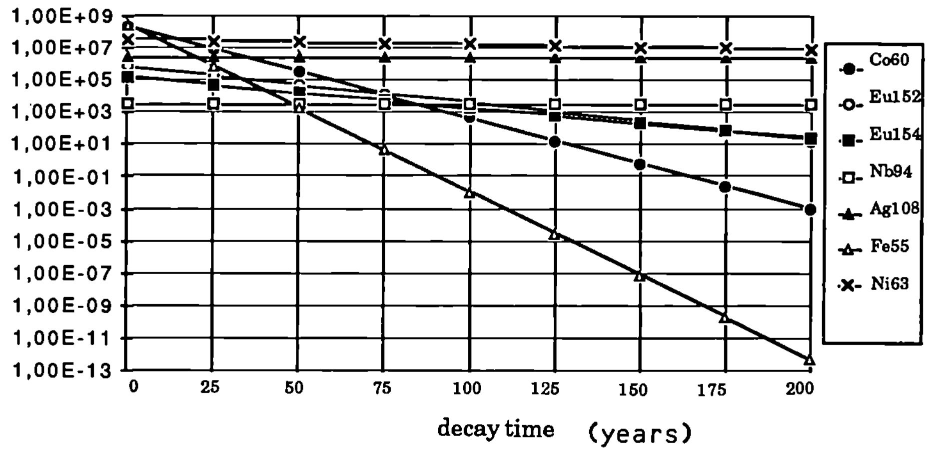


Figure 6: Nitrure coating activity (Bq/g)

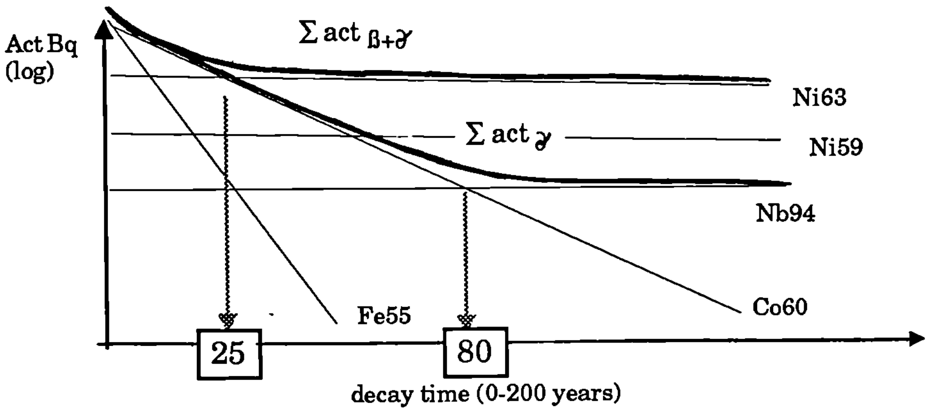


Figure 7: Σ act $_{\beta+\gamma}$ and Σ act $_{\gamma}$ variations

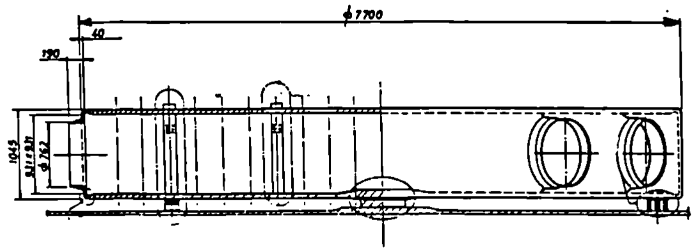


Figure 8: SPX2 Diagrid

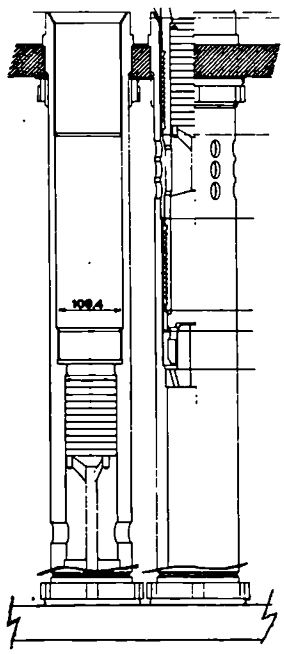


Figure 9: SPX2 Socket Diagrid

8. SECTION C: TESTING OF NEW TECHNIQUES UNDER REAL CONDITIONS

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

Eleven research contracts relating to Section C were executed in 1988, of which three were completed.

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

Contractor: Ente Nazionale per l'Energia Elettrica, Rome, Italy
Contract N°: FI1D-0023
Working Period: November 1984 - June 1989
Project Leader: A. Bertini

A. Objectives and Scope

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

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

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

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

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

B. Work Programme

B.1. Preliminary evaluation of the characteristics of the selected feedheater, including operating data, with respect to water chemistry and radioactivity inventory.

B.2. Determination of the radioactivity inventory of the feedheater including measurements on scrap samples.

B.3. Laboratory investigations on the ultrasonic and chemical procedures on representative samples, including tests on an appropriate treatment of the spent decontaminant.

B.4. Definition and selection of the most suitable procedure for the determination of the residual activity inventory.

B.5. Design and construction of an appropriate decontamination facility.

B.6. Dismantling and decontamination of the feedheater, treatment of the spent decontaminant, conditioning of the secondary wastes and determination of the residual activity inventory.

B.7. Evaluation of obtained results and final assessment for potential application to components of full-size BWR plants.

C. Progress of Work and Obtained Results

Summary

The disassembling work of the preheater No. 4 of Garigliano BWR, in order to sample the tubes to be used in the demo experiments, is reviewed. The cutting of the circumferential shell by using an ordinary manual oxy-acetylenic torch and the cutting and taking off the 1000 tubes, 1 m long, by using a mechanical disk saw are described in particular.

The acquisition of the full-scale ultrasonic machine and the direct gamma radiometric measurement system (to perform the final control of 1 m long tubes) is recognized.

The works to open the preheater and to take off the 1000 tubes were carried out by 4 Garigliano people. The total occupational dose was evaluated to be 0.23 man-rem.

Progress and Results

1. Introduction

The last in-scale testing performed in early 1988 at Garigliano BWR Power Station, on an assembly of 20 tube specimens 40 cm long, showed that the decontamination technique applying ultrasounds in aggressive chemical solution was able to clean the tubes of the preheater No. 4 down to the prescribed radioactivity levels with final values around 0.1 Bq/g of 60-Co. The decontamination procedure set up by laboratory and in-scale testing is summarized in Tab. I.

The demonstration of the total decontamination of the tube bundle of the preheater No. 4 will be performed by "10 full-scale demo tests" with an assembly of 100 tubes, each 1 m long, for a total of 1000 m of tube which represents about 10% of the preheater.

2. Disassembling of the preheater No. 4 for sampling 1 m long tubes (B.6)

The preheater No. 4 has been disassembled in order to allow the cutting and sampling of the 1000 tubes, 1 m long, for performing the "full-scale demo tests" on decontamination.

The main tasks of the disassembling work were:

- removing of the insulation (asbestos, 5 cm thick) from the shell of the preheater, in the cutting area along the circumference (for about 1 m of length); the bottom area of the preheater (for about 0.5 m^2); cutting the connections of the vent pipes;
- applying two drive wheels under the shell in correspondance of a stiffening rib and one more drive wheel under the bottom area, in order to avoid the turnover of the part of the shell to be removed;
- to put a support foot under the shell;
- to cut the vent pipes;
- cutting the shell in the circumferential manner by means of an ordinary oxyacetylenic torch;
- moving the bottom part of the shell by means of an hoist;
- to prepare the area for sampling the 1 m long tubes;
- cutting the tubes by a common electromechanical disk saw (with 250 mm disks).

The cutting of the tubes has been made between two support plates with the disk saw, using also a suction system, manually kept near the disk by an operator. A view taken during the works is shown in Fig. 1.

Two main problems were evidenced during the cutting of tubes. The first was an excessive wear of the disks of the saw: usually a disk needed to by changed after about 25 cuts; to performe the total tube sampling about 2500 cuts were needed and about 100 disds were used. The second problem was due to the fact that after cutting several tubes resulted squashing or damaging at them ends; for this reason, after sampling, all tubes were still machined in order to have the ends well finished.

The disassembling and cutting works have been performed by a group of 4 people. They used common protective clothes and masks.

The dose rate near the preheater (at 10 cm) before cutting the shell and opening was around 3-5 μSv/h with an area dose rate of about 3 μSv/h. After opening the preheater the dose near the tube bundle (at 10 cm) was around 25 μSv/h and at 1 m was around 4-5 μSv/h. The smearable contamination was about 0.3 Bq/cm^2 on the tube bundle and 0.1 Bq/cm^2 on the inside of the shell.

The occupational dose for the disassembling work, was evaluated to be $2.3 \cdot 10^{-3}$ man-Sv (0.23 man-rem) as shown in Tab. II.

3. Design and construction of appropriate decontamination facility (B.5)

3.1 Full-scale US machine

To performe the "10 full-scale demo tests" a new special ultrasonic machine was designed and acquired from ST.IM.IN (Milano-Italy) composed of four main tanks:

- two tanks with ultrasonic transducers, applied on all immersed sides, for performing the decontamination with water (1st phase) and with aggressive chemicals (2nd phase);
- a tank for washing and rinsing with water;
- a tank for treating (neutralization/flocculation) the spent decontaminants.

The machine was installed at the Garigliano Power Station in the controlled area in the decontamination work shop of the turbine building. In order to avoid the release of chemical pollutants in the atmosphere, a special chemical filter was installed on the suction line.

3.2 Full-scale radiometric measurement system

To control the effectiveness of decontamination a special direct gamma measurement system, able to evaluate the 60-Co specific radioactivity of 1 m long tubes, was designed and acquired from EG&G. A view is shown in Fig. 2.

4. Conclusion

The research activity was not completed in 1988, due to some problems in obtaining the full scale US machine which arrived at Garigliano BWR Power Station just in November 1988. Moreover, during the set up test of the US machine some more problems occured.

Table I – Decontamination procedure to be used for "10 full-scale demo tests" on 1 m long tubes taken off from the preheater No. 4 of Garigliano BWR.

1st phase : **US in water**
- test temperature : 60°C;
- test duration : 30 min.

Rinsing with water.

2nd phase : **US in aggressive chemical**
- test solution : 3% vol. HF/5% vol. HNO_3;
- test temperature : 60°C;
- test duration : steps of 30 min. each (with a maximum of 5 steps).

Rinsing with water.

Radiometric control by direct gamma measurement system

The procedure for the final radiometric control will be optimized during the "full-scale tests"; as a preliminary figure radiometric measurements will be performed at the end of each decon step.

The tubes with a residual 60-Co radioactivity less than 0.3 Bq/g will be considered fully clean and excempted from further decontamination.

Tab. II – Evaluation of the occupational dose rate for disassembling the preheater No. 4 of Garigliano BWR and for sampling 1000 tubes 1 m long.

Work task	**Dose rate (μSv/h)**	**Man-hour (h)**	**Occupational dose (man-Sv)**	**(man-rem)**
Opening preheater	3	100	$3 \cdot 10^{-4}$	0.03
Cutting and sampling tubes	10	200	$2 \cdot 10^{-3}$	0.20
Total	–	300	$2.3 \cdot 10^{-3}$	0.23

Fig. 1 - View of tube cutting works of the preheater No. 4 of Garigliano BWR.

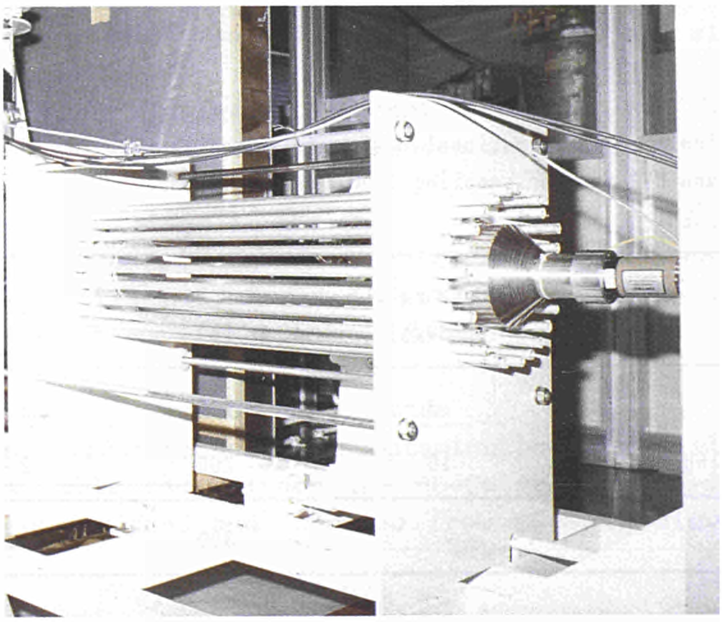


Fig. 2 - View of the full scale direct gamma radiometric measurement system with 20 tubes, 1 m long, to be measured.

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

Contractor: Belgonucléaire, Dessel, Belgium
Contract N°: FI1D-0024
Working Period: July 1984 - December 1987
Project Leader: J. Draulans

A. Objectives and Scope

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

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

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

B. Work Programme

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

B.2. Preparation and safe and leak-tight transportation of five large glove boxes to a special dismantling installation.

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

C. Progress of Work and Obtained Results

Work in this contract has been completed, the final report is under publication.

8.3. Large-Scale Application of Segmenting and Decontamination Techniques

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen, Germany
Contract N°: FI1D-0025
Working Period: January 1985 - June 1989
Project Leader: W. Stang

A. Objectives and Scope

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

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

B. Work Programme

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

C. Progress of work and obtained results

Summary

In the research period 1988 the following activities have been carried out:

- removal and cutting of components of the high-pressure turbine
- comparision of cutting effort for different components
- testing of decontamination by freon
- long-term surveying and optimization of electrolytes.

Cutting of about 25 tons of small metal scrap due to the work programme will not be executed because KRB A decided to melt down and recycle this kind of material for economical reason. This part of the programme was substituted by a new task:

- crushing and surveying of concrete.

Progress and results

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

At the beginning of 1988 the upper housing of the high-pressure turbine with a total mass of 22 tons was lifted and segmented subsequently to suitable pieces for decontamination in phosphoric acid. Removal of this heavy component was done non-destructively by unscrewing 32 bolts. Thereby the flanged joints had to be warmed up. Because of wall thicknesses up to 280 mm propane-oxy-torch cutting has been applied. By this cutting technique the upper housing with an average wall thickness of 80 mm could be segmented into 30 single pieces. All thermal cutting works were done under a mobile tent by operation of air-filtering facilities. The total effort for cutting was 210 manhours. The application of the propane-oxy-torch produced 1400 kg of slag, which is about 6.5% of the initial mass. The contamination of the upper turbine housing was found in a range up to 150 Bq/cm^2. The measurement of the aerosol activity in the cutting tent showed about 12 Bq/m^3. The average man dose was 12 $\mu Sv/h$.

Also a study on cutting velocities was made. By applying the propane-oxy-torch e.g. for 280 mm thick pieces the cutting velocity was 20 mm/min. The cutting velocity has been increased to 55 mm/min for pieces with a wall thickness of 100 mm.
Table I gives the total cost for dismantling and segmentation of thick-walled components such as a cast-steel housing of a turbine. Mass specific cost indicate that dismantling and segmentation of these components could be done very economically compared with thin-walled components. Table II gives a review on specific costs for different plant parts within the scope of the current research programme.

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

At the end of 1988 a small scale study on decontamination procedures was started. From the literature it was already known that freon is successfully applied for decontamination purposes, e.g. during outages of nuclear power plants, mainly to remove slightly adhering particles from electrical components. It should be investigated if freon is also suited for decontamination to unrestricted release.
The decontamination test with freon R 113 (boiling point 48°C) was executed by a mobile facility consisting of two standard containers (20 feet), one maschine and one working container. The decontamination was done mechanically by a high-pressure liquid jet. Thereby freon was pumped from a storage tank to a spray gun. Inside the closed working container the operator had to wear protecting clothes with an air supply from outside. After spraying, the freon has been filtered and pumped back into the storage tank. Also, it has been tried to recover the airborn freon in the working container by condensation. This recovery was found to be insufficient.
Mainly components such as electric motors, transformers and electrical surveying instruments have been decontaminated. A list of plant parts treated by freon for decontamination is given in Table III.
After decontamination about 80% of the parts were found to be below 0.37 Bq/cm^2 by wiping test measurements. However, activity measurements by α,β-monitors indicated that a release for unrestricted reuse is impossible because of the fixed residual activity being still above the surface specific limit value. Finally it turned out that this decontamination technique is only profitable to remove loose adhering particles, such like dust, dirt and oil films.
Decontamination by freon therefore is not recommended for decommissioning works with the aim to release components for unrestricted reuse. The high loss of freon is an additional reason to substitute this technique under the aspect of environment protection.

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

The phosphoric acid is periodically regenerated with the intention of an unlimited reuse. The increase of the dissolved iron content resulted in a loss of decontamination efficiency by exhaustion of the phosphoric electrolyte. By diminishing the acid concentration and adding oxalic acid the electrolyte can be regenerated. So the eletrolyte should be usable for an unrestrictedly long time, provided that no enrichment of other components, especially radioactive substances, will occur.
The activity in the electrolyte consists of Co-60 and Cs-137. Because of the similarity of Co^{2+} with Fe^{2+} it could be expected, that the concentration of cobalt would be reduced by precipitation as well as the iron content. Cesium, however, should not be precipitated by oxalic acid for chemical reasons and therefore a long-term enrichment is expected. Nevertheless, an enrichment of Cs-137 could not be immediately detected by previous investigations.
Because of the significance of a potential enrichment of cesium, the investigations were repeated in a more detailed manner; for several regeneration cycles it was examined how the radioactive nuclides, dissolved in the spent electrolyte, distribute themselves in the iron oxalate during the regeneration process.

It could be demonstrated that the main part of Co-60 appeared in the iron oxalate, while only a small part of Cs-137 could be found in the precipitate. The content of Co-60 was reduced from the exhausted to the regenerated electrolyte by a factor of about 5.3, whereas the concentration of Cs-137 has been decreased only by a factor of about 1.3.
These results demonstrate that a long-term enrichment of cesium takes place in the electrolyte. However, until now an influence on the decontamination efficiency was not detected.

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

In the reporting period 33.7 tons of fragmented concrete has been collected by dismantling inside the turbine house. Because it is impossible to monitor the surface specific contamination KRB A decided to split the material into free and contaminated material by regarding the mass specific activity in order to get a licence for unrestricted release of a part of the concrete.
First the loose concrete fragments have been crushed into pieces of walnut-size and filled into bags by a facility shown in Figure 1.
The first step after filling the concrete into bags was a rough sorting by measuring the dose rate. By this, already about 13% of the concrete was found to be unqualified for unrestricted release. The remaining concrete has been monitored in a surveying camera equiped with two liquid scintillators for gamma-activity. After measuring, all bags, 40 kg each, were sorted into 5 classes of contamination with the limit values of 0.37, 1.00, 2.00, 5.00 and >5.00 Bq/g.
The result of sorting is presented in Figure 2. About 18.7 tons or 55% of the initial mass were found to be within the classes 1 to 3 (<2.00 Bq/g). This material with an average activity of 0.58 Bq/g in total, will be released and transported to a conventional dump site. The remaining concrete has to be stored for final disposal.
The total effort for crushing, transports, measurements and sorting was determined to 1280 manhours. A comparison indicated that the costs for crushing, surveying and splitting the concrete, as practiced, would be nearly equal to the costs for the direct final disposal of the whole mass. The economy of such works depends on the percentage of the mass which can be released as well as on the total mass itself because of investment costs spent for the crushing and measuring facility. For similar projects it is recommended to estimate the percentage of concrete which can be released, e.g. by random test prior to crushing works.

Table I: Dismantling and segmentation costs for the thick-walled turbine housing

staff	700 DM/t
consumable material	210 DM/t
conditioning and final storage	240 DM/t
total	1.150 DM/t

Table II: Dismantling and cutting costs of different components

type of component (ferritic steel)	average wall thick-ness (mm)	cutting technique for		costs (DM/t)
		dismantling	segmentation for decontami-nation	
various pipes	9	thermal	sawing + therm.	3.800
condenser plates	61	non-destruc.	thermal	1.700
condenser plates	25	thermal	thermal	850
housings and shells	80	non-destruc.	thermal	1.150
condenser tubes (brass)	1	non-destruc.	grinding	2.100

Table III: Results of the deconatamination of plant parts by freon

type of component	initial activity		residual activity	
	wiping test (Bq/cm^2)	α,ß-monitor (Bq/cm^2)	wiping test (Bq/cm^2)	α,ß-monitor (Bq/cm^2)
drilling machine	1.13		0.10	0.50
aresol monitor	7.40		0.05	
welding machine	42.00		0.84	6.80
elecric motor	6.87		0.03	
transformer		40.00		2.20
transformer		20.00		10.00
hand lamp		20.00		0.80

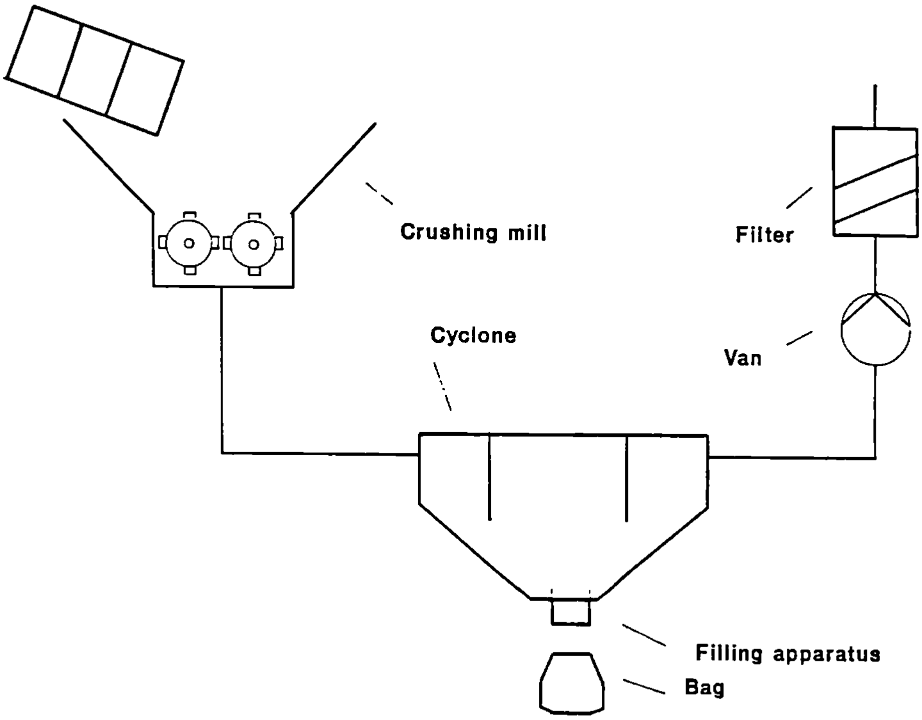


Figure 1: Crushing facility for concrete

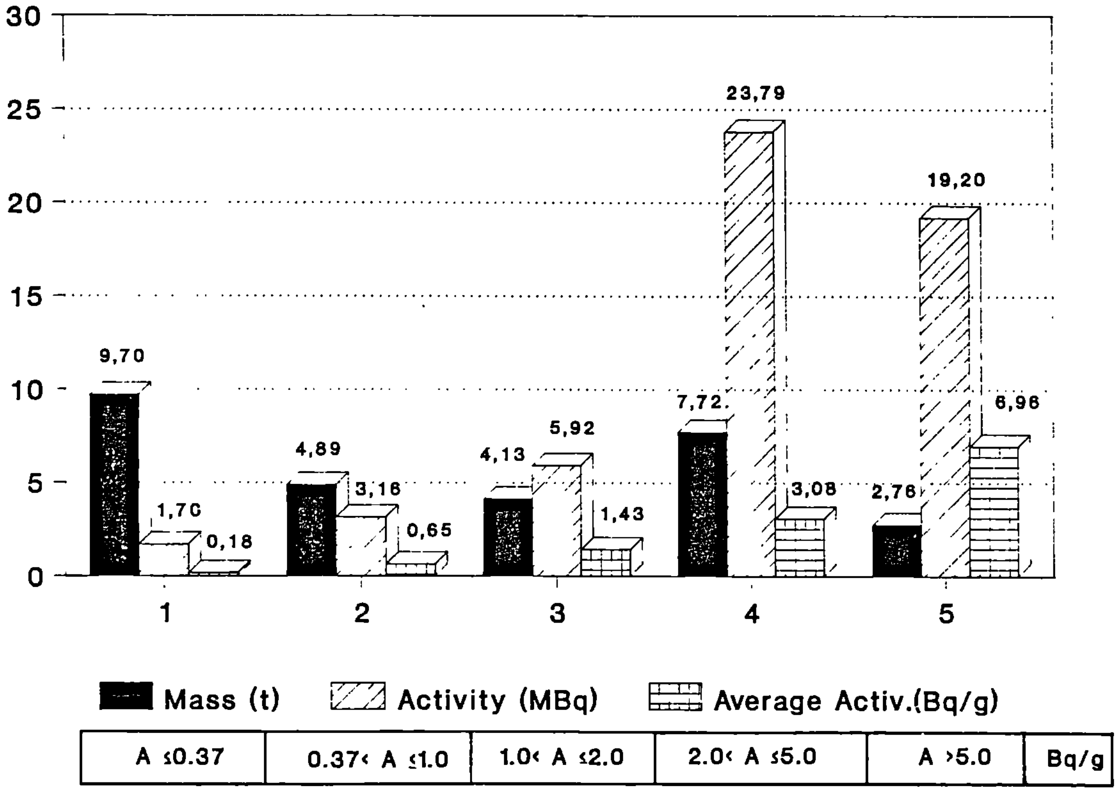


Figure 2: Distribution of mass and activity in surveyed concrete

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

Contractor: United Kingdom Atomic Energy Authority, Windscale Nuclear Laboratories, United Kingdom
Contract N°: FI1D-0026
Working Period: January 1985 - March 1989
Project Leader: J.R. Wakefield

A. Objectives and Scope

The objective of this research work is to develop methods for the decontamination of the heat exchangers of the Windscale Advanced Gas-cooled Reactor. This will demonstrate that such plant can be decontaminated and dismantled for disposal without environmental hazard and without exceeding the prescribed radiation dose limits to operatives. The final purpose of decontamination is to enable hands-on methods of dismantling to be used and avoid expensive and time-consuming remote operations. A further objective is to establish the nature of corrosion and contamination within the heat exchanger gas-side in order to provide data for future decommissioning of similar plant.

By removing a limited number of samples, having regard to the doses incurred by the operatives, it is expected to be able to identify the contaminants and recommend methods of removal. The large-scale work will be limited to the in-situ decontamination of a superheater by spraying of a recirculated mixture of nitric and citric acid with subsequent treatment of the arising secondary waste. This pilot experiment should demonstrate the effectiveness of the decontamination procedure and its appropriateness for an application to the full-scale project. A complete costing and dose inventory will be maintained as the operations proceed.

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

In a supplementary agreement concluded in 1988, the work programme has been reduced for technical reasons to the present version.

B. Work Programme

B.1. Characterisation of surface corrosion and contamination on extracted samples of the heat exchangers, and selection of an effective contaminant allowing for an acceptable disposal of the secondary waste.
B.2. Design, manufacture and installation of a pilot decontamination plant.
B.3. Decontamination of a superheater and treatment of the secondary waste.

C. Progress of Work and Obtained Results

Summary

During the period of the report, NEI-IRD have designed, manufactured, works-tested, installed and commissioned a pilot plant for decontaminating a WAGR heat exchanger superheater. Within a few minutes of starting the commissioning phase, using water only, the prescribed radiation limits on the plant were exceeded and the plant was shut down. Authorisation to revise the safety limits and operating procedures was requested. This authorisation is now awaited. Samples of the process liquor have been analysed and measurements of activity on the axis of the superheater have been made. In view of the delay introduced by this unexpected result, the CEC has been requested to authorise extension of the contract to 31 March 1989.

Progress and Results

1. Characterisation of contamination and specification of reagent (B.1)

The superheater is of the cross-flow type and comprises 676 horizontal tubes within a 2 m square insulated gas-duct. The tubes are of $2\frac{1}{4}$ Cr, 1% Mo steel and are 51 mm OD and unfinned. Total surface area is 450 m^2. Gas flow was upwards through the tube bank.

Earlier work at UKAEA Winfrith established that the tubes have an external corrosion layer about 150 µm thick which is mainly magnetite (Fe_3O_4) with a chromium content varying between 1% and 10%. The total volume of oxide is around 70 litres. Contamination was found to be incorporated in the oxide layer and to consist predominantly of Cs-137, Cs-134 and Co-60. Average values of 16 KBq cm^{-2} for Cs and 0.2 KBq cm^{-2} for Co were obtained, giving a total for the superheater of about 0.09 TBq (~2.6 Ci), when corrected for the activity gradient through the tube bank. The Winfrith tests showed that although Cs is water-soluble, water-washing was almost completely ineffective due to the nature of the contamination deposit. Dilute (0.5 molar) HCl at ambient temperature was recommended as the most suitable reagent, being 4 or 5 times more effective than the equivalent HNO_3. However, due to constraints on effluent disposal via BNFL waste streams, HNO_3 was selected for the pilot experiment.

2. Design, manufacture and installation of the plant (B.2)

A schematic diagram of the recirculating spray system is shown in Figure 2. A cylindrical tank of 5000 litres capacity is located on the ground floor of the reactor containment building directly below the heat exchanger. The tank stands in a cylindrical bund of 4400 litres capacity. Reagent is stored in 1000 litre industrial bulk containers (IBCs) sited outside the containment building and is transferred into the tank by means of a 66 $litre.min^{-1}$ electric pump (P1 on the diagram). A second electric pump (P2) of 500 $litre.min^{-1}$ capacity delivers the reagent, via a 50 µm filter and a flowmeter, to a 120° spray nozzle positioned above the superheater bank, at a height of 16.5 m above ground floor level. Reagent falling through the superheater bank is collected at the bottom of the hot gas duct and returned to the tank through pipework and a 250 µm strainer located within the tank. Spent reagent is transferred from the tank to a road tanker outside the containment building via pump P2 and the 50 µm filter. Any spillage or leakage collected in the bund can be pumped out by pump P1 and either returned to the tank or discharged to the road tanker. The tank, the heat exchanger shell and the chemical analysis room are ventilated via a demister into the existing gas treatment plant, which exhausts into the reactor building stack. It is considered that significant quantities of hydrogen will not be generated therefore no provision is made for purging the heat exchanger. The tank, bund, pipework, valves, 50 μm filter and the spray nozzle are manufactured from copolymer polypropylene. Joints are made by heat fusion or flanges as appropriate. The pumps, 250 μm strainer and spray lance are stainless steel. The filling and discharge pipes within the containment building are fitted with through connectors fixed to the containment shell. Removable hoses connecting these, respectively, to the IBCs and the road tanker are flexible polypropylene reinforced with stainless steel.

3. Operation of the pilot plant (B.3)

Following commissioning of the circuit external to the boiler, using the spray-head by-pass, water was sprayed into the superheater at 300 litres.min^{-1}. After only 3 minutes operation a γ-alarm indicated that the 100 μSv.hr^{-1} limit set for the process tank at contact had been exceeded. The reading was 1000 μSv.hr^{-1}. A 250 ml sample of the tank liquor was taken for chemical and isotopic analysis. Activity readings from the sample, in μSv.hr^{-1}, were:

Before filtration, solution + solids - 320 βγ, 150 γ

After filtration, solution - 200 βγ, 150 γ solids - 10,000 βγ, 100 γ

Preliminary analysis results indicate that the dominant activities are due to dissolved Cs, and Cs and Co particulate. Tank contents (3000 litres) are estimated to contain 6 x 10^{10} Bq (1.6 Ci). Tests are proceeding to validate the measurements and to confirm the predominant activities. Superheater activity measurement, using a string of film badges, was repeated and showed a significant decrease compared to the pre-test measurement (see Figure 1). It is now proposed to clean up the tank liquor by pumping through the system in-line filter (which was not contaminated during the run) and then to dispose of the filter elements and liquor. Water spraying will then be resumed and continued until no more activity is removed. Dilute nitric acid will then be sprayed. Authorisation has been requested to raise the tank and filter radiation limits to 2 mSv.hr^{-1}. It will be necessary to shield these units. The experiment so far has indicated that water-washing is more effective than was predicted, and that the superheater activity is higher than was estimated.

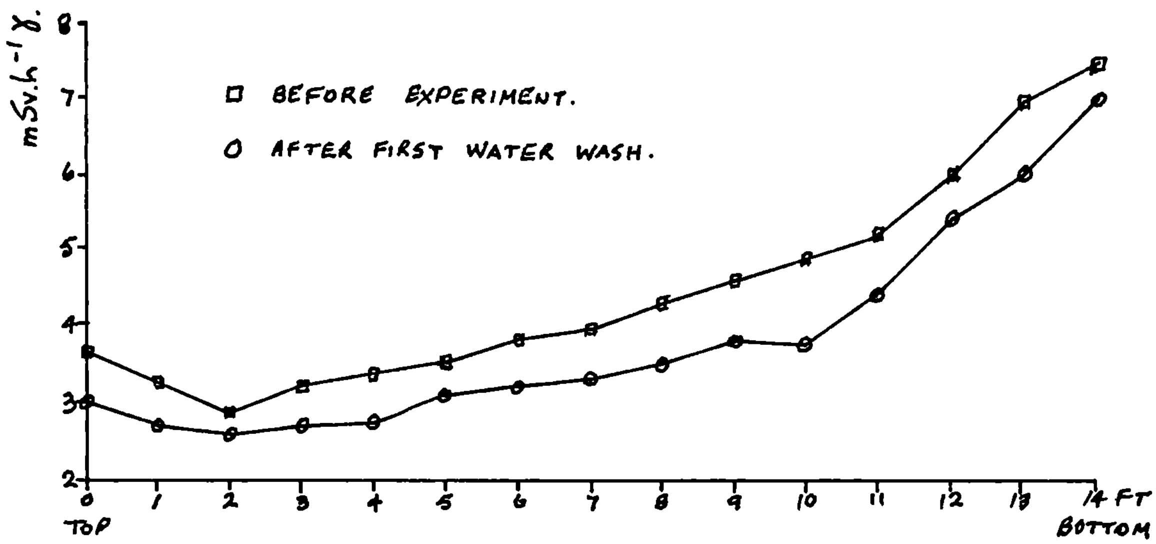


FIGURE 1. SUPERHEATER RADIATION MEASURED ON AXIS. GAMMA.

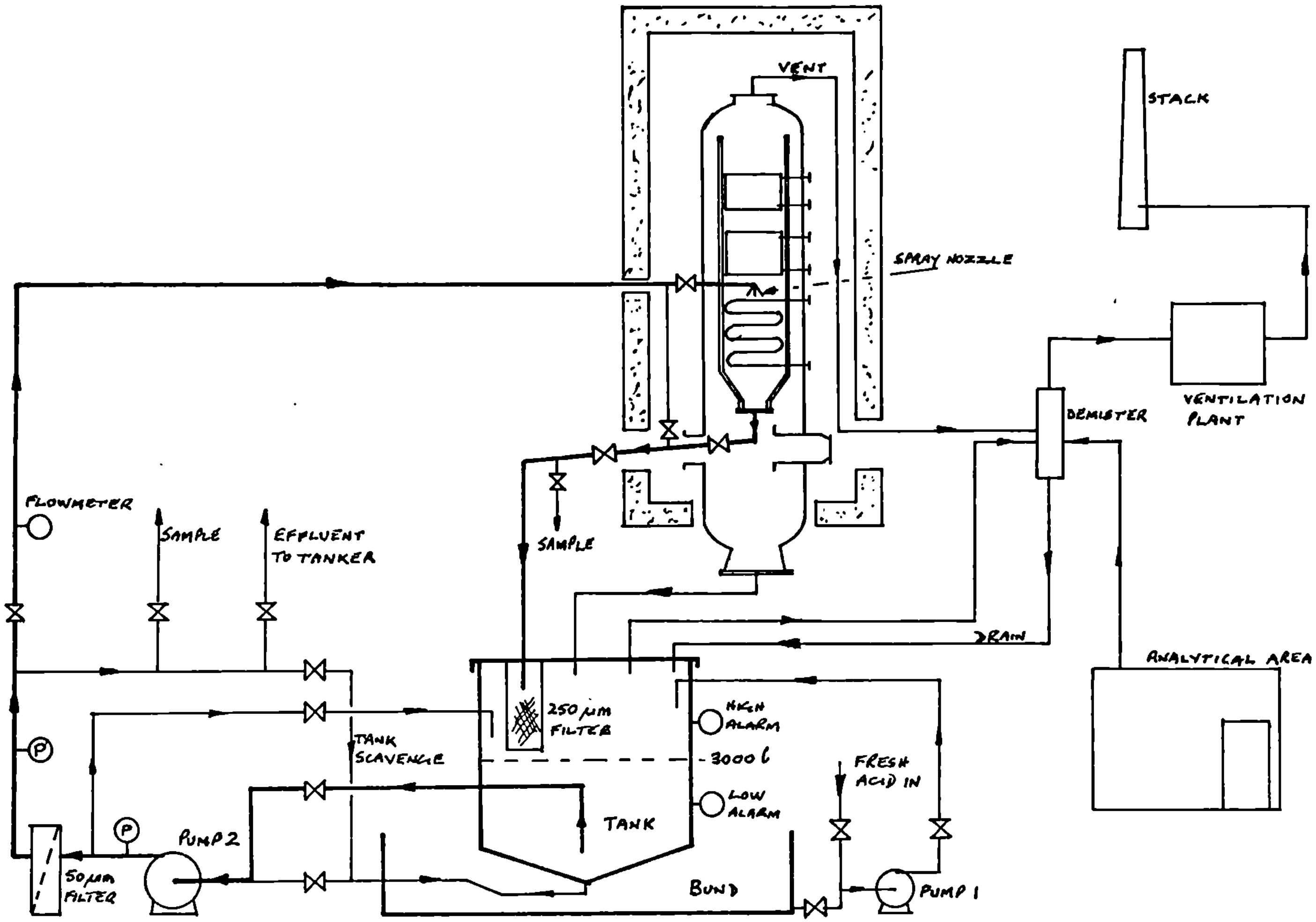


Figure 2: Schematic view of recirculating spray system facility.

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

Contractor: British Nuclear Fuel plc, Sellafield, United Kingdom
Contract N°: FI1D-0027
Working Period: July 1984 - December 1988
Project Leader: A.P. Colquhoun

A. Objectives and Scope

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

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

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

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

B. Work Programme

B.1. Detailed planning for most appropriate decontamination and dismantling, including technical specifications and safety assessments for proposed methods and plant modifications and submissions for company and regulatory approvals.

B.2. Execution of modifications, testing of equipment, rehearsing of proposed procedures on plant simulations, followed by in-situ decontamination.

B.3. Rehearsing of dismantling and packaging procedures on plant simulations, followed by in-situ dismantling and packaging.

C. Progress of Work and Obtained Results

Summary

This report covers continued development work and practical operations in the decommissioning of the mixed oxide fabrication facility.

Practical work in 1988 followed on from removal of the Ball Mill in late 1987 and was principally concerned with removal of the five glove boxes which comprised the "Ball Mill Suite". This part of the facility is shown in Figure 1.

It had been the intention to carry out this work with minimum in situ size reduction but detailed planning of operations showed that a much higher degree of size reduction would be cost effective in this specific situation.

On development aspects the Reusable Modular Containment (RMC) and associated use of Tie Down coatings continue to give good service and are now accepted as standard techniques for these operations. Similarly the use of the in situ plutonium assay equipment has also become routine, although in this case persistent electronic faults are still unsolved so that availability of the equipment has limited usage to measurements essential for nuclear safety purposes.

The Freon cleaning equipment has still not been delivered although final modifications are in hand and use within the overall project is still envisaged. Conversely the proposal to demonstrate electrolytic decontamination as part of this particular project has been abandoned.

Progress and Results

1 Planning and Safety Documentation (B1)

Operations on the main project have continued to be controlled as a series of steps, each covered by a Plant Modification Proposal. This has avoided detailed planning being carried out before the fullest possible information on a given step has become available. Items such as the Freon Cleaning Unit and the In Situ Assay equipment have been subject to detailed nuclear safety assessment and in the case of the former an operational safety appraisal of all aspects of the units operation and application.

2 Removal of the Ball Mill Glove Box Suite (B3)

This suite consists of five glove boxes as shown in Figure 1. The Ball Mill and its housing and shielding had been removed in late 1987 using a RMC structure attached at the end of the Ball Mill glove box shown.

Operations during 1988 therefore started with packaging the Ball Mill and its housing for storage and then cleaning up and dismantling the RMC structure. This was carried out by recoating with Tie Down coating and stripping the resultant layer. Problems were minimal except on the floor area which required some additional cleaning.

All units of the Ball Mill suite were then separately assayed for residual plutonium and redundant external and internal equipment was removed. An enlarged RMC structure was then built completely surrounding the units on the upper floor and a plastic sheet containment was provided on the lower floor. Within the RMC operators in pressurised suit size reduced the units into readily handled portions which were posted out for packaging. The unit on the lower floor was lifted into the RMC for similar treatment - this operation being necessitated by the integral construction.

Finally the RMC was again decontaminated and dismantled.
Table 1 contains a summary of the project so far.

3 Development Aspects
Cost Effectiveness of In Situ Size Reduction (B1)

The original intention had been to carry out minimal in situ size reduction and to enlarge the aperture in the floor to pass down items of large size. Restrictions of short term storage space on the first floor, both inside and outside the RMC forced a rethink and careful study showed that, in this specific case, the course we then chose would be most cost effective.

4 Plutonium Assay Equipment (B2)

Although now a routinely used tool this equipment is not yet robust enough for use by plant operators and persistent electronic problems are being experienced. This has limited use to essential nuclear safety measurements and other work such as following progress of cleaning or rechecking items when removed from the facility has yet to be undertaken.

5 Freon Cleaning (B2)

The unit was essentially completed by the makers early in 1988 and much time has been spent on non-active trials and in particular optimising performance of the heat exchangers which remove solvent vapour from the glove box extract system. Delivery for on plant trials is now imminent.

6 Electrolytic Decontamination (B1)

The intention to adapt an existing on plant glove box for this purpose has been abandoned on cost grounds. Subsequently AERE have undertaken a more general study of the application of electrolytic cleaning in decommissioning operations and have proposed mobile units for this purpose. This work is not practicable within the time-scale of this project however and has been abandoned in this context.

7 Tooling for In Situ Size Reduction (B3)

Following trials noted previously the tools selected have been a "Nibbler" and "Angle Grinder", which have performed most of the work described above. The crimp/shear tool has proved most effective also for severing pipework without loss of containment and techniques are being considered for extrapolating the principle to much larger sizes - although alternative methods will be needed.

Table I: Summary of important features associated to main cleaning and dismantling operations.

Operation	Whole Body Radiation Exposures mSv		Man Hours on Job		Wastes M^3		MO_2 Recovered Kg
	Process	Engineering	Process	Engineering	PCM	SLB	
Access Area Initial Clean Out	3.7	N/A	225	N/A	1.0	2.0	Nil
Engineering Refurbishment	40.5	17.3	1350	600	1.4	Nil	Nil
Liquor Recirculation	7	1.9	670	100	Nil	Nil	See Below
RVF Cabinet Stripping	6	5	560	170	0.7	0.6	"
Recovery Cabinet Stripping	1.7	1.3	220	40	0.6	0.1	"
Precipitator Cabinet Stripping	1.0	1.6	200	70	1.2	1.4	"
Liquor Transfer	4.5	Nil	450	20	N/A	Nil	2.9 Kg as Nitrate
Ball Mill and Product Offtake Preparation	20	5	1450	70	0.4	Nil	1.6
Ball Mill Emptying	1.8	Not applicable	100	N/A	Data not available		26.0
Ball Mill Dismantling and Removal	5.3	11.3	475	590	6.7	0.5	Nil
Cleaning/Dismantling RMC	6.6	N/A	1000	N/A	0.63	Nil	Nil
Assembling New RMC	6.6	0.5	1000	50	N/A	N/A	N/A
In-situ Size Reduction Upper Ball Mill Suite	4.6	9.7	700	900	3.3	Nil	Nil
Removal of Pipework using Crimp/Shear Tool in Tank Room	1.3	2.4	200	250	0.1	Nil	Nil
Removal of Inactive feed lines	2.3	N/A	350	N/A	Nil	5.5	Nil

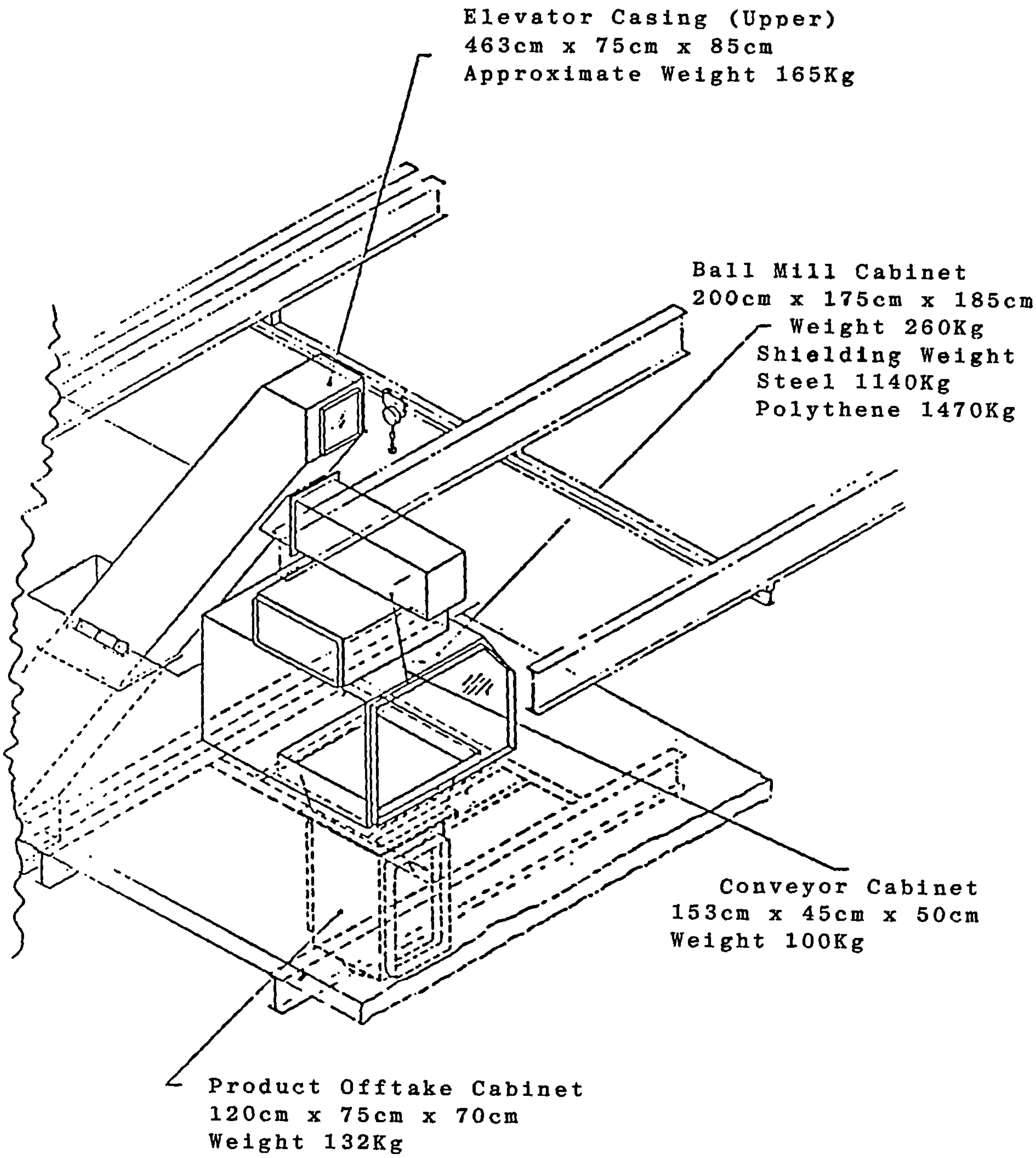


Figure 1: Ball Mill Glove Box Suite

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

Contractor: Nukem GmbH, Hanau, Germany
Contract N°: FI1D-0028
Working Period: July 1985 - September 1989
Project Leader: E. Wehner

A. Objectives and Scope

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

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

B. Work Programme

B.1. Preparation of a map of the distribution of the contamination within different parts of the fuel fabrication plant.

B.2. Determination, by analyses of representative samples, of the penetration depth of uranium and thorium in various parts of the facility.

B.3. Controlled decontamination and dismantling of the internal components and of all auxiliary equipment of the plant.

B.4. Assessment of appropriate conditions for the removal of contamination from the walls of the facility and its implementation after acceptance by the Regulatory Bodies.

B.5. Decontamination of the floors and their removal.

B.6. Testing of new decontamination procedures for less accessible parts.

B.7. Determination of the residual activity and possible further dismantling of less accessible parts.

B.8. Conditioning and assessment of the residual activity of metal waste for reuse by melting.

B.9. Large-scale decontamination of the demolition rubble based on existing laboratory-scale methods.

B.10 Minimising of the secondary waste from decontaminants.

B.11 Testing of a NUKEM procedure for container sealing.

B.12 Evaluation of obtained results.

C. Progress of work and obtained results

Summary

The exemplary removing of contaminated floors with respect to radiation protection was completed.

The determination of contamination of metal waste was performed by using gamma-spectrometry with a newly constructed barrel scanner. Work has been completed.

Progress and results

1. Decontamination of the floors and their removal (B.5.)

An area of about 2500 m^2 in NUKEM-A was covered with the PVC material 'royal' from DLW AG Bietigheim factory. The renewal had to be performed with special respect to highest mechanical performance of the material, lowest inflammability and tightness for outflowing radioactivity. These qualities are certificated for the 'royal'-cover by the German quality assurance institute 'Deutsches Institut für Gütesicherung und Kennzeichnung eV'.

Damaged material was removed after wet decontamination of the surface by water and common cleaners. Further decontamination observing the limits of the German Atomic Law for free use was not carried out because of the high costs and the uncertainty of the measurements due to the rough reverse side surfaces.

The waste was collected in 0,2 m^3 barrels and measured by a barrel-scanner combined with a gamma-spectrometer. These measurements allow declaring uranium- and thorium-contents for safeguards. Values from 0,4 to 27,6 ppm uranium-235/g waste and total activities from 3,1 to 58 Bq/g waste were measured. From these values about 0,08 Bq/cm^2 area can be estimated for the mean contamination of floors in areas where open radioactivity was handled.

Contaminated concrete floors were treated by grinding operations. In the past radioactive solutions penetrated through damaged concrete, thus contamination of earth was observed and removed. In all cases a doubtless qualification of waste is possible by gamma-spectrometry.

Economy of operation results from the comparison of the costs for decontamination plus "free use"-measurement and waste treatment costs in nuclear facilities i.e. removing of uranium and deposition of the residual. For NUKEM the second alternative is more economic.

2. Determination of the residual activity of metal waste (B.7.)

For determination of the residual activity of metal waste two different methods of measurement were investigated.

First the known measurements of surface contaminations by surface liquid flowcounter were modified to get all radiologically important data in a component-register for safeguard and authority purposes. These tables list name, former use and arrangement of components, weights, dimensions, surface areas, alpha-activities, total alpha- and beta-activities, uranium and thorium masses and enrichments. The tables will be used during the decommissioning period of NUKEM-A.

A barrel scanner combined with a gamma-ray spectrometric assembly was planned and constructed to enable a fast and economic determination of radioactivity in nuclear waste. This method is also used for the determination of the residual activity of metal waste.

Metal waste is collected in 0,2 m^3 barrels. For the measurement the barrel is positioned on a rotating table in front of a Ge-detector. This gamma-detector is automatically moved up and down during the measurement.

In order to get an individual calibration of each barrel a known activity is positioned at the center and measured by gamma-routine. From this, the total activity is calculated routinely based on the well known nuclide vector.

The equipment also allows to measure the radioactivity profile in the barrel axis.

8.7. Decontamination and Dismantling of the PIVER Prototype Vitrification Facility

Contractor: Commissariat à l'Energie Atomique, CEN Valrho, France
Contract N°: FI1D-0057
Working Period: July 1986 - September 1989
Project Leader: A. Jouan

A. Objectives and Scope

The pilot vitrification facility PIVER at Marcoule has been operated between 1969 and 1980, first in a discontinued procedure for the vitrification of waste arising from the reprocessing of graphite-gas reactor fuel elements and then for the development of a continuous procedure for the treatment of waste from reprocessing of Fast Breeder Reactor (FBR) fuel. It is planned to reuse the existing cell for a continuous vitrification procedure of waste generated by the reprocessing of FBR fuel (PIVER II).

The objective of the present work consists in a multi-stage decontamination and cleaning of the highly contaminated cell and its equipment, followed by remote dismantling of the internals, with subsequent waste conditioning and treatment on the site.

This task is to be executed in an R&D spirit, the target being the preparation of a conclusive assessment on all important technical and economic aspects of the whole operation.

A supplementary agreement concluded in 1988 provides an extension of the work programme according to the newly defined working package B.5.

B. Work Programme

B.1. Preliminary work consisting in the measurement of the initial radioactivity levels, feasibility studies for the application of appropriate techniques, preparation of dossiers for requests of authorisations for the dismantling and the transport of waste.

B.2. Cleaning and pre-decontamination of the cell and mapping of the remaining radioactivity distribution.

B.3. Introduction of supplementary telemanipulators and of various dismantling tools into the cell.

B.4. Dismounting, conditioning and removal of the heavy concrete shielded induction coils of the vitrification and ruthenium filter furnaces.

B.5. Dismantling, conditioning and removal of pipe work and other equipment, followed by high-pressure washing of the cell and mapping of the residual radioactivity distribution. The supplementary work programme concerns the design, testing and application of a plasma-arc torch with simplified and robust control for in-cell dismantling of large components.

B.6. Execution of a limited amount of direct interventions in the cell for further dismantling and decontamination (in case of acceptable decrease of the radiation level).

B.7. Conclusive assessment of the whole operation of decontamination and dismantling of the PIVER I plant, including applied technologies, economic aspects (costs, man-power) and a balance of the generated radioactive waste.

C. Progress of Work and Obtained Results

Summary

In the decommissioning programme for the PIVER pilot vitrification unit, renovation of the premises and existing equipment was well advanced at the end of 1987, as was the development and delivery of new equipment.

During 1988, various remote handling devices (ANTOINE, MT 200) became fully operational. The direct waste removal path became available and was used to facilitate the operations despite stringent safety requirements.

At the same time, after careful consideration of work organization and dismantling methodology, highly effective organizational structures were set up.

Despite several incidents resulting in delays, decommissioning work advanced steadily in the cell and 18 waste containers were removed.

Progress and Results

1. Setting Up Organizational Structures (B.1.)

The difficulties encountered, due largely to the poor condition of the facility, led us to set up an effective organization during the first half of 1988. This involved the following aspects:

Personnel Organization. The work crew comprises:
- a project chief
- a daytime work crew (one engineer and five supervisory technicians)
- two 8-hour shifts (one shift leader and three operators, including one decontamination specialist)
- an SPR technician for radiological protection (contamination and irradiation monitoring).

Scheduling. Tasks were scheduled as accurately as possible after an examination of the PIVER installation layout drawings. The monthly work schedule defines the program in detail, specifying the tasks assigned and allowing adequate preparation (PIVER Decommissioning Task Reports, personnel and equipment requirements, specification of waste drum or container contents).

Repairs. Definition of repair operations necessary for existing equipment.

This was only possible because of the extensive knowledge of available equipment items and their characteristics, accumulated during the preparation phase and the multiple delays. Execution times for individual tasks could thus be realistically estimated (e.g. cutting & handling time). The result was the definition of a standard "time per waste container" figure, and an overall schedule used to prepare the monthly work schedules.

2 Preparation - Operational Equipment (B.3.)

2.1 Handling and Remote Manipulation

CAROLINE Heavy Robot: The CAROLINE robot manipulator was returned to the cell in December 1987 after multiple failures. It was operated satisfactorily with minor incidents until August 3, 1988, when the robot arm fell 6 meters to the cell floor during a handling operation. CAROLINE was then fitted with a replacement hook suitable for handling drums and has been used since then to carry waste drums out of the cell.

ANTOINE Intervention Device: Nonradioactive tests were completed at the beginning of 1988, and ANTOINE was moved into cell 74 at the end of March. This required six weeks of work, sometimes under difficult conditions considering the size of the unit and the constraints of a radioactive environment. Eight persons participated in this operation, sustaining a collective integrated dose of 4.02 mSv (402 mrem) with a maximum individual dose of 1.1 mSv (110 mrem).

MT 200 Master-Slave Remote Manipulators: After a remote operator station was set up along the cell front wall in 1987, two MT 200 master-slave manipulators were installed in September, 1988. This complex operation, during which the slave arms were installed in the radioactive zone, was conducted by *La Calhène* (the manufacturer) with the assistance of CEA personnel.

2.2 Waste Removal Paths

Normal Path: Waste Conditioning Station (Figure 1). Some minor changes were implemented in the facility after testing, and a few design changes were incorporated in the containers for waste transfer from cell 74 to the conditioning station. With the availability of the direct route, this path has not been used for radioactive transfers, but could become operational if necessary during the first half of 1989.

Direct Path: via the Cell Transfer Lock (Figure 2). The direct waste removal path is now operational.

a) Waste Removal

Waste material is placed in a 100-liter drum in cell 74. After sealing, the drum is placed on the transport cradle mounted on the CAROLINE heavy remote manipulator.

The drum is carried by CAROLINE into the entry lock for dose rate measurement, and then into room 712 where a special crane transfers the drum into a concrete ANDRA shell in a double vinyl wrapper.

Once the concrete shell is filled with one to three drums, depending on the measured dose rate, it is closed and removed. The first vinyl wrapper is removed in room 712 and the second in room 713. On removal from the building the shell is sent to the Marcoule concrete encapsulation facility.

b) Personnel Action - Monitoring

After a safety review of the waste removal procedure, operations in room 712 were restricted to a minimum (entry of the ANDRA shell, sealing of the shell, removal of the vinyl wrapper and radiation monitoring). The other operations in room 712 are remote controlled from the PIVER console. Final radiation monitoring after removal of the second vinyl wrapper is performed in room 713.

A Waste Conditioning Report is prepared on the basis of the monitoring measurements for waste management by ANDRA.

2.3 Additional Equipment

Other equipment items developed and installed to facilitate cutup and waste removal operations include:

- a 5-ton pallet truck for transporting the concrete shells
- a 300 kg crane used to carry waste drums in room 712
- a 2 cm thick lead cask that covers the concrete shell when it is transferred to the concrete encapsulation facility in compliance with irradiation standards applicable to transportation at Marcoule
- 100-liter waste drums
- a plasma torch cutting station for dismantling bulky items (tanks, evaporators) will be installed in the cell early in 1989.
- a fire extinguishing system in cell 74.

2.4 Estimation of Cell Activity

The activity in the cell was evaluated in October 1988 by CEA specialists from Saclay using a γ teletopography method. The recording camera was introduced into the cell through a penetration sleeve after removal of the MT 200 telemanipulator.

Data processing lasted until the end of the year, at which time the operator was provided with views of the cell showing hot spots together with an estimate of the radioactivity levels at these spots. The highest level was found in the recovery pan channel where the dose rate exceeded 200 $rad \cdot h^{-1}$ at one meter.

3. Cleanup Operations (B.4., B.5.)

The organization structure (§ 1) and in particular the detailed operation schedules and preparation documents are used to assign detailed task instructions to individual members of the work crews. A Waste Conditioning Report is filled in for each waste container under the Quality Assurance programme defined jointly by ANDRA and the CEA.

Identification of Cutting Points

A 3-D imaging system using photographs taken through the cell viewing ports allows comparison with the cell layout drawings to ensure accurate identification of pipes to be cut and those that must be left intact for operation of the fission product liquid waste storage facility locate beneath cell 74.

Waste Removal

A total of 24 ANDRA shells had been removed as of the end of 1988 (including 6 in 1987) as well as 7 glass canisters stored in cell 74. Major equipment removed from the cell included the furnace inductors and a few pots. About 20 waste drums containing pipes and cables have been conditioned. Operating wastes accumulated in the cell throughout its service life were also removed during 1988. The total activity removed in 1988 was about 1.1×10^{13} Bq (300 Ci).

Integrated Doses

Waste removal operations resulted in a total integrated dose of 14.12 mSv (1412 mrem) for the 18 ANDRA containers removed in 1988. Eight persons were involved in these operations.

Work done primarily in room 712, notably the installation of ANTOINE and miscellaneous repairs or cleanup operations following accidental contamination accounted for a collective integrated dose of 65.03 mSv (6503 mrem) for an average of 12 persons.

Incidents

One contamination incident occurred in room 712 during removal of unconditioned waste material. Various equipment malfunctions also required operator intervention and resulted in schedule delays.

4. PIVER Liquid Waste Management

The PIVER liquid storage facility for concentrated fission product solutions was restored to operation in 1988 in compliance with the requirements imposed by safety authorities, notably restoration of the storage tank pulse agitation system and installation of emergency backup systems on the storage facility cooling unit.

The first batches of fission product solutions were transferred from the Marcoule Pilot Facility to PIVER in May 1988, and the total figures for 1988 were 1428 liters containing 5.3×10^{15} $Bq \cdot m^{-3}$ representing 144 $Ci \cdot l^{-1}$.

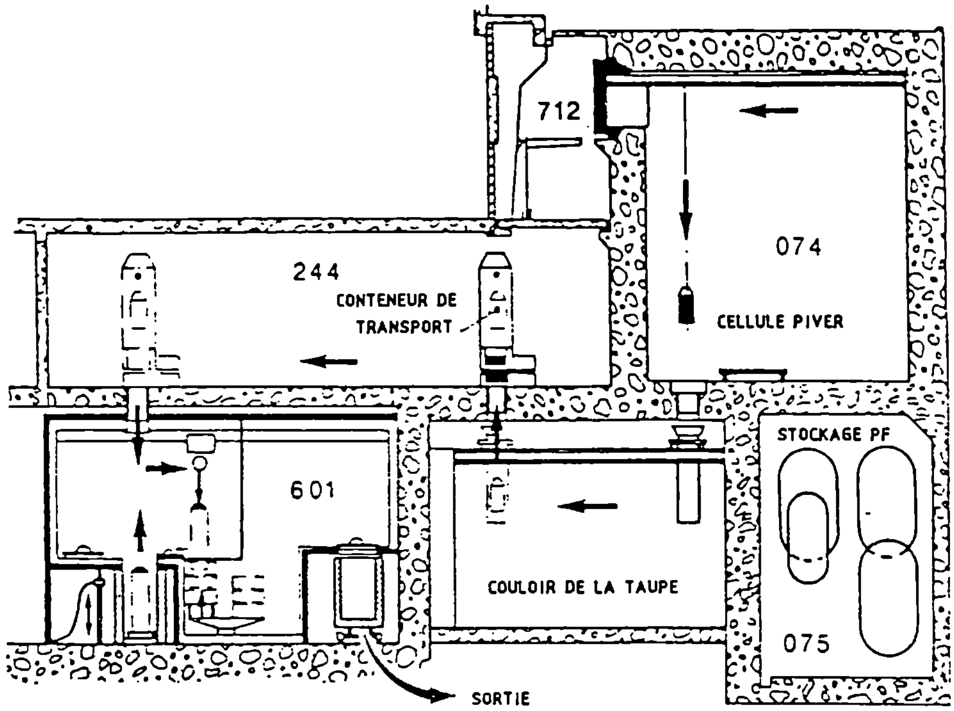


Fig 1 NORMAL EXIT ROUTE THROUGH WASTE CONDITIONING STATION 601

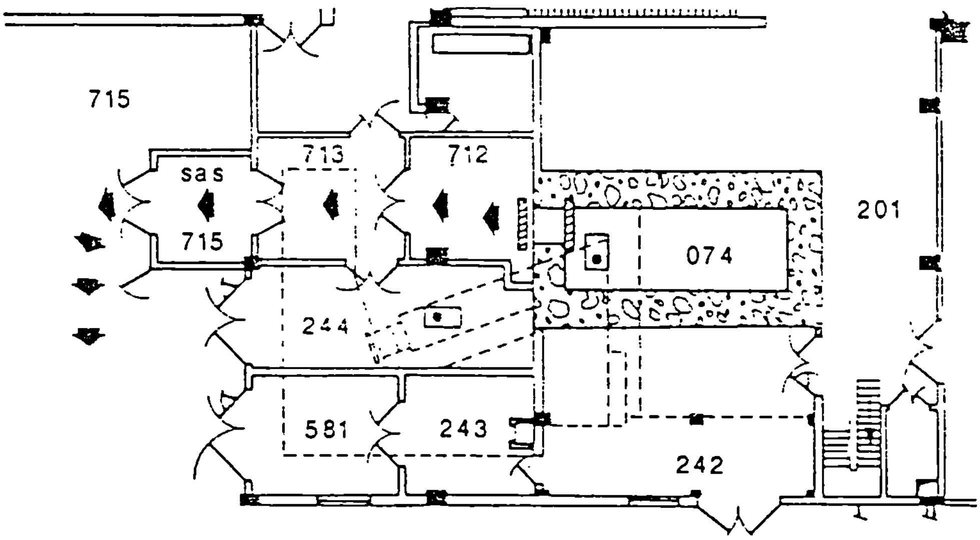


Fig 2 DIRECT EXIT ROUTE THROUGH LOCK-CHAMBERS 712-713 TO LOCK-CHAMBER 715

8.8. Dismantling, Partly In-situ, of a Glove-Box Structure of a Mixed-Oxide Fuel Plant

Contractor: Belgonucléaire, Dessel, Belgium
Contract N°: FI1D-0058
Working Period: January 1986 - August 1989
Project Leader: J. Draulans

A. Objectives and Scope

The decommissioning of standard-size plutonium glove boxes has been performed in several countries for many years. However, the dismantling of very large alpha-contaminated units is still a rather exceptional task.

Plutonium research laboratories and mixed-oxide fuel fabrication plants have to be dismantled in the near future. During these dismantling tasks, severe problems will arise with the decommissioning of large glove-box structures containing big and heavy equipment. Such units have to be partially dismantled on place and then transported to an ad-hoc installation for further dismantling and final disposal.

The techniques used until now for the conditioning of standard-size glove boxes are not directly applicable in the case of a complex glove-box structure, to be dismounted partially on place.

The objective of this research is to develop adequate techniques for decommissioning and partial dismantling of large alpha-contaminated units and to demonstrate their feasibility by carrying out such a task in respect to safety and cost.

This work complements contract N° FI1D-0024 "Decommissioning of very large glove boxes" of the present research programme.

Working packages B.2. and B.3. of the initial work programme had to be abandoned for licensing reasons. In a supplementary agreement concluded in 1988, the present replacement programme was defined, concerning studies into the feasibility of Pu/U oxide-contaminated glove box cleaning by blasting with deep frozen CO_2 grit.

B. Work Programme

B.1. Conception of a work procedure, preparation of special equipment, conditioning and dismounting of the glove box.

B.2. Conception and execution of tests into cleaning with deep-frozen CO_2 grit blasting on non-active glove box samples.

B.2.1. Conception and construction of an experimental facility.

B.2.2. Definition and preparation of non-radioactive samples simulating a Pu/U oxide contamination.

B.2.3. CO_2 grit blasting tests with subsequent analysis of decontamination effects.

B.2.4. Comparative tests with water, freon and CO_2 grit and safety analysis with respect to an application on radioactive components.

B.3. Conclusive assessment of pre-cleaning by blasting of deep frozen CO_2 grit of Pu/U oxide-contaminated glove boxes before dismantling.

C. Progress of work and obtained results

Summary

The programme has started on several tasks, the collection of existing data resulted up to now in a limited quantity of useful information, preliminary tests have been performed to define an experimental equipment to perform solid CO_2 grit blasting, other tests have been performed to make non-active samples simulating PuO_2 contaminated plexiglas, the penetration depth of PuO_2 particles into plexiglas has been defined on some samples.

Progress and Results

1. Modification of the Programme, background

As the original programme had to be abandoned for licensing reasons a replacement programme has been proposed and accepted.

The replacement programme covers the possible use of frozen CO_2 grit as a cleaning agent - by blasting - on U-Pu-oxide contaminated glove box walls.

Indeed the techniques used until now for both the cleaning of alpha-contaminated glove boxes in service and for a primary decontamination of such glove boxes before decommissioning have either a low efficiency or create a considerable amount of contaminated solid or liquid waste.

The use of products in liquid or solid form, who evaporate during use or after being used, results in gaseous effluents, relatively easy to filter. The CO_2 has as supplementary advantage that it is environmental friendly. On the other hand the use of a very cold product in a glove box results in specific problems as there are the behaviour of gaskets, gloves, window panels etc... in contact with cold products.

Both aspects of the problem namely the efficiency of CO_2 grit as a cleaning agent by blasting and the behaviour of glove box materials in contact with cold products have to be examined.
Both aspects have to be studied before any test can be performed in a contaminated glove box.

The replacement programme therefore is limited to activities to be carried out on non contaminated products.

2. Development of a procedure for grit blasting with deep frozen CO_2 and application on non-active materials including a comparison with water and freon cleaning. (B.2)

Several tasks have been started since September 1988. The work started with the consultation of several data banks. The results however are poor and contradict each other in some cases.

Solid CO_2 pellets, having a diameter between 3 and 15 mm as well as particles having a diameter between 0,5 and 1 mm as microparticles, are used. Furthermore dry ice particles are used alone or either with water-ice particles or with a surface active agent. Some information limits the use of solid CO_2 pellets to the removal of smearable contamination, others seem to use these pellets as a replacement for sand in the blasting method.

Work is under way to define a preliminary experimental equipment to perform feasibility tests. Solid CO_2 will be purchased as cigarette-type cylinders. They are delivered at the BN plant in polysterene boxes. Fig. 1 gives schematically the layout of such a preliminary equipment.

These solids have to be cooled before milling in order to increase their lifetime in open air. This cooling step is performed by means of a small stainless steel cylinder filled with the solids placed inside a small dewar vessel containing liquid nitrogen.

After cooling during several minutes, the time period depends on the required solids lifetime after milling, the solids have to be broken. Up to now this operation is performed by means of a plastic hammer. The broken particles are then sieved and fed into the blasting device supply vessel.

The blasting device used up to now is a standard one with a slight adaption of the suction tube in order to use a small supply vessel.

The solid particles are projected by means of a dry air stream.

The whole operation is carried out in an enclosed hood, fed with a dry air atmosphere in order to reduce condensation on all cold parts and to avoid sticking to each other of the solids due to the formation of an ice layer on the solids originating from the humidity in the air.

Small samples have been treated up to now in order to define the required adaptions to the equipment. Problems did arise on the level of the manual milling, on the level of the supply tube and of the spray nozzle due to blocking.

Two different actions have been taken to define representative non-active samples for the programme. First an attempt was made to obtain information of the penetration depth of PuO_2 particles in plexiglass.

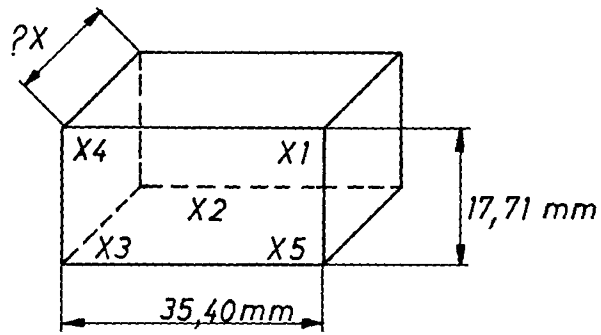
Heavily contaminated plexiglass samples have been polished in several steps, an autoradiography was taken before and after each step - up to the moment that no activity was shown by this radiography method. Penetration depths between 0.1 and 0.2 mm have been found.
The results of 1 sample are given in Table 1. (see also figure 2)

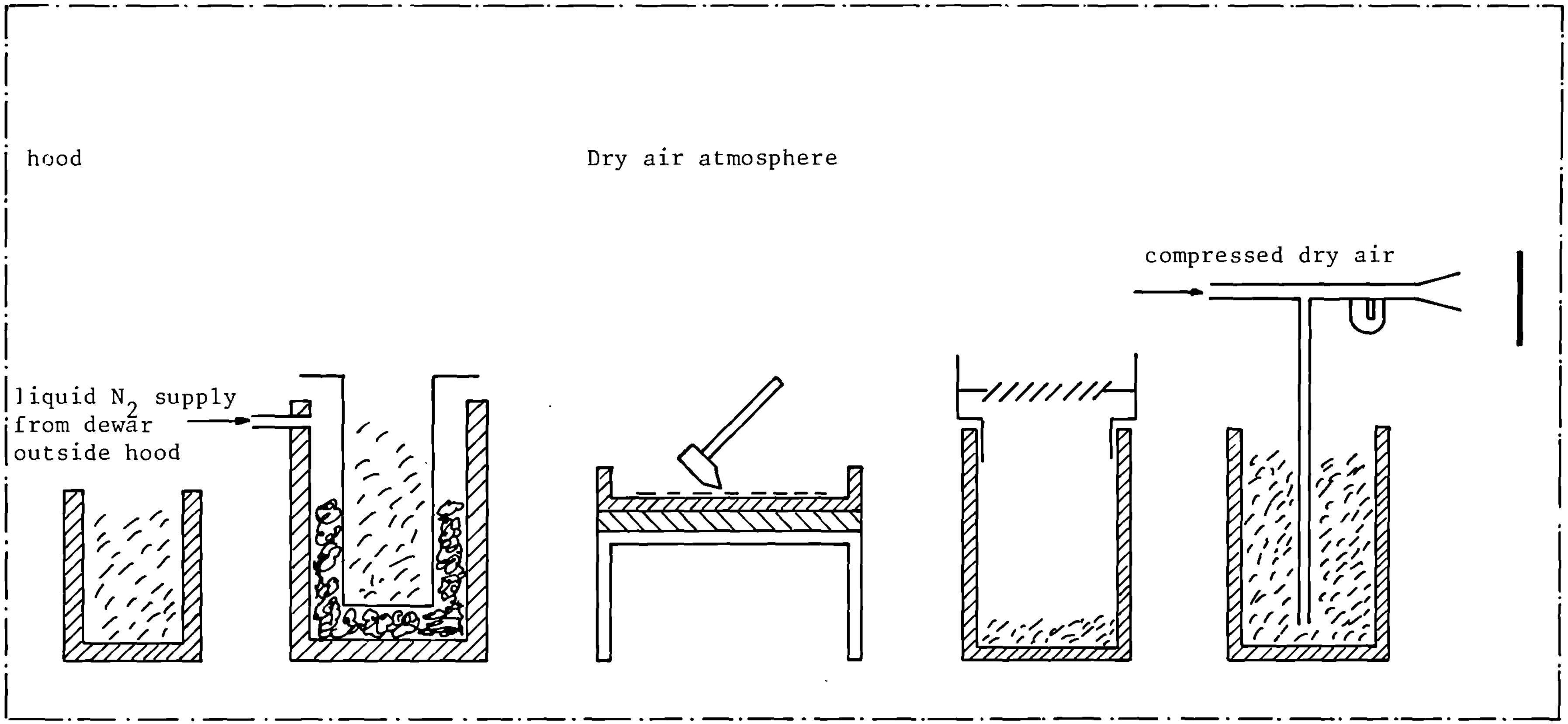
At the same time preliminary tests have been made to make representative non-active plexiglass samples. The best result at this moment is obtained by means of blasting hot metal slag with a diameter between 0.2 and 0.8 mm on plexiglass. The hot particles fixe themselves in the plexiglass. This blasting operation has to be performed in an enclosure to protect the operator.
CO_2 blasting on such samples is scheduled for the next months.

Table 1 : Wall thickness removed during 3 successive polishing operations (sample 1)

Measuring places on the sample (see scheme)	x_1	x_2	x_3	x_4	x_5	Remark
Sample thickness at the start of the test Average : 9,48	9,48	9,49	9,49	9,46	9,47	
Thickness removed during first polishing operation Average : 0,084	0,07	0,10	0,11	0,06	0,08	
Thickness removed during second polishing operation Average : 0,056	0,06	0,05	0,05	0,07	0,05	
Thickness removed during third polishing operation Average : 0,042	0,05	0,05	0,03	0,05	0,03	
Total removed in 3 operations Average : 0,182	0,18	0,20	0,19	0,18	0,16	

Remark : the autoradiography taken after the 3rd polishing operation showed a surface nearly free of radioactive particles.





Polysterenebox with CO_2 solids

dewar for cooling the CO_2 solids

table for milling the CO_2 solids by hammering (mill still to be defined)

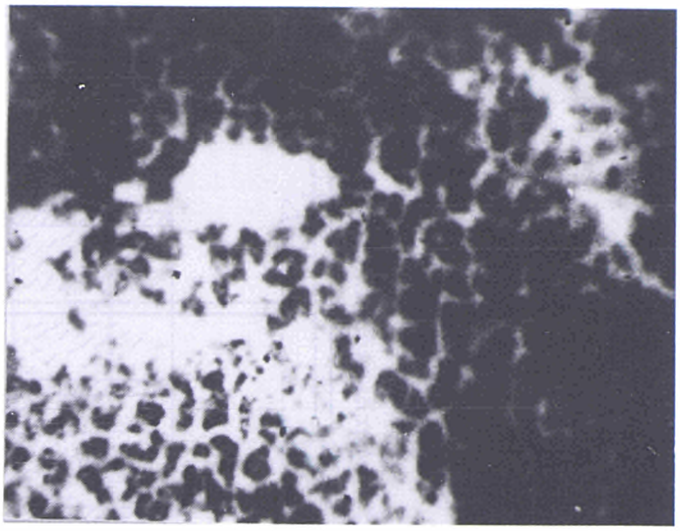
sieve

blasting device

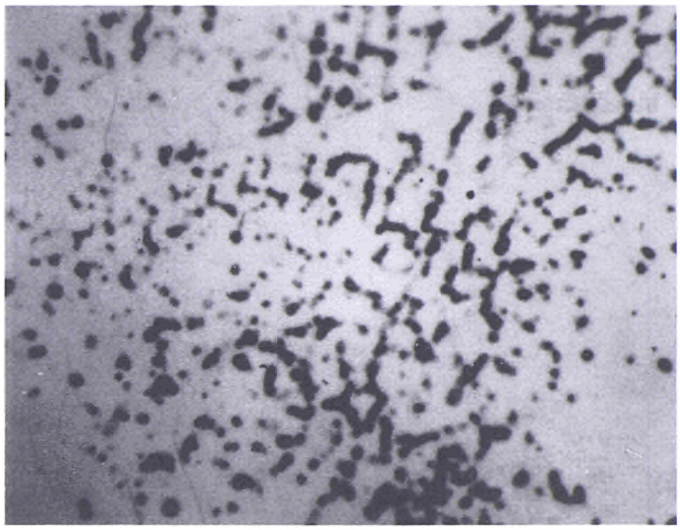
sample

FIG. 1 SCHEME OF THE PRELIMINARY EQUIPMENT

1° Autoradiography
before polishing
contact time : 30 sec.



2° Autoradiography
after first polishing
operation
contact time : 30 sec.



3° Autoradiography
after second polishing
operation
contact time : 120 sec.

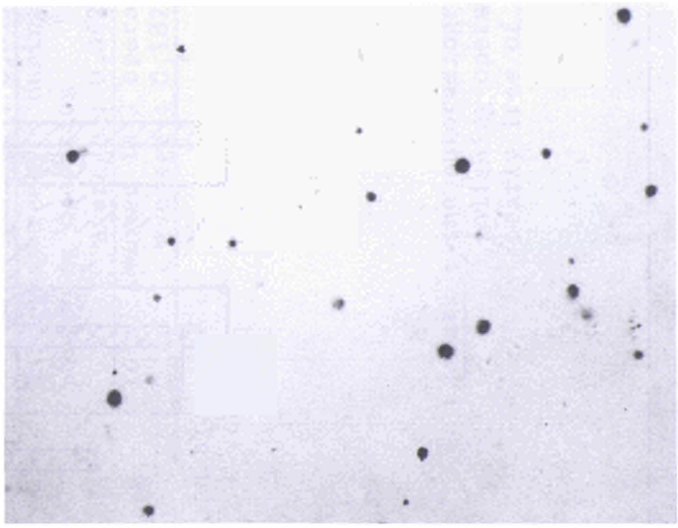


Fig. 2
Autoradiographies of the sample before and after polishing

8.9. Melting of Radioactive Metal Scrap from the KRB-A Plant

Contractor: KRB Gundremmingen GmbH, Gundremmingen, Germany
Contract N°: FI1D-0059
Working Period: July 1986 - October 1989
Project Leader: W. Stang

A. Objectives and Scope

Radioactivity homogenisation and volume compaction of low-level radioactive scrap can be achieved by melting. Then, depending on the average specific activity, the metal can be released in general to the nuclear market, or stored for final disposal. However, melting in standard foundries without controlled containment atmosphere has to be limited to scrap with low specific activity (< 74 Bq/g), and large-scale experience with melting of higher-level radioactive metal scrap is presently not available.

The work programme was, for technical and economic reasons, revised in a supplementary agreement concluded in 1988, after execution of items B.1. to B.2. The revision aims at gaining experience by large-scale melting of about 300 t of metal waste from the KRB-A decommissioning, with radioactivity levels up to 500 Bq/g. The melting will be performed in an induction furnace (capacity ca. 3t) in a controlled zone at the site of Siempelkamp Giesserei Krefeld (SRG), acting as subcontractor, the radioactive waste being transported from KRB to SRG.

The study is expected to result in a statement whether the above procedure has a potential for large-scale application.

B. Work Programme

B.1. Assessment of proposals for services from external contractors, mainly concerning the leasing of an induction melting furnace.

B.2. Definition of a work procedure, including the selection of representative components for melting tests and of appropriate techniques for decontamination, dismounting and cutting, the definition of a procedure for the installation and operation of the melting furnace, and a preliminary planning for health physics protection.

B.3. Preparation of licensing procedures for the installation and operation of a melting furnace on the site of SRG/Krefeld.

B.4. Execution of the melting programme on the site of SRG/Krefeld.

B.4.1. Conception and construction of a facility for the melting of metal scrap up to 500 Bq/g.

B.4.2. Study into the nuclide distribution during melting of contaminated and activated scrap.

B.4.3. Nuclide-specific study into activity releases during two sequential melting processes.

B.4.4. Casting of waste disposal packages containing higher-level radioactive (500 Bq/g) components embedded in a matrix cast out of lower-level radioactive scrap ("Onion package").

B.4.5. Melting of slag and sawing chips arising from dismantling processes.

B.5. Evaluation of the above melting processes on the basis of laboratory analyses and conclusive assessment of the potential for large-scale application.

C. Progress of Work and Obtained Results

Summary

In the period of reporting the working intensity was to prepare detailed plannings, to demand licenses and to supply the components for the plant. A medium frequency furnace, which is needed for that kind of work, was bought. Regarding the furnace some conversion works were necessary caused by radiation protection measures. The required licences were proposed. It is to be expected that they will be granted in a short time.
For the supply of the filtration facility a specification was worked out which provides all requirements according to prevailing regulations.

Progress and Results

1. Supply of a melting furnace and optimization of supplier offers for the filtration apparatus (B.1.)

Melting furnace

The meltings have to be executed with a medium frequency furnace to be able to start the melting process from a cold status without starting block.
After checking the different offers the following medium frequency component was provided meanwhile:
Manufacturer: BBC; type S 1/2, ITMK/6000/2000
To minimize the radiation exposure of the operation staff a transposition of the control panel from the furnace area to the operation desk, which will be assembled in the new melting-hall, is necessary.

Inquiry on the filtration system

For giving all qualified suppliers the same boundary conditions the mandatory requirements with regard to radioactive emission were presented in a specification. This specification was distributed as a recommendation to all filter manufacturers.

2. Detailed plannings (B.2.)

As mentioned in point 1. above a transposition of the control panel is necessary. To execute this transposition the needed detail plannings as well for the optimization of the loading machine (furnace) as for the filtering facility and the furnace capsulation were partly elaborated. Especially the planning of the filtration apparatus turned out to be complicated, because lots of supposes must be fulfilled. The filtration facility shall consist of three filter units, each with three different types of filters. The first unit will be directly connected to the furnace and will consist of three different types of filters as follows:

- Cyclone filter
- Bag-filter
- Hepa-filter

The third filter (Hepa-filter) of these units must not be warmed up over 80 degrees Centigrade, because otherwise - in case of higher temperatures - damage could occur.
Filter unit 2 shall suck off the furnace capsulation; filter unit 3 shall suck off the hall. The chosen combination is to guarantee that by use of controlled ventilation flaps enough fresh air will be led to. All three filter units are exhausting into the same chimney. The actual maximal emission rate must be lower than the allowable value according

to the prescriptions of the radiation protection regulation. The exchange of the bag-filters and the Hepa-filters must be able to be executed free of contamination. The bag-filters are provided as pneumatically cleanable filters. Arising dust must be collected in appropriate dust collectors. Also the level-indicators of these dust collectors have to be optimized because of the low density of dust the control instruments, which react on vibration, could fail. To minimize expenditure all applied components must be standard material. Already existing components are to adapt as far as possible. The requirements for the filtration system are presented in a constructional sketch (figure 1).
During the period of reporting the conceptional planning of rooms, for the furnace including control panel, store and the room(s) for measuring devices was completed (figure 2).

3. Pre-planning of radiation protection measures (B.2.)

By handling radioactive materials a lot of laws and regulations have to be observed. Especially the prescription on radiation protection is to be mentioned. One important issue of planning is the integration of radiation protection measures, for example protective screens. For reaching this aim a specification, which has to be further developed during the construction progress, was worked out.

4. Proposing of the requisite licenses (B.3.)

To install and operate the furnace some licenses have to be obtained. These licenses were proposed to the "Regierungspräsident Düsseldorf" by SGR in the period of reporting. The following FRG-regulations are meant:
- BImSch, Article 19
- StrlSchV, Article 3

After some pre-discussions the granting of these licenses is expected in the current period of reporting.

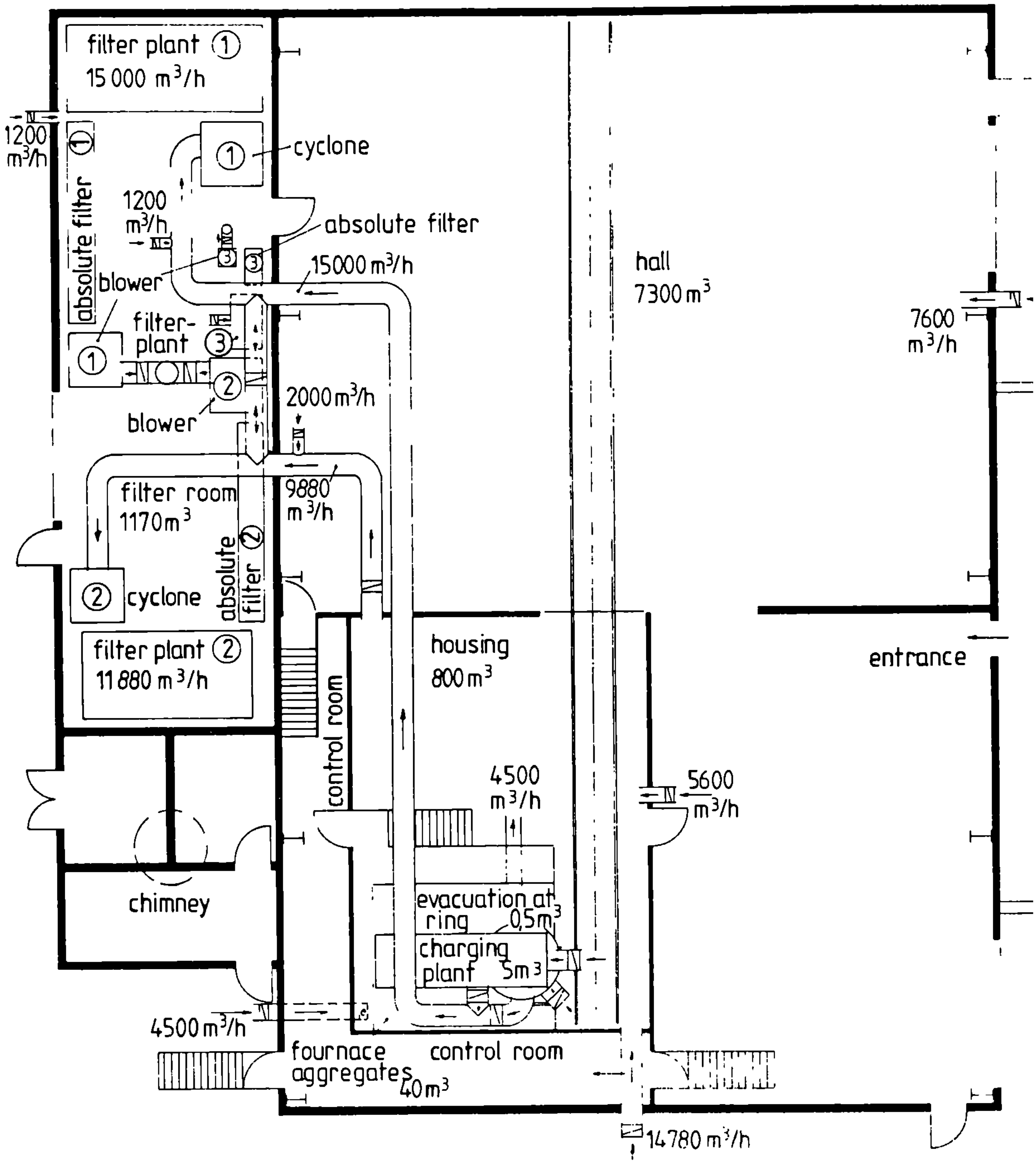


Figure 1 Lay-out of the filter facility

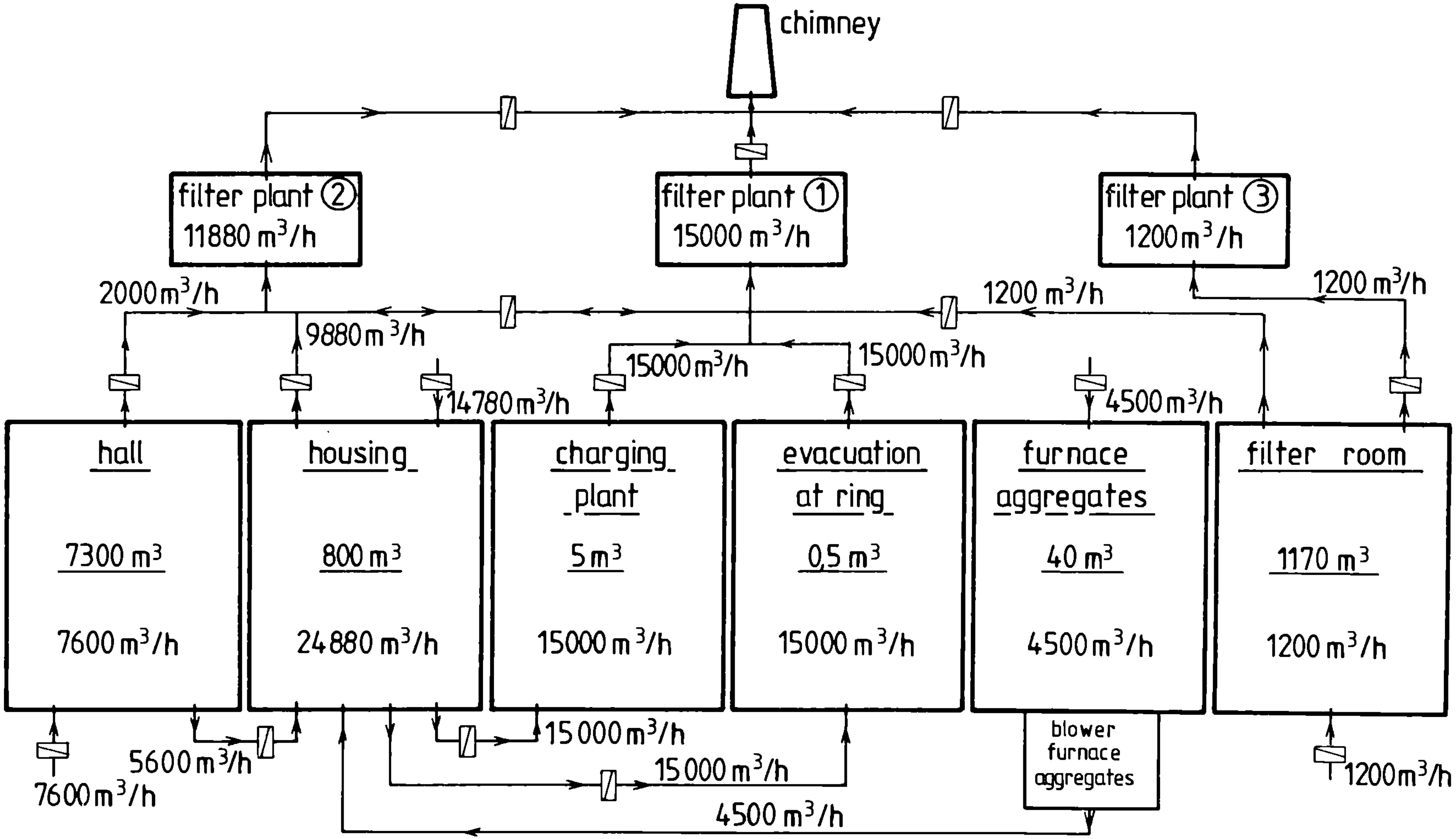


Figure 2 Scheme of the filter facility

8.10 Volume and Plutonium Inventories before and after Dismantling of a Mixed-oxide Fuel Plant

Contractor: Commissariat à l'Energie Atomique, CEN Cadarache, France
Contract N°: FI1D-0060
Working Period: May 1986 - September 1989
Project Leader: M. Beche

A. Objectives and Scope

Several plutonium processing lines belonging to the Fuel Fabrication Complex of SFER-Cadarache (Service de Fabrication et d'Examens Radiométallurgiques), have to be dismantled after long service, in the frame of plant modernisation. After disconnection from the ventilation ductwork and external utilities, the glove boxes and associated equipment are transported to the Waste Treatment Service of the Center, SPPC/SAR (Service de Protection, Prévention et Contrôle, Section d'Assainissement Radioactif) for dismantling, compacting and embedding in cement.

The importance of the plutonium quantities that have been processed in these now obsolete lines, the volume of the glove boxes to be dismantled and the variety of their functions make it of general interest to establish a balance of the volume and plutonium content of the arising wastes. These data will enable the evaluation of the waste management problems involved in the decommissioning of an industrial-size mixed-oxide fabrication facility as well as in defining novel design criteria for the construction of new facilities. Their collection and evaluation is the main objective of this research activity.

A conclusive assessment will give an answer as to whether the applied measuring and waste compacting techniques are appropriate for the present decommissioning task.

B. Work Programme

B.1. Determination of the exact volume of the glove boxes to be dismantled and description of the associated equipment.

B.2. Determination of the exact volumes of the wastes generated by the dismantling.

B.3. Preliminary estimation of the residual plutonium in the glove boxes and associated equipment before dismantling.

B.4. Measurement of the plutonium contained in the waste drums after dismantling.

B.5. Improvement of the existing plutonium monitoring techniques.

B.6. Preparation of a conclusive assessment of the applied measuring techniques, as well as a balance of involved volumes of waste and of masses of plutonium.

C. Progress of Work and obtained results

Summary

The work programme has been delayed for about twelve months due to technical reasons.
In 1988, dismantling operations have only concerned the last part of the fuel transfer tunnel contained in LAB.2 of Building 258 (SFECPu) with the same procedures as described in our previous reports. The resulting waste drums have to be measured using a passive neutron technique because the neutron activation monitoring system commissioned by the waste treatment service of CEN-CADARACHE (SPPC-SAR) will be operative only on the end of 1989.

Three large boxes containing mechanical presses for pelletizing have been cleaned and disconnected in LAB 2 and LAB 9 of Building 258; preliminary neutron measurements have been performed by means of a mobile frame supporting neutron counters, which was especially set up for this operation.

Progress and results

1. Fuel transfer tunnel interconnecting LAB 2, 3 and 4 of Building 258 (B.1., B.2., B.3., B.4.)

The last part of the tunnel, located in LAB 2, has been dismantled; this operation had been delayed because of budget problems in 1987. This part consists of 5 lengths (parallepipedic modules) with a total volume of about 6 m^3. The techniques of estimation and preliminary clean-up have been described in our previous report. Plutonium content in these segments was measured with total neutron counting system and estimated to be 36 g - Afterwards,they have been transported to SPPC-SAR, where they have to be treated as well as the other parts of the tunnel.

These other parts of the tunnel (21 voluminous packages) had been dismantled in 1987 and have been treated this year in the size reduction cell of SPPC-SAR : decontamination by wiping out with moistened cotton, cutting down to small pieces, disposal in metallic drums of 100 liter capacity. These drums have been sent back to Building 272 for plutonium assaying with coincidence passive neutron apparatus.
The balance is obtained as follows :
- number of metallic waste drums : 55 containing 9,4 g Pu,
- number of burnable waste drums : 13 containing 10,0 g Pu.
- From these 68 drums, 17 are estimated to be beneath the shallow land burial limit (370 Bq/T); that should be confirmed by an extra plutonium assaying performed with a neutron activation monitoring system, operated by a specialized Departement (DRP) in Cadarache.

2. Glove boxes containing presses in Building 258 (B.1., B.3.)

Three large glove boxes, two in LAB.2, one in LAB.9, containing mechanical presses used for fuel pellitizing, have been disconnected from the ventilation ductwork and conditionned inside PVC envelopes, after a cursory clean-up and powder recovery (UO_2-PuO_2).

- Volume of the glove boxes :
  n° 093 : 2 m x 1.2 m x 1.2 m, V = 2.9 m^3,
  n° 025 : 2 m x 1.6 m x 2 m , V = 6.4 m^3,
  n° 026 : 2 m x 1.2 m x 2 m , V = 4.8 m^3.

- Approximative weights : about 2 tons each.
- Quantities of recovered powder :
  n° 093 : 2 kg of mixed oxide with 400 g Pu,
  n° 025/n° 026 : 1,1 kg of mixed oxide with 250 g Pu.

Because of the large size of these glove boxes, it was not possible to use the routine apparatus for plutonium assaying in voluminous packages, described in APR-86. So a special mobile frame (h = 3.35 m, 1 = 2.05 m) upon railway tracts was set up in LAB.18 of Building 258, where the background level is low. This frame is supporting eight ^{3}He total neutron counters inside moderator blocks. The system was calibrated using glove boxes already measured with the routine apparatus. Plutonium content of the three glove boxes was still found too high (about 200 g) to send them to SPPC-SAR for dismantling. So it was decided to leave the glove boxes at the Building 258 and to dismantle them on place, inside an alpha-tight tent, which should be set up in 1989.

8.11 Decontamination, before Dismantling, of the Primary Coolant System of the RAPSODIE FBR

Contractor: Commissariat à l'Energie Atomique, CEN Valrho, France
Contract N°: FI1D-0061
Working Period: April 1986 - June 1989
Project Leader: J.R. Costes

A. Objectives and Scope

The large-scale decontamination of FBR sodium loops is an original task, since only a limited number of results with laboratory-scale work is available, so far.

The principal aim of the present contract is the development of an appropriate decontamination procedure and its application to the primary loops of the RAPSODIE FBR in the framework of its Stage 2 decommissioning.

The procedure is developed in a pilot facility, allowing to treat representative samples and providing the selection of an efficient decontaminant, but also with special care in minimising and treating the secondary waste.

After disconnecting the pipework from the main vessel, pipes will be treated by circulating chemical solutions, and containers by spraying liquids or gels.

B. Work Programme

B.1. Characterisation of the primary reactor loops to be decontaminated, including size and nature of structures and type and degree of contamination.

B.2. Construction and commissioning tests of the pilot decontamination loop GROLABO.

B.3. Decontamination tests in the GROLABO facility, aimed at an optimisation of procedures and of secondary waste treatment.

B.4. Safety analysis for the decontamination of the primary reactor loops and ordering of needed equipment.

B.5. Adaptation of the primary reactor loops, including isolation from the reactor vessel and equipment with specific components and instrumentation.

B.6. Assessment of the above developed decontamination procedures and decontamination of the isolated primary reactor loops.

B.7. Treatment of effluents.

B.8. Final assessment of obtained results and recommendations for future work.

C. Progress of work and obtained results

Summary

During the last year, the systems were prepared for decontamination ; this included isolating the reactor vessel which will not be affected by the operation. Additional equipment items were designed and are now being manufactured. The safety report discussing the risk analysis and the planned operating procedure was also submitted to the regulatory authorities.

Progress and Results

1. Modification of Rapsodie systems for decontamination (B.5)

This work was partly completed before alcohol rinsing, for which the system had to be in a configuration similar to the one planned for decontamination. The alcohol rinse was not specified in the CCE contract, but was carried out to eliminate any traces of residual sodium from the main sodium reservoir (REna 300) and all the sodium lines.

1.1 Isolation of the reactor vessel

The isolation of the reactor vessel was carried out in cell B1 and B3, containing the South and North primary loops and the vessel argon system, respectively. Work began in the largest cell (B2) to allow the workers to become familiar with the operations under good conditions. Air masks and sodium protective clothing were worn for all cutting and welding operations on the primary loops.

Each primary loop includes :

- a sodium inlet pipe (200 mm ID) and its protective sheath (255/259 dia)
- a clad failure detection line (36 dia) and its protective sheath (70 dia)

The same procedure was carried out on the South loops and on the symmetrical North loops. First, the main sodium pipe suspensions were immobilized, then plasma torches were used to cut the sodium inlet and outlet pipes and their sheaths, as well as the loop vessel. Annular flanges were welded to seal off the sheaths.

During the cutting operations, balloons were inserted through 60 mm dia holes and inflated on either side of the work zone to maintain leaktight containment.

The inner surfaces of the primary pipes were clean. A very thin sodium film adhering to the inner walls was removed using cotton swabs moistened with alcohol.

In cell B6 containing the sodium purification rack, the purified sodium return line and its protective sheath were cut in two places. Sealing was restored by welding two plugs onto the sodium pipe and an annular flange to the vessel side of the sheath.

1.2 System modifications

The reactor vessel sodium inlet and outlet pipes in the South loop were interconnected by a pipe fitted with two valves, and connected to the decontamination loop, the off-gas discharge system and the nitrogen scavenging system via additional valves. Similar connections were also provided for the North loop inlet and outlet pipes.

The REna 300 drain line was connected across a valve to the inlet orifice of pump PPal 463 in order to allow for agitation and removal of the decontaminating fluids outside the reactor building.

Five 4π spray nozzles with shutoff valves were intalled at the top of the five sodium pumps and exchangers reservoirs and connected to the discharge line from pump PPal 481.

1.3 Integrated dose

All these operations lasted four months, and were satisfactorily completed despite the cramped space and difficult working conditions.

▪ The integrated dose incurred during removal of the biological shielding and thermal insulating material in cells B1 and B2 was 10.4 mSv for 5 workers.

▪ Separation of the reactor vessel and modification of the pipe runs resulted in an integrated dose of 31.6 mSv for 7 workers.

2. Additional equipment installation (B.5)

An additional equipment was installed for decontamination purposes :

- a feed station for chemical reagents
- a demineralized water feed system
- nitrogen supply and off-gas removal systems
- an on-line measurement cell
- a station for collecting liquid wastes prior to transfer to the Cadarache waste treatment facility
- a local control station where data from the work site was centralized and decontamination operations were controlled and monitored.

A filling line with valves was provided for reagents supply from delivery trucks at ground level.

The process equipment is now being manufactured.

3. Operation safety considerations (B.4)

This paragraph is a brief excerpt from the operational safety analysis. Conventional risk assessments (chemical hazards, fire hazard, flooding, etc.) have intentionnally been omitted from the following discussion.

The system will be first washed with a caustic soda solution. The decontamination process then uses a hot (approx 60°C) aqueous solution of sulfuric acid and nitric acid which slowly etches away the metal surface of the primary cooling system. The surface layer of the pipes and reservoirs is thus dissolved to a depth of a few microns (i.e the thickness containing most of the residual activity). Oxidizing agents may be injected to enhance the action of the acid solution.

3.1 Irradiation hazards

The primary system components located in cells B1 and B2 (pipes and reservoirs) are radioactive, with dose rates ranging from 0.1 to 6 $mGy.h^{1}$ measured in March 1988. The hot spots are well known, however, and standard precautions are taken (limited exposure time in the cells, minimum safe distance from hot spots, authorization of the Radiation Protection Department, etc.).

3.2 Liquid contamination hazards

Origin : fluid leakage in the decontamination loops is principally due to localized corrosion by the acid mixture. Steel is not corroded by sodium hydroxide at the planned process temperature (60°C). Vulnerable components include valve bellows as well as pipes common to several decontamination loops and thus subject to repeated exposure.

Preventive measures : the decontamination process was developed during testing at the Grolabo facility on components taken from Rapsodie as discussed in the preceding annual report. Process implementation for up to 100 hours did not produce any metal deterioration liable to result in leakage.

All the system pipes are welded together, as are the pipe junctions with valves or pumps. All the welded joints necessary for the modification work prior to decontamination were 100% X-ray inspected to ensure a quality level equivalent to the weld seams in the remainder of the primary system.

Moreover, the leakage rate due to corrosion in a valve bellows will be limited to seepage by the presence of the packing seal.

Detection : the cell containing the circuits to be decontaminated will be provided with detectors to monitor the aerosol activity in the atmosphere, with provision for adjustable alarm thresholds.

The primary system is equipped with two types of leak detectors :

- spark plug type leak detectors at the bottom of the protective sheaths around large diameter pipes and reservoirs as well as on the valves between the bellows and packing seal ;
- detectection wires with insulating beads in contact with pipes that are not surrounded by protective sheaths, especially near valve positions. In addition, in situ inspections will be performed by personnel equipped to handle chemical hazards (gloves, goggles, aprons) both inside the cells and around all pipes carrying process solutions.

3.3 Hydrogen inflammation

Hydrogen gas is produced by sulfuric acid in contact with stainless steel. Assuming an etching depth of 5 µm on the entire loop metal surface area over a 4-hour period (the optimum time based on the Grolabo test results) calculation shows that about 240 liters of H^2 would be produced, i.e less than 4% of the loop volume.

In air, hydrogen forms a flammable mixture at concentrations above 4%; the H^2-air mixture cannot ignite if the oxygen content is less than 5 vol%. The hydrogen explosion hazard can therefore be prevented by maintaining the oxygen content below 5% and/or the hydrogen content below 4%.

Detection and Preventive Measures

Inside the primary system, the oxygen content is well below 1%, and is continuously monitored by an oxygen analyzer AO 843 in the off-gas system, swept by a continuous nitrogen stream of at least 1 $m^3.h^{-1}$. The oxygen analyzer is accurate to within ±0,1%. Calibration tests at 0% and 21% using nitrogen and air streams are performed at the beginning of the decontamination procedure and at weekly intervals thereafter.

The off-gas stream released to atmosphere is monitored by an explosion meter AH 846 in the ventilation duct immediately after the HEPA filters to ensure that the hydrogen content is limited to 1% (the maximum value authorized by the safety engineer). The measurement accuracy is ±0,3%. the detection system is calibrated with a standard gas mixture at the beginning of the decontamination operations. If the specified threshold is exceeded, the operating technician reduces the hydrogen content by reverting to manual control of the nitrogen inlet valve and the off-gas (and therefore the hydrogen) flow rate, but this results in slight overpressure in the primary system.

Hydrogen leakage could result in accumulation inside the containment cell. Dispersion measurements of a simulated hydrogen leak inside the containment showed that the ventilation system quickly homogenizes the atmosphere, preventing any dangerous concentration at ceiling level.

8.12 Automated Measuring System for Waste from Dismantling of the KKN Plant, to be Released

Contractor: Nuklear-Ingenieur-Service GmbH, Hanau, Germany
Contract N°: FI1D-0062
Working Period: May 1986 - June 1989
Project Leader: I. Auler

A. Objectives and Scope

An important task in the decommissioning of nuclear installations is the proof of very low radioactivity levels, allowing for free release of the generated waste. This proof involves long measuring times on a great number of representative samples out of important masses of metal structures and concrete, and considerable radiation exposure of the measuring staff.

The main objective of the present research is the development, construction and large-scale testing of a prototype for an automatic measuring system, appropriate to treat important masses of waste, with low-level activities and different nuclide compositions and shapes. It is expected to minimise human errors by automatic operation.

The measuring system will be designed as a mobile unit, with a modular structure allowing for a general purpose application to LWR typical waste arisings, at different decommissioning sites. The practical testing will be done with a total mass of 1000 Mg in the framework of the KKN decommissioning.

The study will be completed by a conclusive assessment of the merits of the developed measuring system for large-scale operation.

B. Work Programme

B.1. Conceptual studies for the definition of the requirements for a measuring system, including assessment of existing low-level activity measuring techniques, definition of the types of waste to be treated, and health physics protection considerations.

B.2. Preparation of a design of the complete measuring system, including detectors, control and transport system, general purpose software for measuring data processing, followed by a call for tenders and the choice of manufacturers.

B.3. Preparation of a licensing dossier for experimental operation of the measuring system in the framework of the decommissioning of KKN.

B.4. Execution of a large-scale test programme.

B.5. Conclusive assessment of the appropriateness of the developed measuring system, considering technical and economic aspects.

C. Progress of Work and Obtained Results

Summary

Mounting of the measuring device was started at Kernkraftwerk Niederaichbach (KKN) in April 1988. First test measurements with the device were performed in June 1988. In October 1988 the features of the whole measuring device and its calibration were demonstrated to the inspection agency. After the final positive statement of the licensing authority for the operation of the device during KKN dismantling the routine measurements were started successfully with isolation cover steel sheets and isolation wool. The measurements will be continued.

Progress and Results

1. Preparation of Design (B.2.)

1.1 Analysis of Components to be measured

A detailed analysis of the surface-mass-ratio with components to be measured showed that the surface of e.g. tubes, plates or structural steel can be estimated easily and with sufficient accuracy by weighing. The premises are, that these components are sorted by kind of component, material and group of wall thickness. In total, sorting into about 20 groups of components will be sufficient to estimate the surface and to follow the activity of all parts to be measured.

1.2 Simulation of Source Geometry

As a basis for preparing software for the analysis of measurement results, the measuring effect of different idealized source geometries were calculated as a function of their coordinates inside the measuring chamber. The geometries were simulated for the calculation three-dimensionally, both, for the detectors and the different sources. The results of the calculations have shown, that the effect of the position of source inside the measuring chamber is relatively small compared to the total value from all detectors.

1.3 Mounting the Device at KKN

The measuring chamber without detectors was mounted at KKN in April 1988. After mounting the chamber, the conveyor track, weighing unit and controlling panel were installed and put into operation. The measuring chamber was assembled with twelve detectors and connected with the data acquisition and control unit in June 1988 (Figure 1). In the meantime, the software for the data exchange among control unit and data processing unit and the software for the user surface were prepared and tested. With the delivery of the data processing unit and its periphericals end of June the whole device was ready for test operation (Figure 2).

2. Licensing Procedures (B.3.)

The measuring device had to be approved by the licensing authority for its application during KKN dismantling. For this procedure, a document for preliminary examination was prepared and submitted to the licensing authority. After preliminary examination of the document, the licensing authority made a positive statement in January 1988.
Mid of October 1988 the features of the whole measuring device and the calibration were demonstrated to the inspection agency.
The licensing authority gave its final positive statement for the operation of the device at KKN in November 1988. Immediately after green light of the authority the measuring device was used for routine operation at KKN.

3. Execution of Test Programm (B.4.)

3.1 Test Operations

The aim of the test operation was to check the specified objectives of the device, to get detailed information about the different effects of the device such as stability and reproducibility of measuring and sensitivity of detection as function of

- counting rate
- source position
- source geometry
- gamma radiation energy
- shielding
- statistics.

The measurements confirmed the excellent features of the device. The minimum detectable activity is considerably lower than specified. The stability of measuring is better than one percent per day. The effect of variation of the source position at the practically used volume amounts to +/- 25 %. The difference of counting for a 100 dm^2 source compared to a 1 dm^2 source amounts to - 25 %. As a function of gamma energy the measuring efficiency decreases compared to the energy of Cobalt 60 (about 1250 keV) to 90 % for Cesium 137 (662 keV) and to 65 % for Barium 133 (about 350 keV). The reduction of counting rate due to the shielding of a Cobalt 60 source surrounded by 2 cm steel amounts to 34 %.

3.2 Routine Operations

The routine measurements were started with isolation steel cover sheets and isolation wool. The isolation sheets were measured as bunches with mass amoung 100 and 300 kg and with a basis of 100 cm length and 90 cm width. Up to the end of 1988 among other parts, 13 Mg of isolation sheets had been measured for unrestricted release. Less than 2 % of this isolation sheets had an activity slightly above the limit value of 3300 Bq per 10 kg of material. The specific activity averaged over the actual mass of a bunch of isolation sheets was lower than about one thirtieth of the specific limit value (0.33 Bq/g). The consumption of time for manual measurements of isolation sheets would be hundred times higher as with the automated measuring device. Due to the long manual measuring time these isolation sheets would have been disposed as radioactive waste in total if the automated measuring device had not existed.

The measurements with the device will be continued at KKN with other types of dismounted parts such as cables, tiny parts of steel, concrete rubble and ventilation ducts.

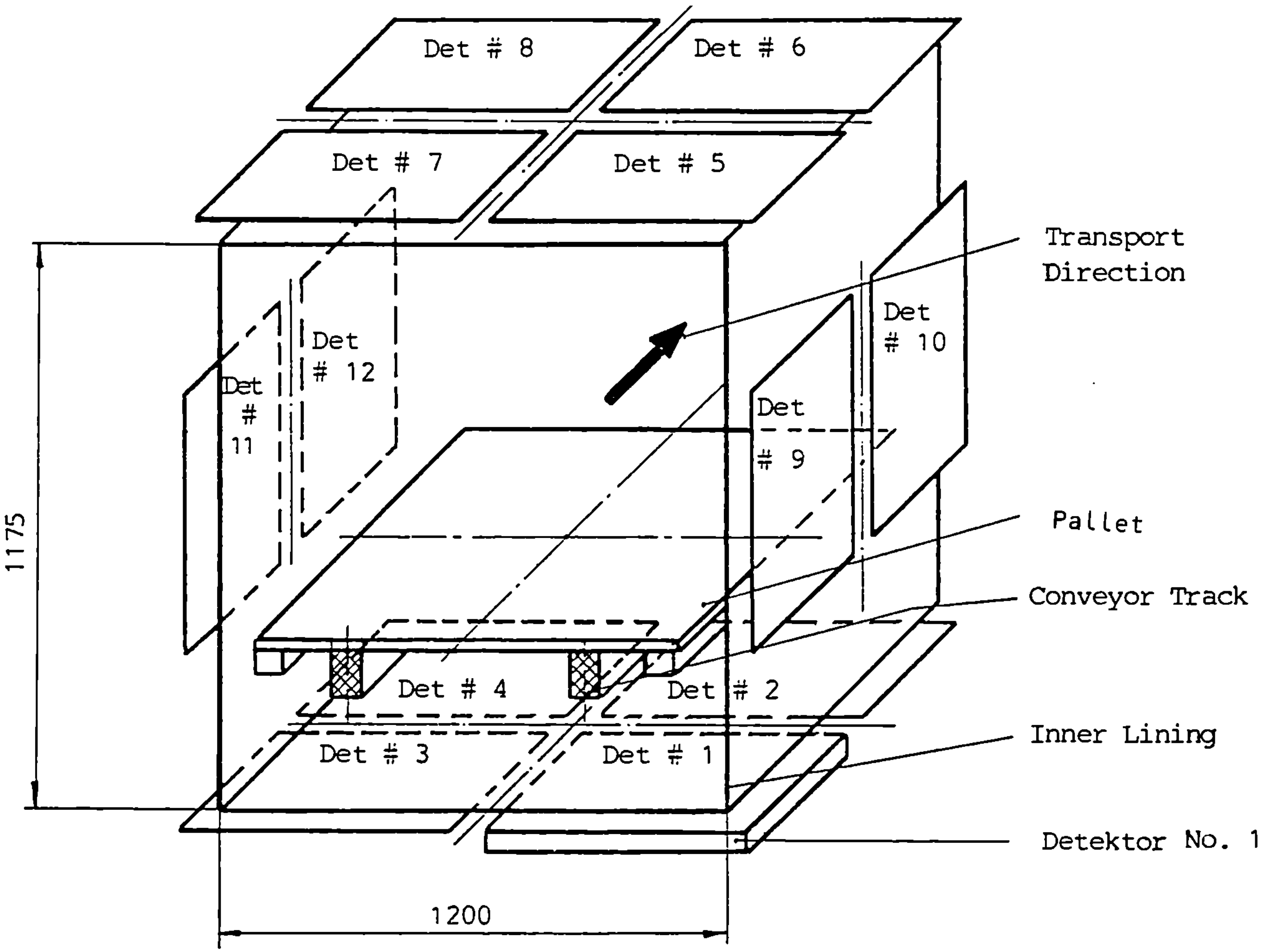


Figure 1 : Position of Detectors

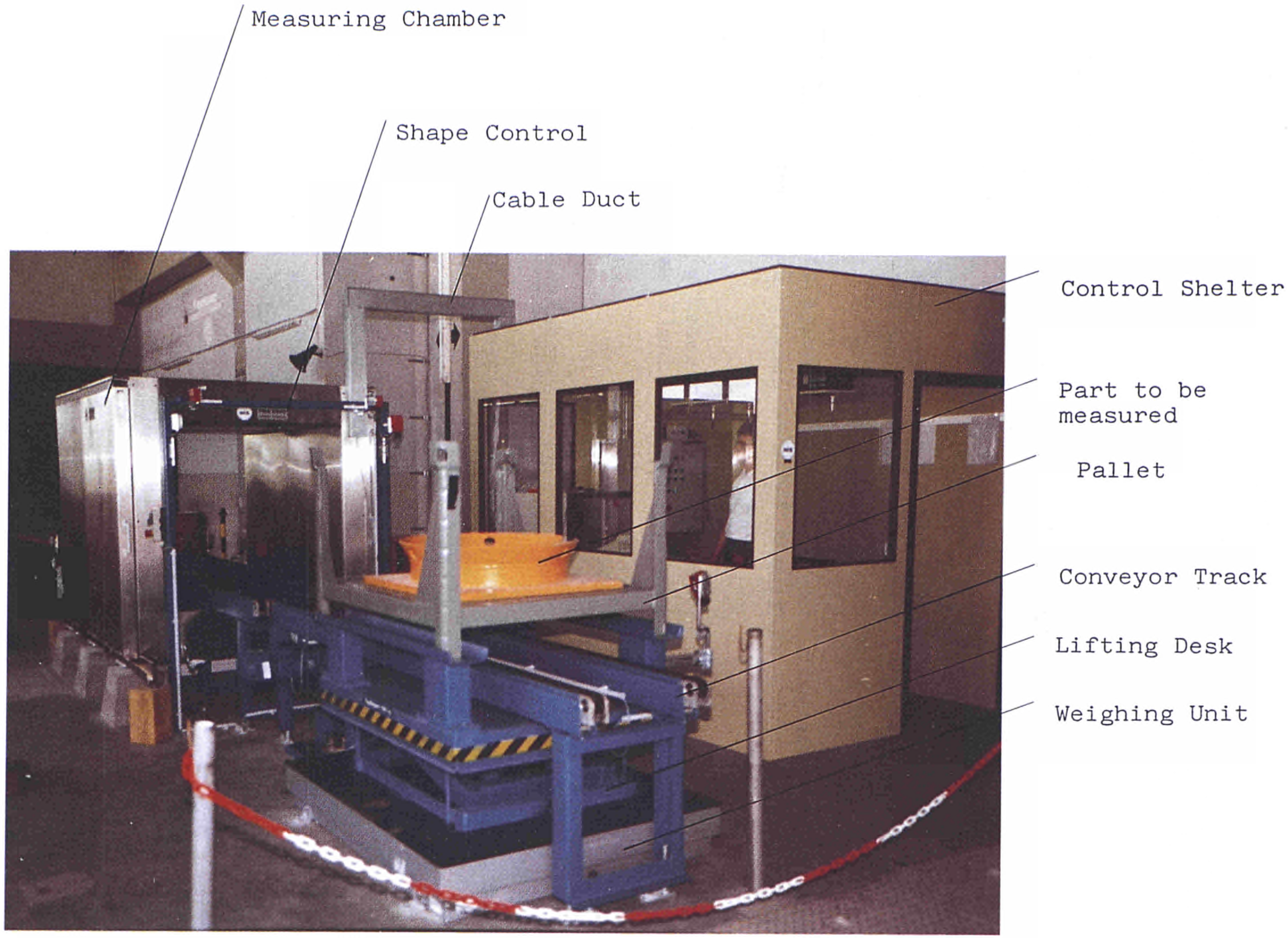


Figure 2 : Arrangement of Measuring Device

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.

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ANNEX II

LIST OF PUBLICATIONS RELATING TO THE RESULTS OF THE 1984-88 PROGRAMME ON THE DECOMMISSIONING OF NUCLEAR INSTALLATIONS

A. Annual Progress Reports

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - First Annual Progress Report (year 1985)", EUR 10740, 1986.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Second Annual Progress Report (year 1986)", EUR 11112, 1987.

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B. Final Contract Reports

Janberg, K. (1987). Economic comparison of management modes for contaminated metal scrap. EUR 11149.

Allibert, M. et al. (1987). Séparation par transport en phase vapeur des constituants d'aciers inoxydables. EUR 11296.

Ahlfänger, W. (1988). Vollständige Dekontamination einer Primärdampfleitung des Kernkraftwerks Lingen. EUR 11435.

Ebeling, W. et al. (1989). Untersuchung und Optimierung von Filtersystemen zur Abscheidung von Stäuben und Aerosolen bei der Dekontamination von Betonoberflächen. EUR 11995.

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(1) This Committee was established by the Council Decision of 29 June 1984 dealing with structures and procedures for the management and coordination of Community research, development and demonstration activities (OJ N° L 177, 4. 7. 1984, p. 25).
(2) Till June 1988.
(3) From July 1988.

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