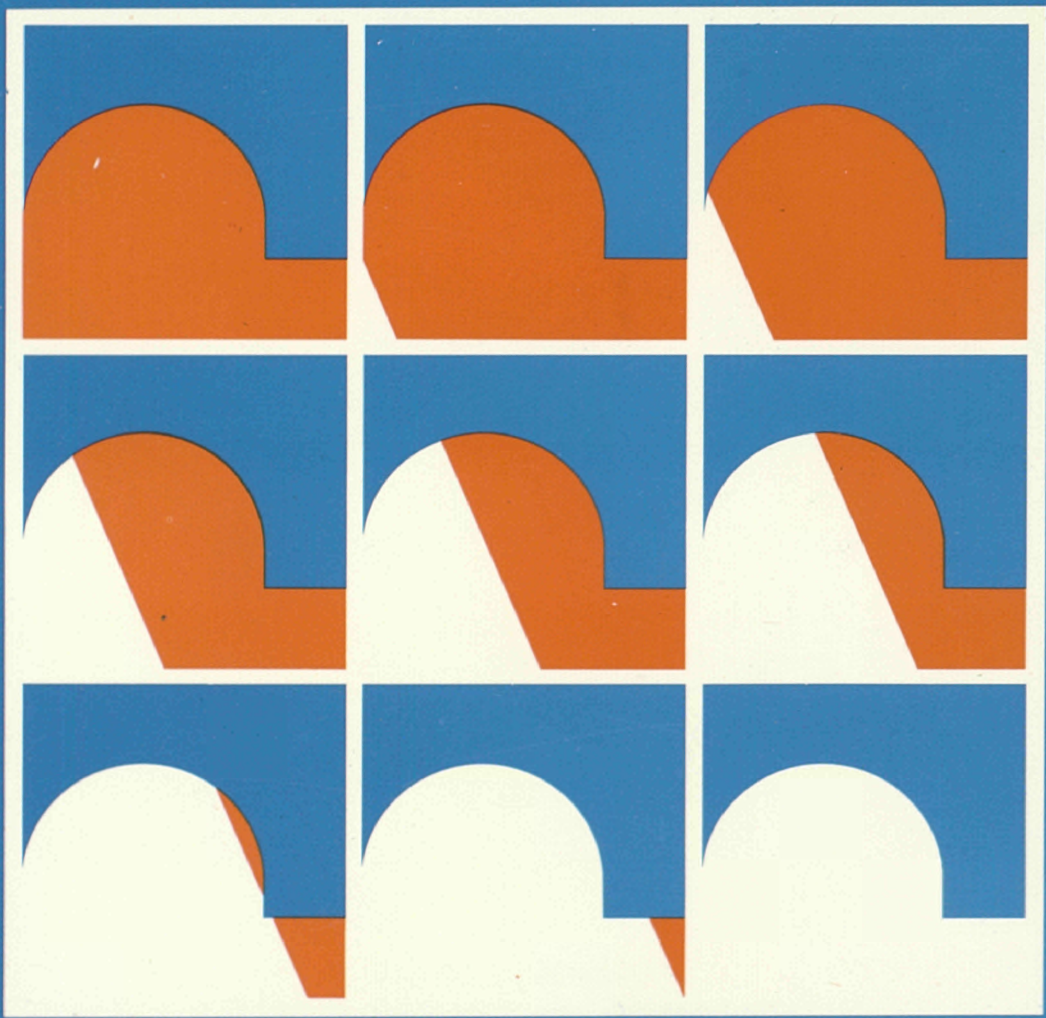
Commission of the European Communities

nuclear science and technology

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

Third annual progress report 1987



Report

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Third annual progress report 1987

Directorate-General
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FOREWORD

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

The Council of the European Communities adopted the programme in January 1984 /3/, 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 69 research contracts, including 11 new contracts concluded in 1987 as well as 10 contracts of which the execution has been completed in 1986 and 1987. Moreover, 3 contracts were still at the stage of negotiation at the end of the year.

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 1987.

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/ 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.

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

A. Objective

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

B. Research performed under the 1979-83 programme

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

C. 1984-88 programme

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

D. Programme implementation

Three research contracts relating to Project N°1 have been executed in 1987 including two new contracts concluded in 1987.

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: I. Ll. Davies

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

Suitable sites for survey work are expected to be agreed with Central Electricity Generating Board (CEGB) shortly, and it is also now likely that the UK Atomic Energy Authority (UKAEA) site at Windscale will be available for survey work. A preliminary survey of the latter is due shortly, with the main survey commencing as soon as possible thereafter. Detailed plans of in-situ testing, samples to be taken, and sample testing have been prepared, subject to finalisation after preliminary surveys. Development of the predictive model continues, in anticipation of results from surveys and sample analysis.

Progress and Results

1. Selection of Sites and Concrete Types (B1)

The research programme is based on surveys of existing nuclear power plants in the UK to provide quantitative data on the state and rate of degradation of the associated buildings. Constraints within the programme dictate that it is not practicable to survey all existing nuclear power stations in the UK, and therefore, to meet the objectives of this programme, stations surveyed should represent a wide range of environmental conditions for the materials sampled and tested. To improve the calibration process for the deterioration prediction package, the availability of previous survey data is also being considered in station selection, for example work carried out by the CEGB under the framework of the CEC's Phase 1 research programme on Decommissioning of Nuclear Power Plants /1/.

Discussions are currently in progress with the CEGB to identify suitable sites for the survey work and to agree on the programme of work for each site with the aim of commencing the surveys in February 1988. It has also emerged that the UKAEA site at Windscale should be available for survey work. A preliminary survey for familiarisation and planning of survey areas and sampling locations is to be undertaken 28, 29 January 88; dates for the main survey will also be established on that occasion.

2. Literature Survey and Deterioration Prediction (B2, B6)

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 initiated, 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 is currently concentrating 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

- definition of the threshold levels

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

3. In-Situ Testing, Laboratory Tests on Samples, and Results Analysis (B3, B4, B5)

To predict the future deterioration of the buildings concerned, the processes to be modelled require data to be collected in surveys; in the following sections, the work planned is outlined. Survey areas will be defined based on initial site visits, layouts of structures, and reviews of any previous similar work. It is planned that all the tests should be carried out both on bioshield/pcpv concrete structures and on non-critical structures of similar concretes, with the exception of core sampling which would only be carried out on 'similar' concretes.

3.1 Environment

Environmental variables such as temperature, relative humidity, CO_2 content of air, chloride content of air, rainfall, all affect the possible deterioration processes and should therefore be measured, (except where suitable information is available from records). To compare the deterioration and concrete performance of the bioshields/pcpv's with those of the reactor hall concrete or other concrete structures, it will be necessary to monitor the above environmental factors both inside and outside the reactor hall, so that the effect of different environments can be included in the overall assessment. Deterioration of the building fabric currently protecting the bioshield/pcpv concrete could worsen the internal environment, thereby increasing the risk of concrete deterioration. Therefore a visual inspection of typical components will be carried out with further test work if necessary, to identify the potential 'weak-link' components requiring routine checks or maintenance.

3.2 Corrosion Activation

The main variables to be measured are as follows:

3.2.1 Depth of carbonation: measured by drilling or otherwise forming a shallow hole in the concrete surface and spraying a fresh-fractured surface with a chemical indicator solution. The variability of carbonation depth will be measured by making measurements at as many points as possible - at least 25 points being desirable (and preferably more), at randomly chosen locations, whilst taking account of any major environmental variations noted. Wherever possible carbonation depth measurements will be at holes made for other purposes (eg chloride drillings, reinforcement connections, core holes, etc).

3.2.2 Chloride content profile: Measured by incrementally collecting dust drillings, for subsequent chemical analysis. Since chloride ingress is unlikely to be significant so far in concrete not exposed to sea-spray, in general only a small number of locations would be checked (say 5 or 6). For marine exposure, however, more results are necessary, to measure the variability (as for carbonation). Measurement of chloride diffusion coefficients on cores from similar concretes is also necessary.

3.2.3 Depth of Cover to Reinforcement: Measured by covermeter with comparative checks from direct measurement of 'as-built' cover after exposing short lengths of reinforcement. The distribution of cover should preferably be measured for each reinforcement bar type/location, again by taking a large number of readings randomly located. (Non-destructive apart from comparison points). Reinforcement layouts and sizes will be pre-checked from drawings if available.

3.2.4 Reinforcement Potentials: Survey by reinforcement potential measurement (potential wheel) to estimate present extent of corrosion activation. Potentials assessment criteria will be checked by exposing reinforcement, in areas of high, low and typical potential. Electrical connection to reinforcement is required, together with checks of electrical continuity over large areas or between different elements. (Non-destructive apart from assessment criteria checks & electrical connections).

3.3 Corrosion Propagation

There are two main aspects of corrosion propagation which together imply the time from corrosion activation to damage occurrence:

- corrosion rate, R
- amount of corrosion, M, the formation of which on a steel bar causes a predefined level of damage to occur.

Therefore the time from activation to damage = M/R

The corrosion rate, R will be estimated from a combination of survey work (linear polarisation, concrete resistivity, initial surface absorption tests), and laboratory tests on samples (water sorptivity, oxygen diffusivity, moisture content).

The amount of corrosion on a bar, M, which results in damage is estimated from modelling and previous research work on the tolerable stresses, together with information collected during surveys (reinforcement diameter, cover, spacing; current extent of corrosion on reinforcement).

<u>References</u>

1. LEWIS, G H Degradation of building materials over a lifespan 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 1987, the following aims of the research programme were developed:

- special structural elements were selected in accordance with their relevance and various test methods were discussed, in order to analyse the long-term behaviour (B.1.3.);
- an extensive study concerning the corrosion mechanism was performed (B.2.2.),
- a test programme on the long-term behaviour of plastic seals was drawn up (B.2.1.),
- ultrasonic tests were carried out in order to detect corrosion in inaccessible areas (B.2.3.).

Progress and Results

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

Structural elements were selected concerning their relevance for the functioning and long-term stability of the reactor building. A further criterion for the selection was the range of utilisation of materials already under dead load. For application on these main structural elements, various test methods like monitoring procedures, especially non-destructive ones, were discussed. These test methods are for instance:

- visual checks for gaining a rough survey over the actual state of all accessible areas,
- taking core samples for getting accurate measurements of the main accessible areas,
- ultrasonic pulse velocity (UPV) tests as one of the important non-destructive test methods which are still in development,
- and other non-destructive tests for checking the concrete cover, carbonation depth and chloride concentration.

Hereby the properties of concrete like strength, cracks, voids, alkalinity, concrete cover, etc., are of main interest. Concerning the reinforcement, the conditions for corrosion have to be established by test methods. No severe effects on the stability of the structural elements are expected if the durability of anchorage elements or bearings will be less than the entire period preceding dismantling.

Generally, it is recommended for all inner structural elements and especially those which are located in inaccessible areas to maintain corrosion-delaying climatic conditions.

Regarding the monitoring procedures, the stability of the reference plants can be expected to continue at least 100 years after decommissioning.

2. Classification and assessment of damage caused by corrosion (B.2.2.)

The aim of finding relevant works by means of a thorough search of the literature on the subjects "Classification and assessment of the damage causes by corrosion" has not yet been fully achieved. In the course of search of literature at various computer data banks, some 250 works have been evaluated so far. However, none of these works were of direct relevance, i.e. treating the corrosion behaviour of the containment or the steel liner. Therefore, reference can be made only to theoretical corrosion models that describe the behaviour of the materials used under certain conditions (e.g. corrosion medium).

The work found during searches of the literature on the behaviour of concrete steels or of reinforced concrete can be applied only indirectly to the long-term corrosion behaviour of the containment and must be adapted to take account of the specific circumstances.

The behaviour of the materials used was examined under the given corrosive conditions by means of theoretical corrosion models. As far as the surface corrosion of unalloyed steels and low-alloy steels in weakly acid to weakly alkaline media is concerned, the available data in the literature indicate rates of about 0.09 to 0.3 mm/a. Data on the corrosion rate with ground craters and holes are not available for the corrosion/chemical conditions in the containment. This is because the data frequently depend on indefinable local circumstances, such as heterogeneity of the surface, porousness of rustproof coatings, growing concentration of chlorides, etc., and thus, cannot be recorded by their very nature. Under the prevailing circumstances, the conditions for stress corrosion cracking do not exist. During the further investigations, proposals will be made for prevention of corrosion damage.

3. Long-term behaviour of plastic seals (B.2.1.)

After establishing the concept, the requirements of plastic seals were determined; they result from the operating conditions of the PWR plants presently being constructed. In addition, a test programme was drawn up with reference to ageing and time extrapolation.

The literature indicates that, as a matter of principle, the ageing of silicones can be determined with the Arrhenius- and Williams-Landl-Ferry-equations; this is to be proven by the study. For that reason, 200 test pieces of silicone material were made in accordance with the processing guidelines and the instructions for producing test pieces at one of the convoy sites. These tests were expected to take 1.5 years.

These test pieces were installed in special devices with 25% compression strain and stored in heating furnaces at 40, 60, 80 and 120 degrees. After 24, 72, 168, 336 and 672 hours, samples were taken and the residual compression strain values determined.

The curves show a steady increase of the residual compression strain and seem to have reached a maximum after about 14 days, since the values decrease again slightly after this period.

A first evaluation with an attempt to extrapolate the results will be made after a storage period of about six months (mid-1988). Adhesive strength tests will also be carried out on the samples available by then and the results evaluated.

In the meantime, studies of the literature, i.e. specialised literature on the ageing of silicone rubbers in particular are continuing.

4. Non-destructive tests (B.2.3.)

Non-destructive tests, using ultrasonic methods, are to prove corrosion damage in inaccessible areas between reactor containment and reinforced concrete.

Tests on a test piece with the ground craters and tests on the reactor containment of a nuclear power plant with corrosion damage aimed at determining suitable test heads for detecting these types of defect. As a result of the coating or the surface roughness of the reactor containment, the best results were obtained with a 45° angle test head.

The test results showed that the sensitivity difference of the echo height from the deflection areas and the individual indications of the corrosion scars is max. 6 dB. However, this is too little to prove corrosion with certainty.

Further tests and improved test methods aim at clarifying whether this signal-to-noise ratio can be improved.

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 inventory of the relevant plant characteristics and of the key issues to be considered, as well as the radioactive inventory of the safe enclosure (SE) were carried out. Up to mid-1987 the required instrumentation was installed and since July, the measured values were recorded. The status of the plant is as follows:

- no fissile material is in the plant;
- all systems are drained and all drain-valves are open;
- the stored waste is mostly non-conditioned, so that the resins and ion-exchange powder are only drained and still in their storage tanks.

It is important to know that a small ventilation system is under operation. The system circulates a volume of 3,500 m^3 of air per hour, which is dried and a volume of 600 m^3 per hour, which is filtered and released through the stack.

Progress and Results

1. Radioactivity inventory (B.1.)

The activity inventory was performed for the reactor and auxiliary buildings:

	Inventory of the reactor building (Bq)	Inventory of the auxiliary building (Bq)
Activated material	8.2 E15	-
Contaminated surfaces inside systems	2.1 E13	1.4 E10
Contaminated building surfaces	1.7 E 9	1.7 E 9
Stored waste	1.4 E12	1.8 E14

2. Estimation of activity release (B.2.)

The airborne activity release will occur due to three effects:

- temperature variations inside the SE: this effect can be neglected;
- atmospheric pressure variations: this effect is very small, but it is calculated. The results are given in Table I;
- pressure differences caused by the wind: this gives the main effect and depends on the tightness of the plant. The leakage-rate as a function of pressure differences is evaluated and the strength of the wind as a function of time is measured. The leakage is calculated twice:
  - . under the condition of a running off-air system (with an off-air rate of 600 m^3/h and a negative pressure of 10 Pa);
  - . without the effect of negative pressure.

3. Implementation of the measurement programme (B.3.)

The measurement programme is fully under operation. However, due to the unusual weather conditions, some of the measured values are not representative. In addition, the plant itself reached the technical status of the SE in mid-1987 (shutdown of the original ventilation system), but there was still some water in the plant. The results for the second half year are as follows:

- The specific aerosol activity is in the order of 4×10^{-3} Bq/m^3 (95% Cs-137 and 5% Co-60). There are some differences between the different buildings. Probes were taken by filter sampling.
- The tritium activity measured in the condensate accumulated in the air-drying system is in the order of 1 MBq/m^3. The tritium activity in the air with a water content of 6 g/m^3 is about 6 Bq/m^3.

- The radon activity, measured by the Karlsruhe Nuclear Research Centre, varies between 50 Bq/m^3 in the auxiliary building and 200 Bq/m^3 in the containment.
- The temperature and the relative humidity of the air are measured in all three buildings (containment, angulus with connecting building and auxiliary building) in each case at the bottom, in the middle and at the top;
  . the temperature variation is shown in Figure 1. (The figures are not typical due to the unusual weather during this winter);
  . the relative humidity in the containment is given in Figure 2.
- The wind force is printed out continuously as a percentage of time for some ranges of velocity. An example is given in Figure 3.
- The leakage of the SE was measured by variation of the off-air rate. The results are pointed out in Figure 4. The leakage is a non-linear function of the pressure difference against the outside atmosphere.

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

It is too early for a final discussion of the results. Table II shows the calculated monthly leakages for the SE split into the leakage due to wind and atmospheric pressure drops. The calculated leakage and the measured tightness are the data for the complete area of the SE, i.e. the auxiliary building, the connecting building and the angulus. The leakage of the containment alone ought to be small against this.

One can notice that there is a factor of two between the leakage with and without a ventilation system.

The activity output due to the leakage is very small. Without a ventilation system, the aerosol output is higher by a factor of two, but the tritium output is lower by a factor of two than with a ventilation system. The reason for this is that the aerosols in the off-air system are reduced by means of a HEPA filter, i.e. by a factor of 1,000 or more; the tritium can only be reduced by drying up the air, i.e. by a factor of less than two. The results are shown in Tables III and IV.

Table I: Monthly leakages due to variations of atmospheric pressure

Month	Sum of pressure drops (Pa)	Expansion of air volume (%)	Air expansion of containment (m^3)	Air expansion of total SE (m^3)
July	75	7.5	5,042	5,237
August	67	6.7	4,504	4,678
September	73	7.3	4,908	5,097
October	55	5.5	3,698	3,840
November	93	9.3	6,252	6,493
December	79	7.9	5,311	5,516
Total:			29,716	30,861

Table II: Monthly leakages for the total area of the SE due to wind

Month	Leakage rate with ventilation (m^3)	Total air exchange with ventilation (m^3)	Leakage rate without ventilat. (m^3)
July	81,980	513,980	220,266
August	64,646	496,646	202,858
September	57,900	489,900	192,750
October	374,766	806,766	464,600
November	34,849	466,849	149,148
December	102,085	534,085	239,113
Total:	716,226	3308,226	1468,735

Table III: Monthly aerosols activity output of SE total area (Bq)

Month	Without ventilation system	With ventilation system		
		by leakages	via stack	total
July	1,000	390	2	392
August	920	310	2	312
September	880	280	2	282
October	2,100	1,700	2	1,702
November	690	180	2	182
December	1,100	480	2	482
Total:	6,690	3,340	12	3,352

Table IV: Monthly tritium activity output of SE total area (MBq)

Month	Without ventilation system	With ventilation system		
		by leakages	via stack	total
July	1.8	0.7	3.5	4.2
August	1.7	0.5	3.5	4.0
September	1.6	0.5	3.5	4.0
October	3.7	3.0	3.5	6.5
November	1.2	0.3	3.5	3.8
December	2.0	0.8	3.5	4.3
Total:	12.0	5.8	21.0	26.8

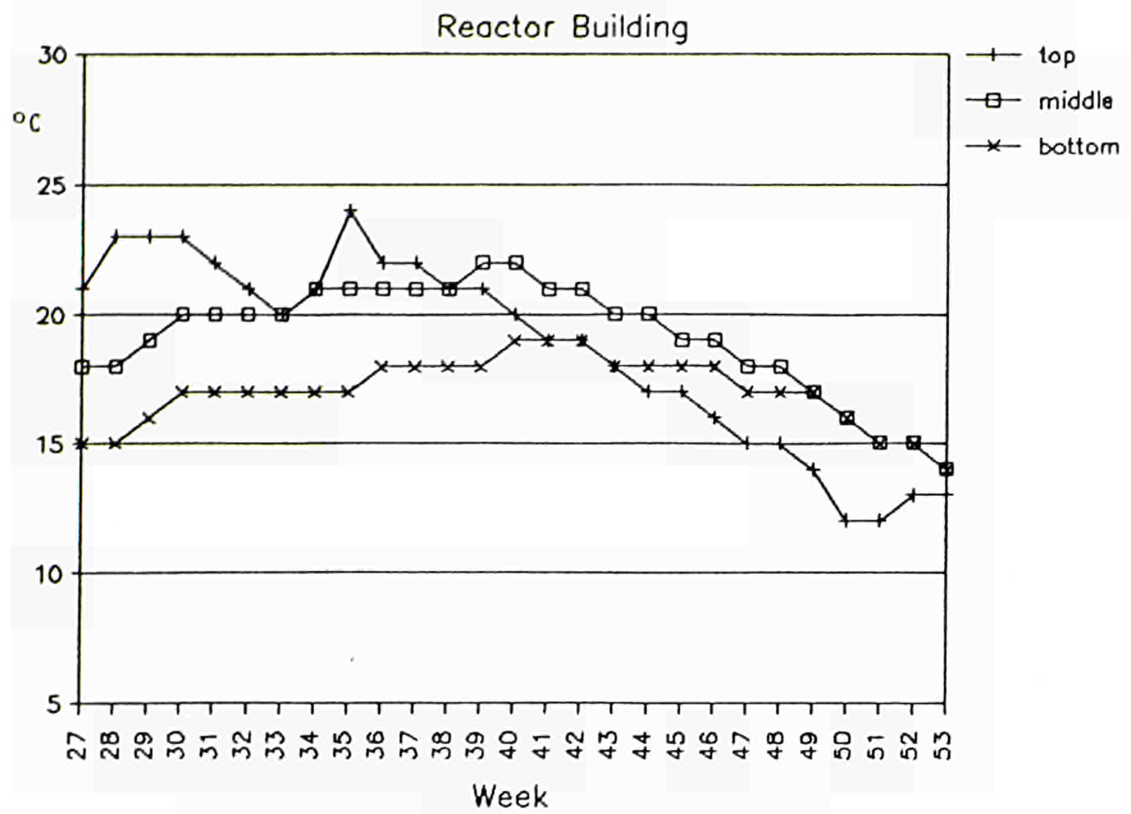


Figure 1 : Evolution of the temperature

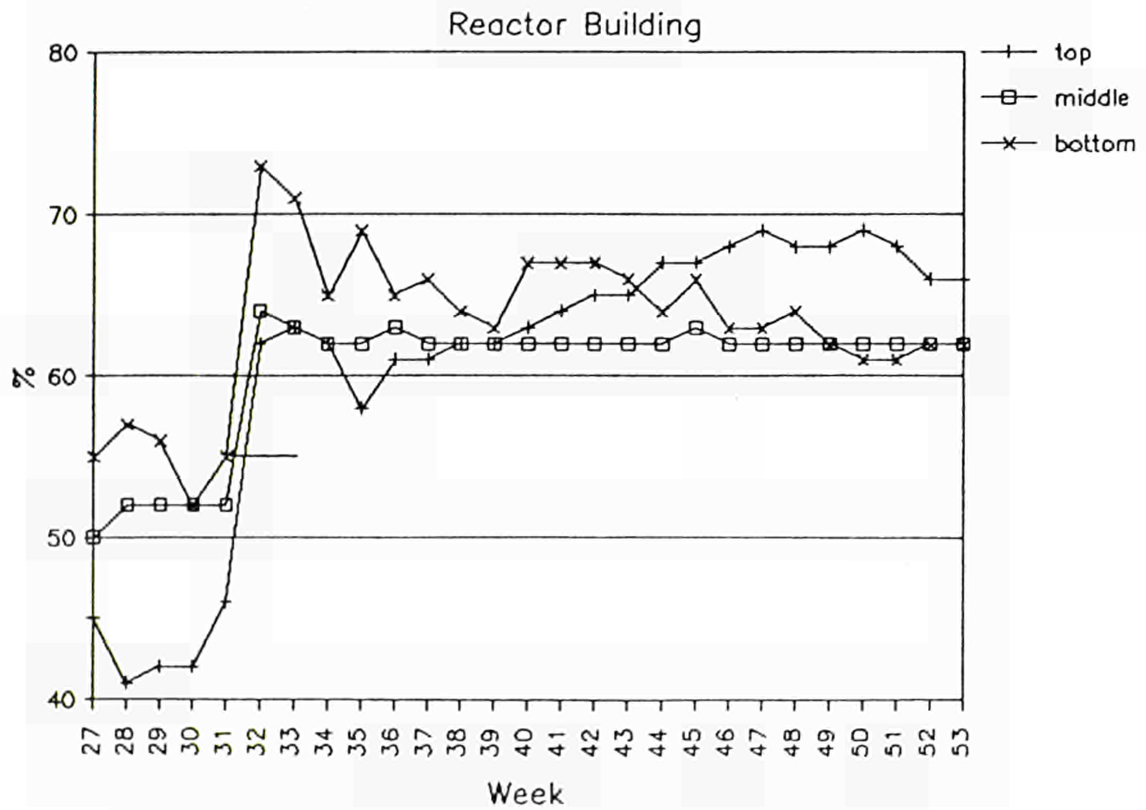


Figure 2 : Evolution of the relative humidity

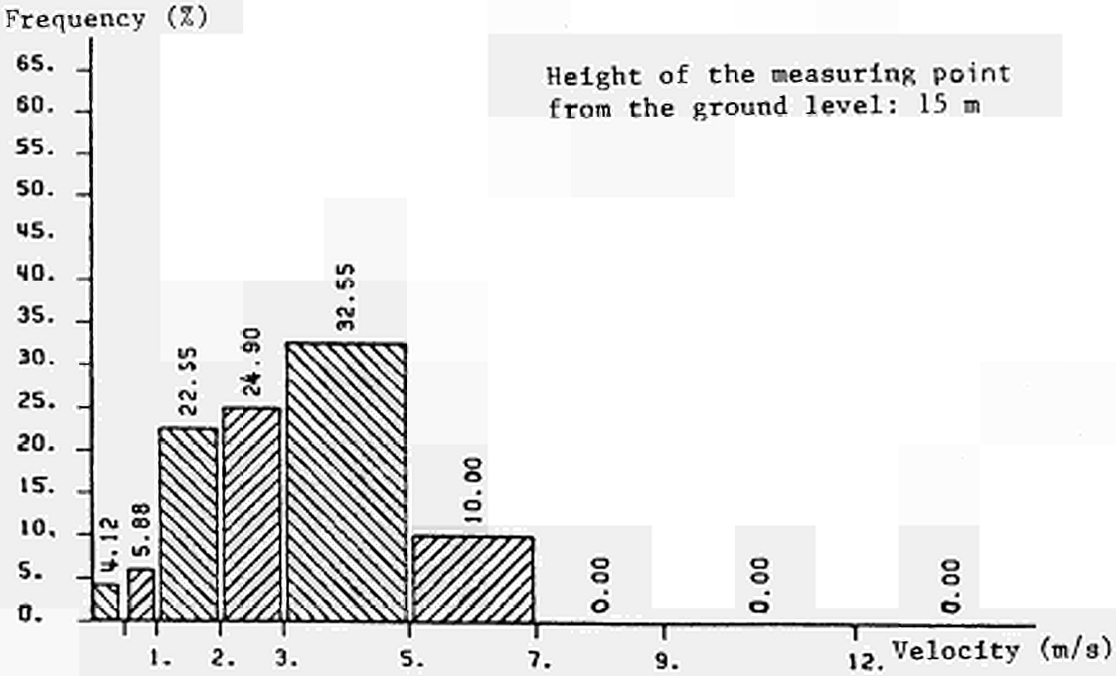


Figure 3: Histogram of the wind velocity during December 1987

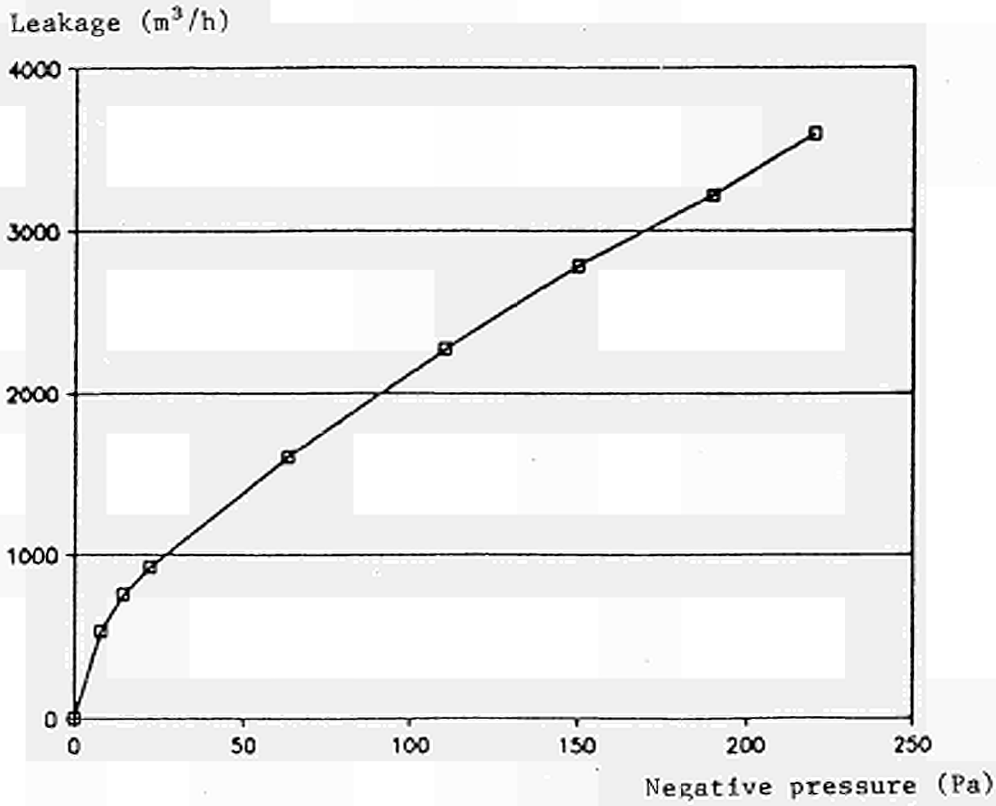


Figure 4: Leakage of the safe enclosure area

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

Twelve research contracts relating to Project N°2 were being executed in 1987, including two new contracts concluded in 1987 as well as five contracts whose execution has been completed in 1987.

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 under publication.

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 Obtained Results

Summary

As was shown in the experiments performed in the previous two years, good results can be obtained with chemical decontamination with aggressive acids, related to a final radiometric level up to unrestricted release limits, only in the case of simple and relatively small components.

As a consequence, in order to decontaminate all components as thoroughly as possible, the chemical actions need to be enhanced by another action such as mechanical or electrical. Ultrasound appears to be the best way of applying an effective mechanical action.

In this progress report a literature review of the cleaning action of ultrasound in aqueous solutions and a series of decontamination tests by ultrasound in aggressive chemical baths are described.

The literature review reveals the process parameters affecting the cleaning effectiveness while the decontamination tests show the enhancement action due to ultrasounds (B.4., B.5.).

Progress and Results

1. Literature review (B.4.)

Ultrasounds are widely used in the nuclear industry as a decontamination technique. Nevertheless ultrasounds are generally used by immersing the piece to be decontaminated in water or water with detergent solutions, because this decontamination technique has been finalized for reuse of the piece. There is no literature on the performance of ultrasonic decontamination with aggressive chemicals, as was investigated in this research.

Two main factors play a fundamental role in the action of ultrasounds: the cavitation threshold and the scrubbing factor. The cavitation threshold (Pct) is the difference in pressure inside the fluid which allows the cavitation phenomenon to take place. It is directly correlated with the specific ultrasonic power which is put in the solution.

The scrubbing factor, usually measured by weight loss (ΔP), is the direct result of the ultrasonic cleaning action. In contaminated surfaces it could be directly connected with the decontamination effectiveness (or DF).

It is obvious that the cavitation threshold must be minimized (in order to use less power) and the weight loss must be maximum (in order to have high decontamination effectiveness).

Cavitation threshold and weight loss depends on many factors connected with both ultrasonic and solution characteristics. The analysis of the influence of these factors on Pct and ΔP is very difficult; because of the lack of experiments and data only qualitative behaviour was revealed.

In particular the effect of the following parameters was evaluated: f, frequency of ultrasounds; p, pressure of the solution (or vapour tension); σ, surface tension of the solution; ν, viscosity of the solution; T, temperature of the solution; t, time of application of the ultrasounds.

As qualitative dependences for the cavitation threshold:

$$Pct \div W = F\ (f,\ P,\ \sigma\ ,\ \nu,\ T,\ \ldots)$$

one has: Pct increases when the ultrasound frequency, the pressure of the solution, the surface tension and the viscosity of the solution increase; while Pct decreases when the solution temperature increases. Concerning the ultrasonic effectiveness in terms of weight loss;

$$\Delta P = G\ (f,\ P,\ \sigma,\ \nu,\ T,\ t,\ \ldots)$$

the qualitative dependences are: ΔP increases when the pressure of solution, the surface tension of the solution, the temperature up to 60°C, and the time of application of ultrasounds increase; while ΔP decreases when the ultrasound frequency and the viscosity of the solution increase.

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

Two series of laboratory tests have been performed: the former on samples taken from the valves already decontaminated in previous tests and the latter on simple flat contaminated specimens from a large pipe of Garigliano BWR.

All tests have been carried out in the DECO laboratory using a common US device with a frequency of 21 kHz, a total volume of 2.5 l and an US power of 35 W. As aggressive chemicals the following solutions were tested: 1.5% HF + 5% HNO_3 and 4% HCl.

The tests were usually 1 h long with different steps. Of course reference tests both with US in water and in aggressive chemicals without US were performed.

2.1 Tests on Specimens from already Decontaminated Valves

These experiments were carried out on complex specimens in order to see the effectiveness of the process on complex surfaces with deep crevices still full with contamination.

All tests were performed at 60°C and some tests were performed without US; a plug, weld and real specimens were used in particular.

The results of the experiments allow us to draw the following conclusions:

- in all tests a DF greater than 1 was obtained: this means that by cutting the valve the separated parts can be decontaminated more than in the case of the assembled valve;
- all tests performed with ultrasound show DF greater than in tests without ultrasound: this means that the effect of ultrasound is always beneficial;
- the DFs increase with test time but, apart from the case of the plug specimens, it appears that the major decontamination effect is in the first 10 min step and then the decontamination effect tends to decrease;
- the behaviour of the DF on the plug specimens can be explained by the particular geometry of the specimens which have a very deep crevice which takes longer to clean;
- the fact that the DFs is greater than 1 in the tests without ultrasounds also (as in the chemical solution tested in DECO loop in the previous stage of the project) appears to be mainly explained by the difference in

test temperature (60°C vs 40°C);

- the DFs in the tests with ultrasound appear to be greater using the HF/HNO_3 solution rather than the HCl solution; one exception is for the tests on stem seal specimens. Up to now this is not clearly understood;
- remembering that without ultrasounds the DFs in the HF/HNO_3 solution were lower than the HCl solution, it can clearly be seen that the ultrasounds work better in the HF/HNO_3 solution.

2.2 Tests for investigating process parameters

To reveal the influence of some process parameters, such as test temperature, test time and chemicals, many tests were performed on simple flat specimens.

The test matrix is given in Table I; about 50 experiments were performed both with and without US at different temperatures.

The test results are very dispersed and difficult to analyze. The number of tests is quite consistent and the reproducibility of data are quite good, nevertheless it does not seems simple to separate the effects of chemicals and ultrasounds.

The reference tests carried out in water with ultrasounds do not show any appreciable difference and they do not indicate any temperature effect both in weight loss and in decon effectiveness.

The only possible treatment of data is in terms of the ratio between the joint effect of chemicals and ultrasounds and the effect of chemicals without ultrasounds.

For the solutions tested (1.5 HF/5 HNO_3 and 4 HCl), the previous functions are summarized in Table II. One can see that the $\Delta P_{US+CH}/\Delta P_{CH}$ ratio decreases with test temperature from greater than 1 to less than 1: in the 1.5 HF/5 HNO_3 solution it is around 1 between 60 and 75°C, while in the 4 HCl solution it is around 1 between 40 and 70°C.

The DF_{US+CH}/DF_{CH} ratio is usually greater than 1, but it is significantly high (around 10) only in the 1.5 HF/5 HNO_3 solution between 60 and 70°C.

The behaviour of the $\Delta P_{CH+US}/\Delta P_{CH}$ ratio which decreases when the temperature increases means that the enhancement action of ultrasounds is effective only at lower temperatures. At temperatures higher than 70°C the $\Delta P_{CH+US}/\Delta P_{CH}$ ratio is less than 1 and this indicates that the mechanical action of ultrasounds removes the chemicals from the surface to be decontaminated slowing the pure attack action of chemicals.

The fact that the behaviour of the DF_{CH+US}/DF_{CH} ratio is different from the behaviour of the $\Delta P_{CH+US}/\Delta P_{CH}$ ratio indicates that the effect of chemicals and ultrasounds varies on the oxide and on the clean base stainless steel. Looking at the data it appears that the enhancement effect of ultrasounds is more obvious on the oxide.

TABLE I . Decontamination process with US in aggressive chemicals.
. Laboratory testing on AISI 304 contaminated specimens taken from Garigliano BWR.
. List of the experiments for investigating test temperature and chemicals.

Decontaminating solution (in US tank)	Presence of US	Test temperature (°C)									
		25	40	45	50	55	60	65	70	75	80
1.5% HF + 5% HNO_3	NO	ST-1 ST-2	ST-3	ST-4	ST-5		ST-6	ST-7	ST-8	ST-9	ST-10 ST-11
	YES	US-12	US-13		US-14	US-15	US-16	US-17	US-18	US-19 US-20	US-21 US-22
4% HCl	NO	ST-23 ST-24	ST-25 ST-26		ST-27		ST-28		ST-29		ST-30 ST-31
	YES	US-32	US-33		US-34	US-35	US-36	US-37	US-38	US-39	US-40 US-41 US-42
Water	YES	US-43 US-44					US-45 US-46				US-47 US-48

TABLE II - Summary of the effect of ultrasounds in aggressive chemicals investigated in laboratory tests on AISI 304 contaminated specimens.

Effect	Ratio	Aggressive chemicals	
		1.5% vol HF + 5% vol HNO_3	4% vol HCl
Weight loss	$\frac{\Delta P_{US+CH}}{\Delta P_{CH}}$	>1, up to about 60°C; $\simeq 1$, between 60 and 75°C; <1, around 80°C;	>1, at 25°C; $\simeq 1$, between 40 and 70°C; <1, around 80°C;
Decontamination factor	$\frac{DF_{US+CH}}{DF_{CH}}$	$\gg 1$ ($\simeq 10$), between 60 and 70°C; >1, between 40 and 80°C; $\simeq 1$, below 40°C.	>1, around 60°C; $\simeq 1$, below and over 60°C;

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

Summary

During this year, two activities have been developed:

- an industrial application of gel spraying associated with chemical agents to decontaminate the ferritic steel of cooling circuit of gas-graphite reactor G2;
- the optimisation of decontamination by abrasive blasting, which will be applied to clean the ferritic steel surfaces of the Superphenix fuel storage drum intervessel gap.

Progress and Results

1. Decontamination by spraying of chemical gels (B.2. and B.4.)

This decontamination method was applicated to cooling circuit of gas-graphite reactor G2, before dismantling, with the following sequence: basic gel spraying, rinsing, acid gel spraying, rinsing. Larger scale and in-situ tests have been performed.

Larger scale tests

These tests were carried out in order to confirm the decontamination factors and to evaluate the waste volume which will be generated by the decontamination process. Following chemical gels are selected:

- a basic gel containing sodium hydroxide (3 mol/l) with or without sodium permanganate,
- an acid gel containing sulphuric acid (3 mol/l) and phosphoric acid (3 mol/l).

The acid gel spraying can be repeated once if necessary.
With this method, tests have been performed on plate samples (50x50 cm) from the "cold leg" (after the heat exchanger) and from the "hot leg" (before the heat exchanger) of the primary circuit. The main results of these tests are the following:

- mass of gel sprayed: about 200 g/m^2 for each step,
- volume of rinsing water: about 28 l/m^2,
- initial activity: 30 Bq/cm^2 (cold leg), 100 Bq/cm^2 (hot leg),
- residual activity: 0.4 Bq/cm^2 (cold leg), 0.5 Bq/cm^2 (hot leg),
- effluent volume: 56 l/m^2 for two steps (basic gel and acid gel),
- effluent activity: 8000 Bq/l,
- effluent activity after sludge separation: 1000 Bq/l.

In-situ decontamination tests

These tests, performed inside the pipes of the G2 cooling circuit (diameter = 1.6 m, length = 10.9 m, surface = 55 m^2), confirmed and even surpassed the previous results: the residual activity and the effluent volume are reduced. The operator entered the pipe and operated manually for gel spraying and rinsing, using an airless compressor. The results of this operation are the following:

- mass of basic gel sprayed: 24 kg (435 g/m^2),
- mass of acid gel sprayed: 17 kg (310 g/m^2),
- initial activity: 100 Bq/cm^2,
- residual activity: 0.2 Bq/cm^2,
- volume of basic and acid effluents: 396 l (7.2 l/m^2),
- activity of effluents after mixing and sludge separation: 810 Bq/l.

2. Decontamination by abrasive blasting (B.1.)

In March 1987, a sodium leakage was detected in the gap between the double vessels of Superphenix fuel storage drum; about 25 m^3 of sodium flowed in this volume (63 m^3). After draining the sodium from the fuel storage drum and the intervessel gap, both vessel walls must be entirely cleaned according to requalification criteria, i.e. the sodium must be completely eliminated without any formation of aqueous sodium hydroxyde at temperatures above 60°C. Flushing the gap with air will convert the residual sodium into soda and then into carbonate.

The aim of this study is to find a mechanical method able to clean both the vessel walls of the sodium carbonate coming from the flushing in air of residual sodium after the draining of the intervessel gap.

At first, several abrasive types (spheric and angular particles of glass, iron oxide, aluminium oxide, steel etc.) were tested with a view to the following criteria:
- sodium carbonate cleaning efficiency,
- support erosion,
- abrasive pollution during recycling,
- recycling evaluation cost.

The sintered iron oxide microbeads were selected as abrasives.

Then, parametric tests were performed to define exactly the optimum process conditions, which are:
- incidence angles: 45° and 75°,
- nozzle/target distance: 6.5 cm,
- air projection pressure: 5 bars,
- moving speed of blast pipe: 60 m/h,
- abrasive granulometry: 200 - 1000 µm.

Finally, these optimal conditions were tested on:
- the laboratory samples and
- an air intervessel demonstration model of scale 1 (height = 11 m, width = 1.2 m).

This demonstration testing gave good results and validated the blasting of sintered iron oxide microbeads as a method for cleaning the vessel walls of the sodium carbonate.

An application to the Superphenix fuel storage drum will be carried out probably during the second half of 1988.

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

Summary

This report covers tests undertaken during 1987 in the Karlsruhe Nuclear Research Centre for which primarily real radioactive samples were used. This was actually done in the electropolishing plant which was developed, designed and constructed in the course of this R&D programme, using the alternative electrolyte which operates on the basis of acetyl-acetone and KBr.

Very extensive tests under real conditions were made with this electrolyte, and components removed from the shut-down FR2 experimental reactor were used as samples. The results obtained were not, altogether quite satisfactory, mainly due to the rough sample surface caused by the electrolyte action, and the resulting long treatment and processing times.

However, under the aspect of handling and secondary waste management, the alternative electrolyte attained in any case the expected results, thus having major advantages as compared to the conventional inorganic electrolytes.

Progress and Results

1. Electrolyte testing (B.3.)

The electrolyte acetylacetone was selected. The following composition was used: acetylacetone (5%) + KBr (1M) + iso-propanol (25%) + water at the pH = 4. The tests were made with a current density of 20 A/dm^2. The residual activity of the samples was measured after each treatment (of 5 to 10 minutes) and was recorded accordingly. A closed-loop cooling system was intended to maintain the electrolyte temperature between 20 and 40°C. Analogous to the sample geometry, it was decided to have a tube as cathode. The clearance between the electrodes amounted to approximately 30 mm. When selecting the samples it was made sure that the contamination (2 - 10 Bq/cm^2) adhered so firmly to the surface that the smear test proved to be absolutely negative. The bottom activity limit was defined to be 0.37 Bq/cm^2. In order to create favourable thermodynamic and hydrodynamic conditions in the electropolishing bath (Figure 1), the electrolyte was circulated with a pump at a rate of 60 l/min.

2. Radiological service life (B.5.)

As expected, the metal salts which were being formed (ferric acetylacetonate, cobalt acetylacetonate), settled out of the electrolyte solution in the course of the tests under real conditions, using radioactive stainless steel samples and the alternative electrolyte. As a result of the metal salt particle size, it was easy to separate them from the liquid-solid mixture by sedimentation. Tests undertaken with coarse filters (above 10 microns) also ensured a satisfactory separation. This separation and settling out of the salts (approximately 1.5 kg for 1 m^2 electropolished surface) extend the radiological service life of the new electrolyte. Due to the very low contamination of the samples, it was not possible to give evidence of the calculated radiological solubility limit of 10 MBq/m^3.

3. Waste treatment (B.4.)

The radioactive metal salts settled out during the tests or retained in filters, were subsequently dried and then compacted. Compaction at 16 MN produced a volume reduction factor of 3, reducing the primary waste volume to 0.5 l for 1 m^2 electropolished surface. If required, it is possible to process and convert the electrolyte into a state fit for ultimate storage, by evaporation and subsequent conditioning of the evaporator concentrate.

4. Electrolyte handling (B.7.)

With a view to its handling, the electrolyte is not chemically aggressive and does not raise any toxicity problems, if the legal safety requirements are respected as prescribed by the "Unfallverhütungsvorschriften, UVV".

5. Results (B.3.)

For stainless steel, the highest decontamination factor obtained in the tests was DF = 36, which required a treatment of 50 minutes electropolishing.

In Figure 2, the surface activity is shown as a function of the process time. The initial activity (5-9 Bq/cm^2) declines relatively rapidly to 0.8 Bq/cm^2. To obtain, however, a value below the limit for unrestricted release (0.37 Bq/cm^2), a relatively long treatment is required.

It is possible to interpret the obtained results as follows: material was removed from the surface at various spots, which resulted in a rough surface. This means that the surface was not removed as an even film, and the residual activity was therefore found in the part of the surface presenting irregularities.

A few tests with carbon steel samples showed that the electrolyte had satisfactory and even good results. A comparably short process time was required, resulting from a relative smooth surface after treatment.

Samples which were covered with a solid oxide film of magnetite or haematite could not be treated and processed, which means that it was not possible to remove the sample surface, whereas secondary reactions went on in the electrolyte.

In Figure 3, the activity of the electrolyte is represented as a function of the experimental time. It is evident that the estimated radiological activity limit of 10 MBq/m^3 was not reached. This was mainly due to the low contamination of the samples.

The radiological examination showed that the specific activity of the sediment is a five times higher than that of the electrolyte, and this allows to conclude that the radioactive salts contained in the electrolyte settled out.

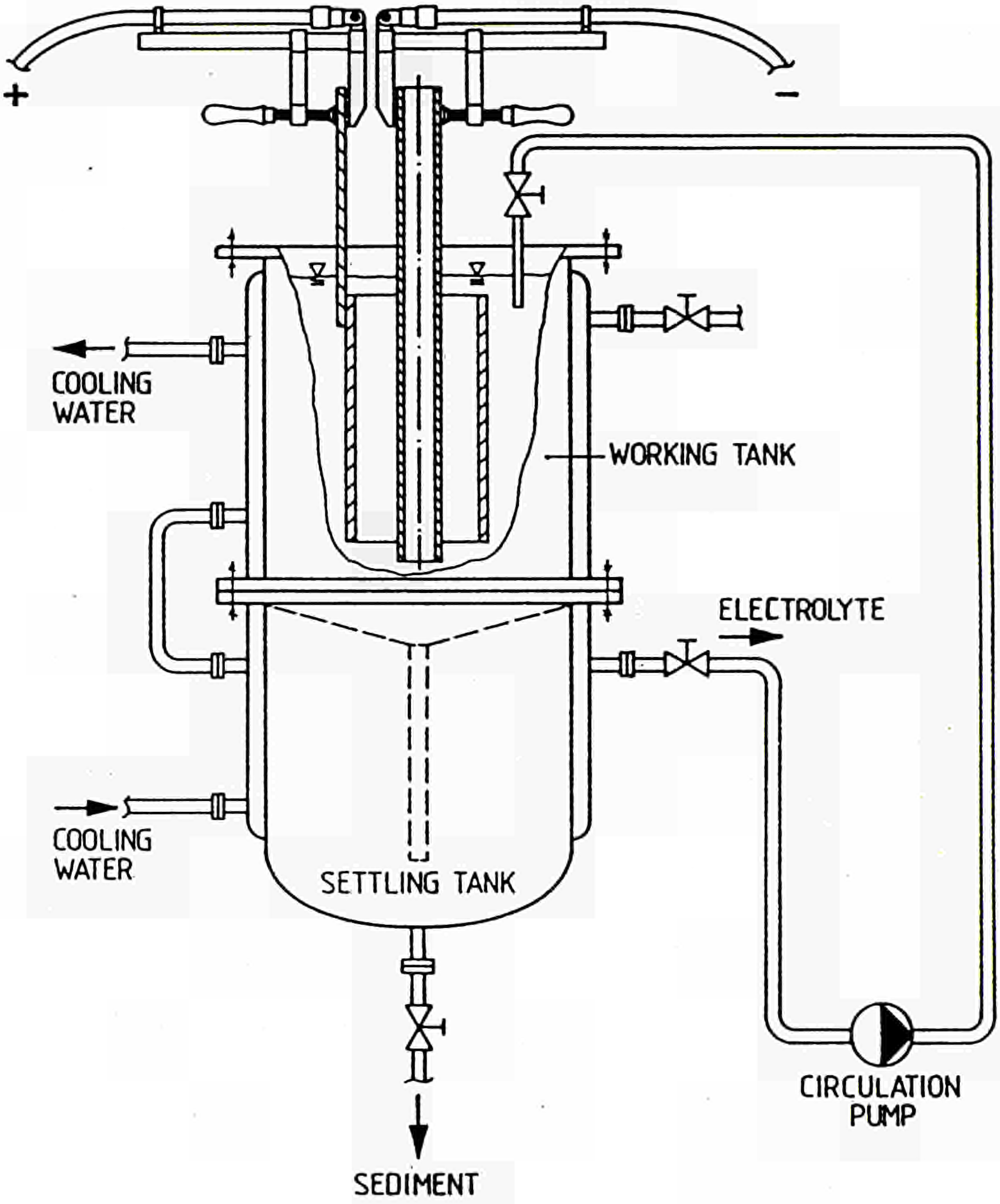


Figure 1: The electropolishing pilot plant 400 A

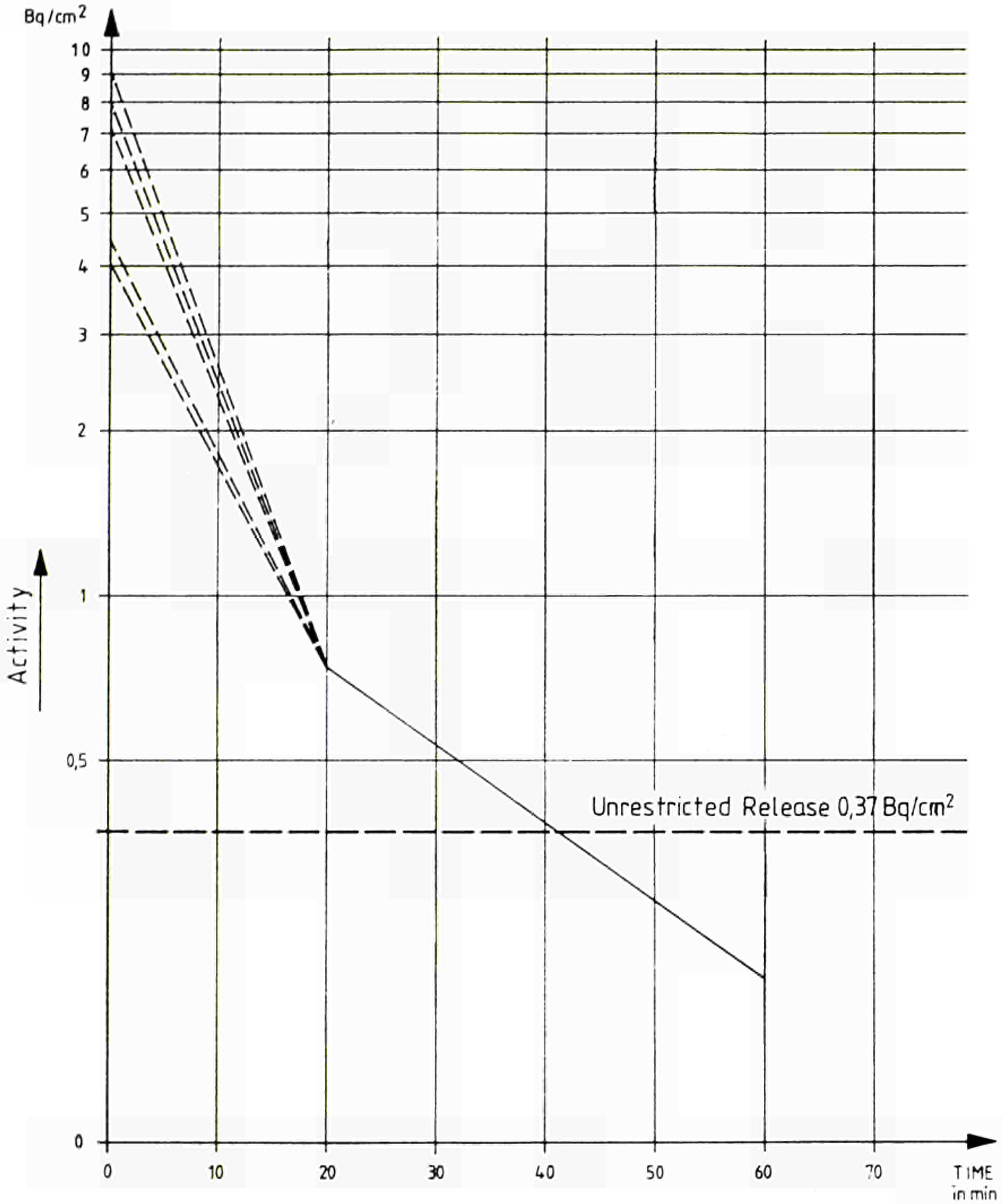


Figure 2: Surface activity as a function of time

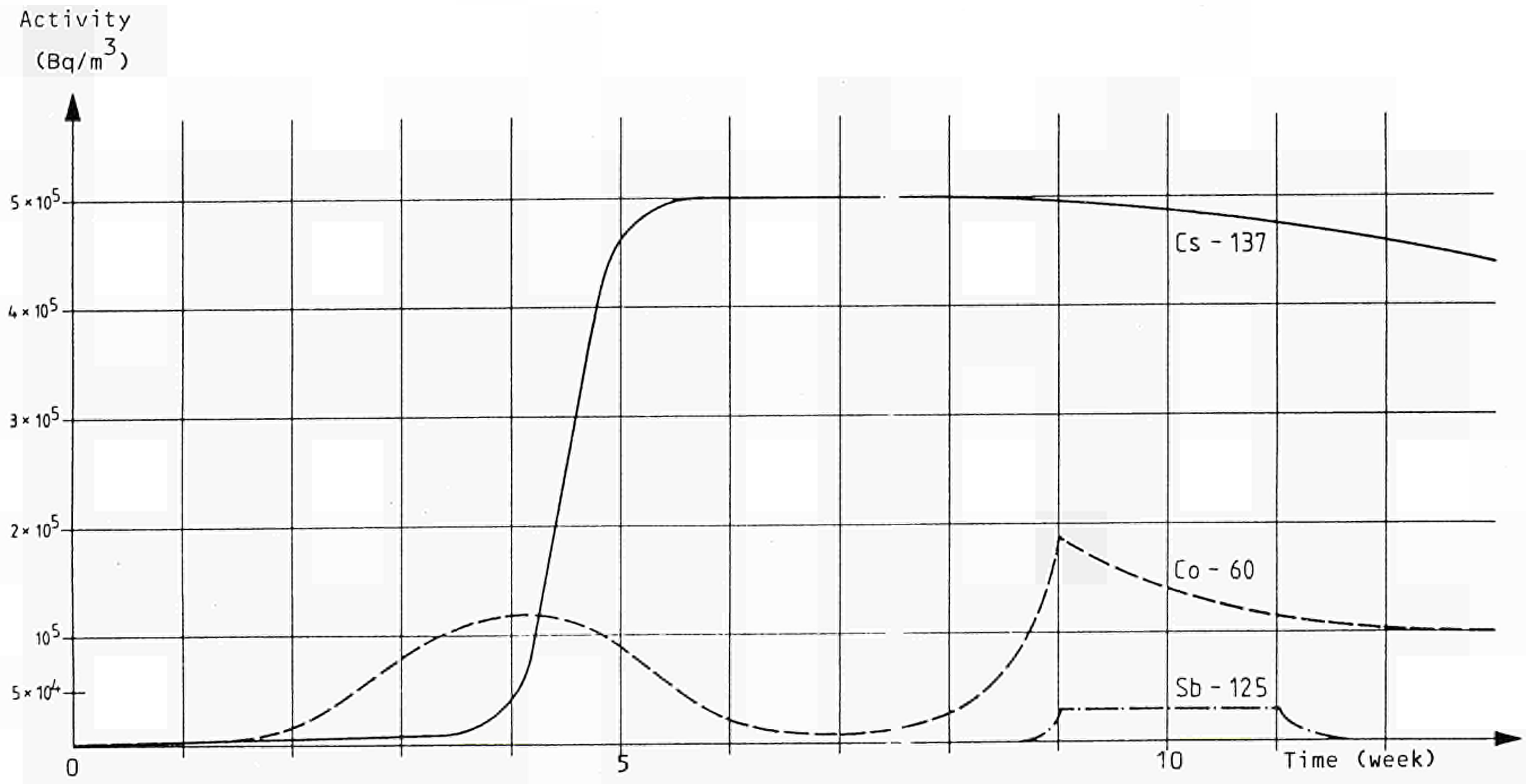


Figure 3: Activity of the electrolyte as a function of time

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, multistage 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

Summary

In the Annual Progress Report 1986, a filter system was presented, the dust separation rates of which could not be determined because of a faulty measuring device. Measurements for this filter system were repeated in 1987.

To determine the separation behaviour, the dust contents in the crude gas and clean gas were measured. Moreover, grain size analyses were carried out before and behind the filter.

The total separation degrees during separation of the dust, which arise during flame scarifying and the subsequent brushing of the concrete surface were more than 99.99%. The pressure drop in the non-dusted filter was 120 Pa, the filter was cleaned at a pressure drop of 1200 Pa. After cleaning, the pressure drop was 500 Pa.

The filter system described in the Annual Progress Report 1985 was tested at the premises of the GKSS Research Centre, where contaminated steel plates from the Brunsbüttel nuclear power plant were decontaminated by grinding. About 250 kg dust were separated from the exhaust air during machining of an area of about 200 m^2. The suitability of the filter for decontamination operation could thus be proven.

Progress and Results

1. Tests on uncontaminated concrete samples

The investigations carried out in the latter half of 1986 produced no usable results due to malfunction of the measuring device used to determine the clean gas dust content and the particle size analysis. For this reason, the measurements were to be carried out at the same filter plant in 1987.

All investigations carried out to date have shown that the highest raw gas dust loadings are achieved with flame scarfing and subsequent cleaning with a wire brush. As the particle sizes of the dusts arising from this removal method are also smaller than with the other processes (chisel hammer, spike hammer, grinder), the tests were continued using only the flame scarfer and the wire brush. If these dusts, which are unfavourable for a filter plant, can be sufficiently separated, there is no problem to be expected in the filtering of the dusts arising the other removal processes.

In the tests, the dedusting plant of the DELBAG company was used, which is described in the previous annual report. The filter was a cartridge filter, type ITK.

1.1. Dust loadings and separation performance of the filter

The dust loadings measured in the raw gas and the clean gas show that both with flame scarfing and with subsequent cleaning of the concrete, excellent separation degrees were achieved with the investigated filter (Table I).

1.2. Particle size analyses

The particle size analyses upstream and downstream of the filter can be seen in Figures 1 and 2. A shifting towards smaller particle diameters in the clean gas can also be seen with this filter plant.

1.3. Filter cleaning

As already described in the previous annual report, the automatic filter cleaning is adjusted to meet the conditions present in the raw gas. According to the manufacturer, the filter should be cleaned after reaching a pressure drop of approximately 1200 Pa with the dusts arising during flame scarfing and brushing. In continuous operation, the automatic timer is set at the appropriate values. In trial operation, the cleaning unit was manually started after reaching the given maximum pressure drop.

The degrees of separation of the filter were comparably good after several periods of operation at full power and cleaning. The pressure drop of the new filter was 120 Pa ($V = 2500\ m^3/h$). After cleaning of the full filter, the pressure drop was approximately 500 Pa. Figure 3 shows the pressure drop at the filter as a function of the amount of dust filtered.

2. Decontamination tests

In decontamination operation by Noell GmbH on the premises of the GKSS, the regenerative filter of the Ama-Filter company described in the Annual Progress Report 1985 was used.

2.1. Tests performed

During the tests, contaminated plates were decontaminated using grinders. The contamination was composed of approximately 80% Co-60 and 20% Cs-137. During the grinding of these plates, attention was paid to generate as much dust as possible, deviating in this regard from realistic conditions.

The grinding of the plates was carried out under a tent. During extraction, a differentiation was made between:

- direct suction at the location of the dust arising,
- extraction of the air in the tent.

The extracted dust is retained in the filter. When the filter is cleaned, the dust falls into a collection vessel installed below the filter plant. By-pass lines were installed in the raw gas and clean gas lines of the filter line. Using a dust sample collector, filter paper was exposed to dust in these by-pass lines. The exposure time of the filter was approximately 5 minutes and the air volume approximately 4.5 m^2. The filter papers exposed to the dust in this way were evaluated radiologically in a counter with a large surface area.

2.2. Results

An area of approximately 200 m^2 was decontaminated during the tests. The specific activity of the contamination layer was determined by scraping off samples. The average specific activity was 22.5 Bq/g. The dust collected in the vessel below the filter plant had a weight of 250 kg, a volume of 0.05 m^3, and an activity of 2.5 MBq (specific activity: 10 Bq/g). This means that approximately 55% of the dust results from the wear of the grinding discs. As therefore approximately 45% of the dust results from the contaminated surface, i.e. 112.5 kg, this indicates an average removal of 57 mg/cm^2 over an area of 200 m^2. The separation rates of the filter for both types of suction are given in Table II.

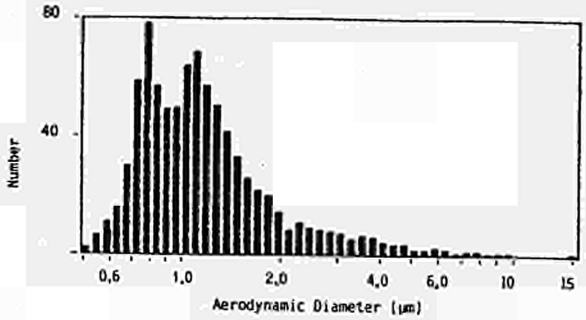
The results clearly prove the suitability of the filter for decontamination operation. For safety reasons (rupture of the filter), however, any filter plant in decontamination operation should be equipped downstream with a filter of "S" filter classification.

Table I: Dust loadings in raw gas and clean gas and separation degree of ITK-filter

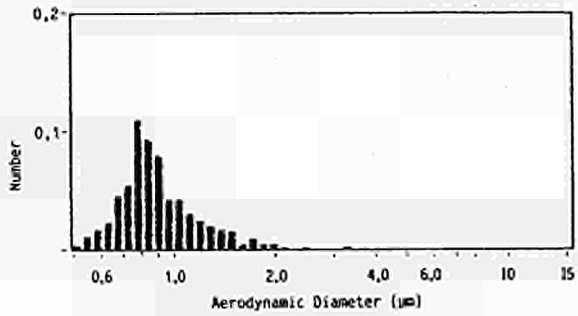
Stripping method	Dust loading in raw gas (mg/m^3)	Dust loading in clean gas (mg/m^3)	Degree of separation (%)
Flame scarfing	55.4	2.14×10^{-3}	99.996
Wire brush	687.5	2.45×10^{-3}	99.9996

Table II: Degree of separation of the AMA-filter in the decontamination operation

Type of suction	Average contamination in raw gas (Bq/m^3)	Average contamination in clean gas (Bq/m^3)	Degree of separation (%)
Direct dust suction	1,443	7	99.52
Suction of the air in tent	397.5	4.5	98.9

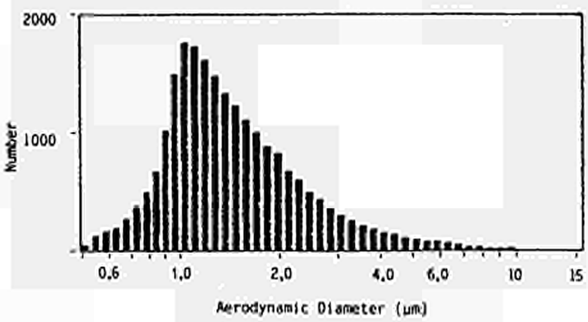


a) Raw gas

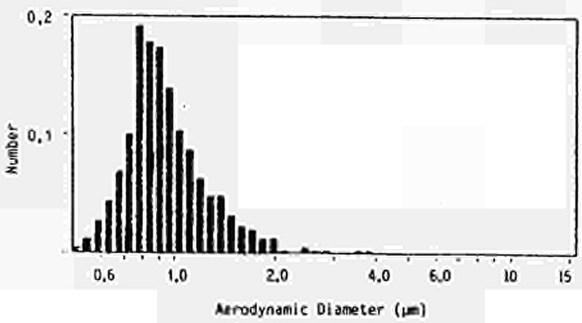


b) Clean gas

Figure 1: Histogram of particle size distribution (flame scarfing)



a) Raw gas



b) Clean gas

Figure 2: Histogram of particle size distribution (wire brush)

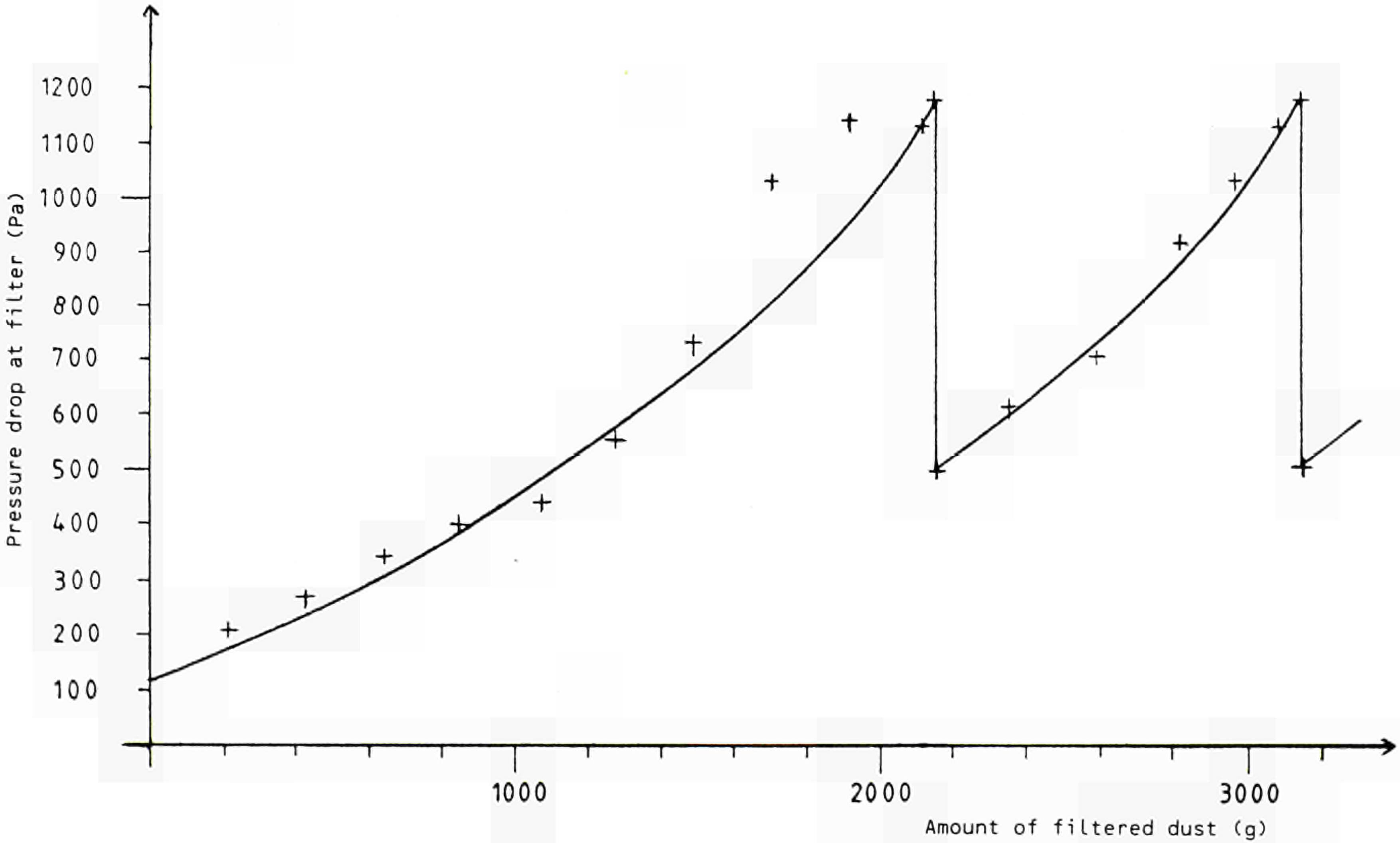


Figure 3 : Pressure drop at filter as a function of the amount of filtered dust

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 - December 1988
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.1.2. 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

On the basis of previous laboratory scale experiments, the detailed design of a 0.3 m^3 electrolytic immersion decontamination tank has been completed, and the fabricated unit should be delivered in March 1988. The microprocessor controller construction should also be completed by this time for subsequent commissioning trials. Due to this slight delay, commissioning and active use will be correspondingly deferred.

In order to optimize decontamination effectiveness, a reference electrode system is required to help control the cell potential. A range of possible systems have been experimentally evaluated, from which a short list of five has been chosen for further examination on the basis of having a voltage stability better than 10 mV over prolonged periods. These included a sheathed glass electrode, palladium and niobium metal/metal oxide and platinum and 310 stainless steel dynamic electrodes (where an external current is passed through the electrode to drive an irreversible reaction). Final selection will be made on the basis of potential stability with respect to acidity changes.

The construction and inactive commissioning of a mobile electropolishing hot-spot decontamination unit has been completed. As a result of these preliminary tests, several alterations were made to the system to improve its operation. Although one or two of these remain to be completed, full electropolishing tests have been performed successfully. System modification is already in hand to prepare the unit for fully active use - comprising the transfer of the vacuum system HEPA filter and waste electrolyte collection tank into the containment. In addition, a small probe (15 mm diameter) for the treatment of curved and less accessible areas has been fabricated and successfully tested.

Progress and Results

1. Design and construction of full size immersion electroetching unit (B.1., B.1.2)

The detailed design and manufacturing drawings of a 0.3 m^3 immersion electrolytic decontamination tank unit have been finished and fabrication is now expected to be completed by March 1988. A microprocessor controller to manage the decontamination system has been designed and is currently being constructed. This automates the system and also permits safe operation without the need for operator intervention due to extensive monitoring of process parameters.

The decontamination operation is controlled through the voltage between the titanium current feeders in contact with the item being decontaminated and a monitoring reference electrode in order to optimize performance. During the final phase of treatment, there is also an option to reverse the cell current so as to minimize any readsorbed activity. Electrolyte is then removed to an enclosed storage vessel while the potential is still applied. Rinsing and drying are then carried out in the same tank to minimize activity spread - the rinse water being continuously pumped out to a waste container. If it is desired to re-use the electrolyte, this can be dosed with concentrated acid equivalent to that consumed during the previous run prior to being pumped back into the decontamination tank.

The calomel system (SCE) normally used as a reference electrode in the laboratory is unsuited for hot-cells as it contains chloride and mercury, and requires frequent maintenance. A programme has therefore been carried out to identify suitable alternative systems. This was done by immersing the candidate electrodes in 1M HNO_3 and comparing the variation of their potential in time with that of the SCE. Three types of system were examined:- membrane electrodes (glass electrode protected by a sheath), metal/metal oxide electrodes and dynamic electrodes (where an irreversible electochemical reaction is carried out). Five possible electrodes were identified as being capable of providing reference stability to within 10mV over prolonged periods:- glass electrode; palladium and niobium metal/metal oxide; oxygen evolution from platinum, and 310 stainless steel dissolution dynamic electrodes. As some of these systems are sensitive to acid concentration even at this low pH (eg metal/metal oxide), these may be less useful in the decontamination tank where acidity is progressively changing - although this effect may be minimized in the protected electrolyte tube where the reference electrode is located, due to negligible mixing with the bulk solution. However, such acid sensitive systems may be of use in readjusting acid strength back to 1M after each run. Further work is continuing on this topic.

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

The construction of a prototype mobile decontamination unit has been completed. This comprises a stainless steel trolley on which power supply, microprocessor controller, electrolyte feed and effluent tanks, solution pumps, vacuum pump and electromagnetic valves are mounted (Figure 1). The system is equipped with an extensive range of sensors to enable safe operation of the system. The decontamination head (Figure 2) is connected to the trolley through an umbilical carrying electrical cables and fluid tubes.

The various components were tested individually after installation, and then in assembled sections - initially with only very dilute nitric acid. Only when the complete system was operational was the electrolyte changed to 8M nitric acid.

During commissioning, several modifications were carried out - including the addition of a lime tower to adsorb any acid fumes from the vacuum pump; and several electrical circuit changes (eg to prevent interference from a noisy mains supply). Some alterations to the decontamination head were also found necessary to minimize the effect of any trapped air in the system disturbing the uniformity of fluid flow, and also to retain more securely in the front face, the seals which contact the surface being treated. However, full electropolishing tests were successfully performed, although several areas of work for further system improvement have been identified.

As part of the programme to examine alternative decontamination head geometries, a circular device treating a 15 mm diameter has been fabricated and successfully tested. Due to its small size, the electrolyte flow requirements during decontamination were negligible.

System modifications for fully active use of the trolley have also been put in hand. This comprises the transfer of the vacuum pump HEPA filter and electrolyte effluent tank and their association traps into the containment on a separate framework.

List of Publications

/1/ TURNER, A.D., JUNKISON, A.R., POTTINGER, J.S. and LAIN, M.J. Electrochemical Decontamination: Annual Progress Report April-December 1986, AERE-R12508.

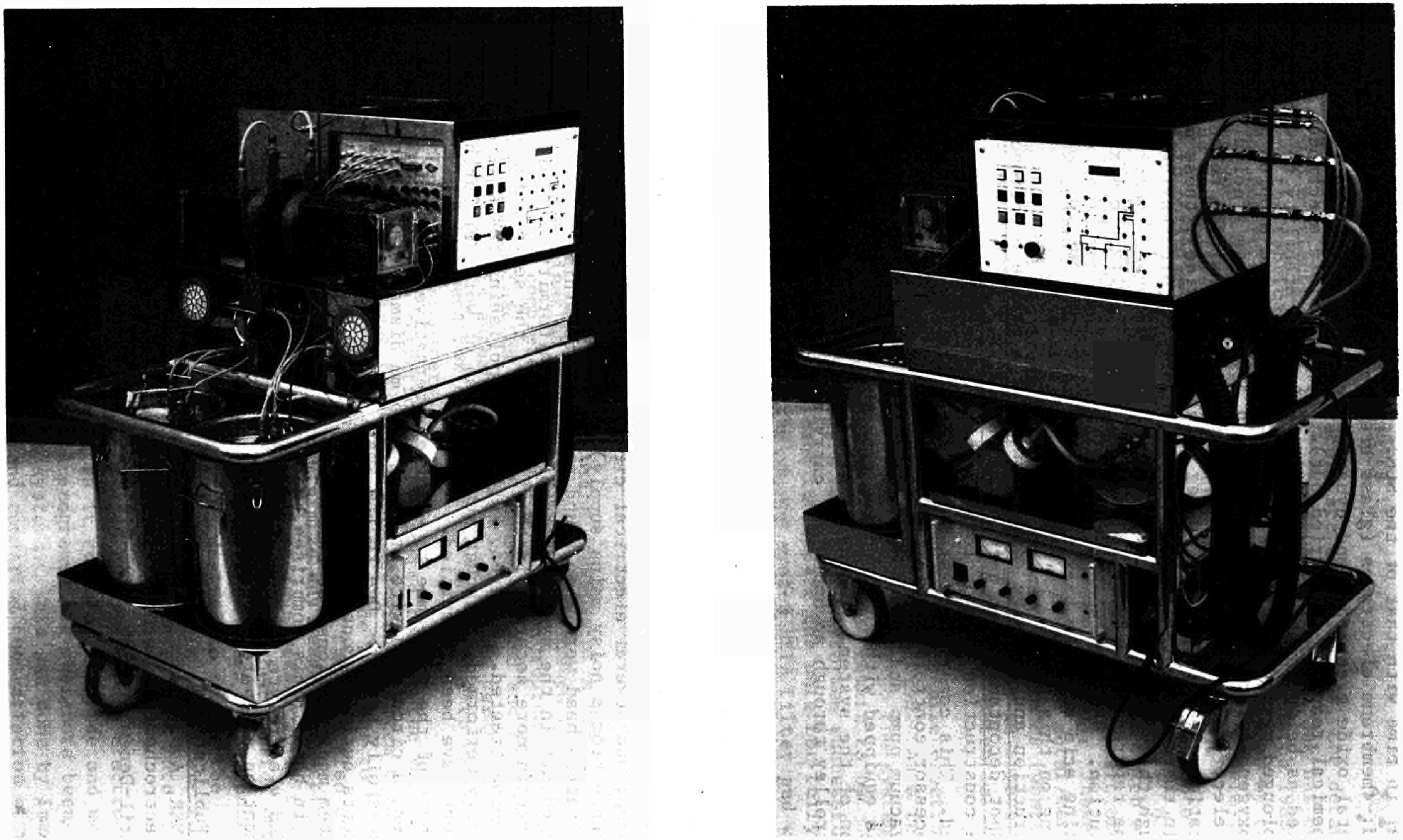


Figure 1 : Mobile decontamination unit

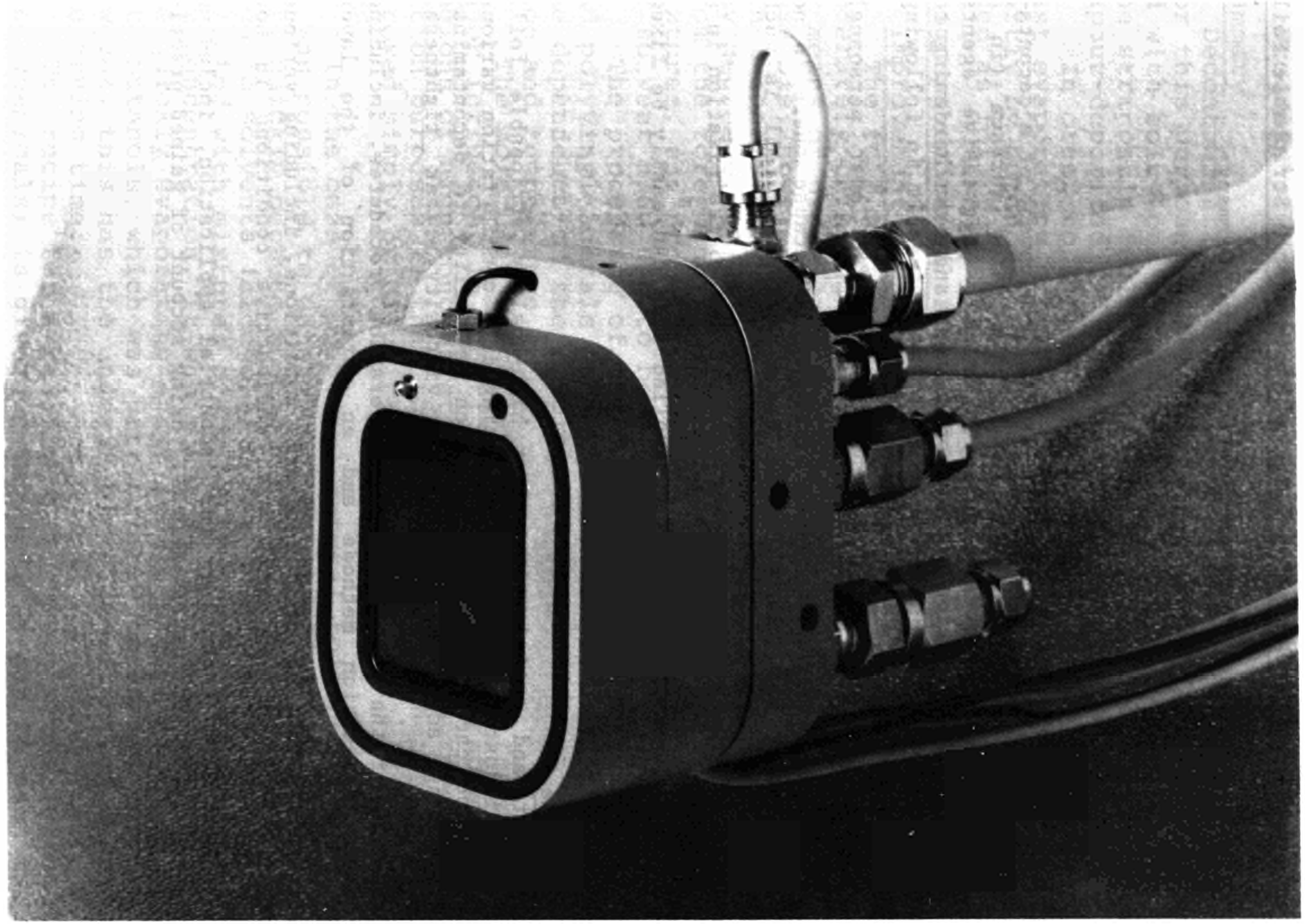


Figure 2 : Decontamination Head front face

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

Decontamination factors were systematically determinated. For this purpose samples of various materials were contaminated with solutions of salts of α, ß and γ radiating isotopes. The strippable foil was also applied for decontamination of vapour-deposited ThF_4 layers on Al substrates.

In order to improve the sprayability of foil material the fast evaporating ethyl and isopropyl alcohols were replaced partly by butanols, and less viscous polyvinyl butyrals were used as foil base material.

Progress and results

1. Improvement of sprayability of foil material (B.2.)

The foil mixture used hitherto could only be applied by moderately thinning it by the usual paint-spraying method (i.e. by means of propellant gas and compressed air). It was not possible, however, either to atomize the material in its original consistency or to spray it by the airless method after any degree of thinning.
From the foregoing it was concluded that the viscosity of the mixtures, the properties of the solvents used and the quantity of solvent used had to be modified.

The processing of the foil material (polyvinyl alcohols and polyvinyl acetals and butyrals) is governed particularly by the dependence of the rheological properties on the concentration and temperature.

The molecular weight and saponification factor have a major influence. In addition, association phenomena occur in regions of high concentration. An important property for application purposes is that the associates are not necessarily destroyed simply by diluting a thickened PVA solution with water.

In the case of polyvinyl butyral foil containing solvent it was possible to determine compressed-air-sprayable mixtures more quickly because the solubility of the base material in organic solvents (alcohols) meant that the viscosity could be selected with mixtures of solvents in addition to variation of the concentration. It was possible, for example, to replace the quickly evaporating ethyl and isopropyl alcohols partly with butanols, which was particularly conducive to atomization. However, this has the undesirable side-effect of increasing the drying time.

The recipe for a readily sprayable mixture (without pickling chemicals) is given in Table I. It can be seen that the more viscous product BL18 has to be relatively strongly thinned to allow it to be used for the compressed-air spraying method.

The large amount of n-butanol added then leads to undesirably long drying times. Furthermore, it is difficult to achieve a sufficiently thick film in one coating without the material sliding down vertical surfaces.

2. Systematic determination of decontamination factors (B.3.)

Test procedure: square samples (side length 50 mm) of stainless steel, C steel, aluminium, painted C steel, PVC and plexiglass were prepared in order to determine the decontamination effect of the strippable decontamination foil. These samples were contaminated in a specific manner. A solution of

$$Cs137 + Sr90 + V90 \text{ in } 0.1N \ HNO_3$$

was prepared for beta and gamma radiation. The specific solution activity was 0.4 μCi/ml. For alpha contamination a plutonium solution of 0.422 g Pu with the following isotope composition in 100 ml solution was prepared: Pu 238 0.1 %; Pu 239 88.5 %; Pu 240 10.1 %; Pu 241 0.8 % and Pu 242 0.2 %. A series of samples were contaminated by spraying the surface of each with 1 ml beta and gamma contamination solution. Another series of samples were likewise alpha-contaminated with a Pu solution.

After being dried, all samples were coated by spraying them with strippable decontamination foil. After 4 days the activity of the contaminated samples A_1 coated with the foil were measured. The foil was then stripped off and the residual contamination A_2 on the samples was again measured.

The experimental results are presented in Tables II and III.

Examination of the decontamination factors shows that the values determined are in the region of DF = 10 and P = 90 %.

The strippable decontamination foil can thus be assigned to class 2 of the decontamination agents according to NFT 30-901: "Peintures pour l'industrie nucléaire; essai de décontamination".

These results are in agreement with the values obtained in first practical tests. The decontamination factors of the strippable foil are higher (better decontamination effect) than the usual values for conventional decontamination agents on the basis of tensides, which are generally assigned to class 3 or 4.

3. <u>Decontamination of vapour-deposited ThF_4 layers</u> (B.2.; B.3. B.4.)

All substrates coated with vapour-deposited ThF_4 layers were decontaminated by repeated application of strippable decontamination foil. The substrates have been used in an industrial vacuum coater for different purposes: sample holder, heat and radiation shields, shutter and evaporation masks. During the fabrication of optical systems, the substrates became coated with a variety of different chemical substances e.g. MgF_2, MgO, SiO, ZnS, CaF_2 and ThF_4, which is radioactive and emmits α and γ radiation. Therefore, these substrates had to be decontaminated before being re-used.

In some cases the application of the strippable foil had to be repeated; this was necessary especially when the vapour coatings were produced by oxide layers.

The results of this application of the strippable foil are given in Table IV.

Table I

Recipe for a readily sprayable mixture (without pickling chemicals)

Foil material	BL 18 600 g	BN 18 600 g
Solvent I	iso-propanol, 2400 ml	
Solvent II	poss. mixt. of 50% eth.	10% n-but.
Softener	15 % OEG 400	450 ml PEG 400
Thinner	30 % 1-butanol	n-butanol
Flow agent	1–2 % No. 803127	max. 1 % No. 803127

Table II

Beta and gamma decontamination tests with strippable decontamination foil.
Background Ao = 5.000 ipm

Impulses Material	A_1	A_2	DF	P[%]	Remarks
stainless steel					
sample 1	260,000	24,000	13.5	92.5 %	very good strippability
sample 2	190,000	14,000	20.5	95.1 %	of the foil
aluminum					
sample 1	150,000	23,000	8	87.5 %	surfaces of samples slightly corroded by HNO_3; foil not strippable
sample 2	170,000	21,000	10.5	90.3 %	some difficulty in stripping foil
C steel					
sample 1	150,000	?	?	?	samples very highly corroded by HNO_3
sample 2	170,000	?	?	?	foil not strippable
painted C steel					
sample 1	230,000	22,000	13	92.4 %	some difficulty in stripping the foil owing to corrosion, but strippability
sample 2	270,000	35,000	9	88.7 %	better than with Al samples
PVC					
sample 1	195,000	35,000	6.5	73.7 %	very good strippability
sample 2	175,000	23,000	9.5	89.4 %	of the foil
plexiglass					
sample 1	190,000	25,000	9.5	89.2 %	very good strippability
sample 2	95,000	13,000	11.5	91.1 %	of the foil

A_1 = Activity before decontamination (ipm)
A_2 = Activity after decontamination (ipm)

Table III: Alpha decontamination tests with strippable foil

Material	A_1	A_2	DF	[%]	Remarks
stainless steel					
sample 1	17,000	4,000	5	80 %	very good strippability
sample 2:	foil removed during handling, no measurement				
aluminum					
sample 1	15,000	3,000	6	84 %	some difficulty in stripping the foil;
sample 2	19,000	3,000	8	87.5 %	carry-over of contamination?
painted C steel					
sample 1	45,000	2,000	34	97%	some difficulty in stripping the foil;
sample 2	13,000	2,000	9.5	89.5 %	carry-over of contamination?
PVC					
sample 1	30,000	4,000	9	88.5 %	very good strippability
sample 2	28,000	21,000	?	?	of the foil
plexiglass					
sample 1	15,000	2,500	8	87.5 %	very good strippability
sample 2	28,000	2,000	21	95 %	of the foil

A_1 = Activity before decontamination (ipm)
A_2 = Activity after decontamination (ipm)
Background activity: A_o = 700 ipm

Table IV: Decontamination by multiple application of foil MO/140/4 and MO/140/5 on aluminium samples

Item No.	Initial activity α/γ (ipm)	Activity of samples after foil application (ipm) MO/140/4 1st	MO/140/4 2nd	MO/140/4 3rd	MO/140/5 4th	DF overall
019	200/30	150/20	200/40	250/80	40/12	6/3
020	150/50	100/30	100/30	150/60	40/ 6	4/23
021	40/ 4	40/ 4	40/ 8	15/ 4		4/
022	400/100	300/60	250/50	200/60	30/ 6	17/48
023	50/10	40/ 8	40/ 8	30/15		2/
024	100/45	70/20	50/15	30/ 6	15/ 4	12/21
025	300/110	40/10	20/ 4	15/ 4		37/100
026	150/50	40/ 4	15/ 4			18/50
027	60/20	40/ 6	30/ 4			2/20
028	100/50	70/25	40/15		40/15	not termin.
029	70/18	20/ 3	15/ 4			8/14
030	70/15	15/ 3				8/11
031	100/25	15/ 2				12/21
032	150/40	100/50	100/50	80/50		not termin.

Background activity: α_o = 7 ipm, γ_o = 4 ipm

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

Summary

Items B.1., B.2., B.3. were carried out. The tests performed in 1987 with the rack-torch prototype allowed to remove only 2 or 3 mm deep of concrete irregularly. Moreover, in some places, the surface is glazed, and it is then impossible to remove more concrete, even after removing the glazed layer. The addition of liquid nitrogen cooling, in order to intensify the thermal shock, did not permit to remove a thicker layer of concrete.

This is a good start for concrete decontamination, but the thickness removed is not sufficient in all cases. Therefore, the contractor did not realise the planned industrial prototype (B.4., B.5.).

Progress and Results

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

The prototype was designed for the following purposes:

- optimisation of parameters to remove a concrete layer as thick as possible, without and with liquid nitrogen.
- tests in inactive environment, exclusively.

This equipment did not require a particular study, but an adaptation of the equipment designed and manufactured by the contractor for tests anterior to this contract (see Figure 1).

Various thermal surfacers loaned by the company Air Liquide were also tested.

2. Optimisation of the concrete fissuration technique (B.2.)

During this optimisation campaign, much more tests than planned were performed, in order to find a way to remove a layer of concrete corresponding to the depth into which contamination can penetrate (5 cm). The tests showed that the use of a thermal surfacer to decontaminate concrete surfaces is not satisfactory, for the following reasons:

- the average removed layer is only 2 mm thick, while contamination can penetrate as deep as 5 cm;
- the surface obtained by thermal surfacing is very irregular, which makes necessary the use of a mechanical process (appropriate rotary brushes) to remove all the particles of fractured or melted concrete that came off the surface before a new passage of the thermal surfacer;
- the use of liquid nitrogen in order to cool down rapidly the concrete surface and intensify the thermal shock does not substantially improve the process. The cooling with liquid nitrogen hardly increases (7%) the temperature difference causing the thermal shock;
- after a first fracturation, additional passages do not produce further fracturation and too many of them can glaze the surface.

3. Scraping test on various types of concrete (B.3.)

The numerous tests performed on various types of concrete showed the possibility of fracturing a 2 to 3 mm thick layer of concrete at the first passage, the surface being very irregular because of the presence of big craters.

A second passage never brought any improvement, as the surface was either glazed or even more irregular.

The tests were performed on the following types of concrete:

- ordinary concrete plates,
- concrete civil engineering,

- "nuclear" concrete used for the radioactive wastes containers stored at "La Manche storage centre".

These three types of concrete are a good sample of the various existing types, from the least mechanically resistant (classical concrete) to the more resistant (nuclear concrete), including the civil engineering type.

4. <u>Design and manufacturing of the industrial prototype</u> (B.4.)

As the tests results were considered unsatisfactory by the contractor, the industrial prototype was not manufactured. The design of the industrial prototype (see Figure 2) includes the following components:

- a cover coated with special paint,
- driving wheels,
- pivoting wheels,
- a ventilation device with a decantor, a filter, and aspirating ventilation,
- a rack to collect concrete splinters,
- a rotary brush,
- two aspiration pipes,
- the thermal surfacer,
- the cooling rack,
- a protection grid.

5. <u>Application of the fissuration technique to a nuclear installation (B.5.)</u>

As the industrial prototype has not been manufactured, no tests have been performed in a nuclear facility.

6. <u>Conclusions</u>

The numerous tests carried out on various concrete types and with different thermal surfacers showed that it was possible to remove concrete of 2 to 3 mm thickness by the first application, but the surface after fracturation is very irregular. A second application never brought anything but the appearance of vitrification or a still more irregular surface.

The bibliographical research and temperature calculations confirmed that, concrete being a very bad heat conductor, it is impossible to create a deep thermal shock.

The contractor considered that these results did not justify the manufacturing of an industrial rack-torch prototype. The use of a mechanical process in order to decontaminate concrete by fracturation is better adapted than the one of a thermal surfacer, more delicate to operate, allowing to remove only 2 to 3 mm of concrete and necessitating the use of a mechanical process.

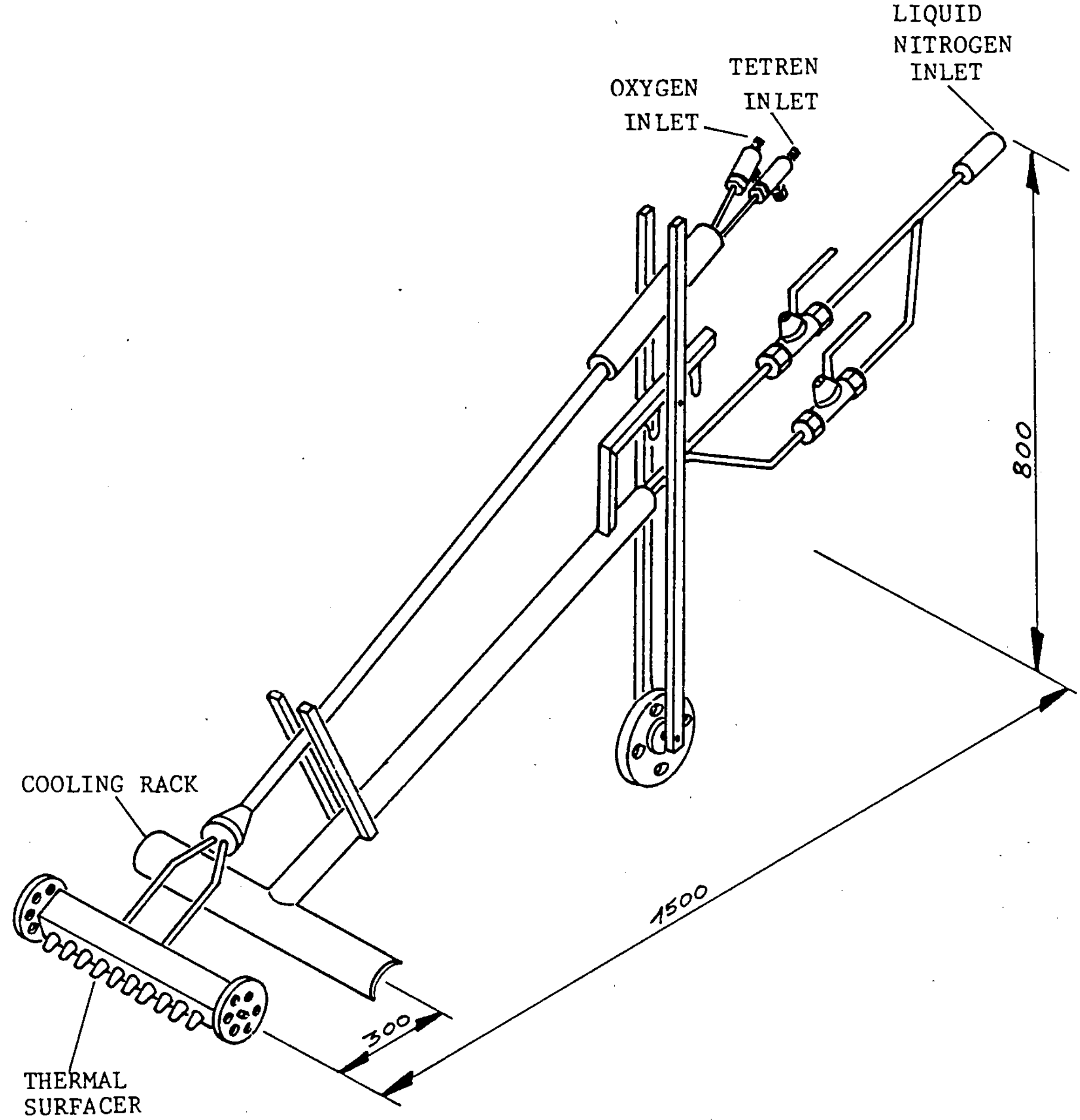


Figure 1 : Rack-torch prototype

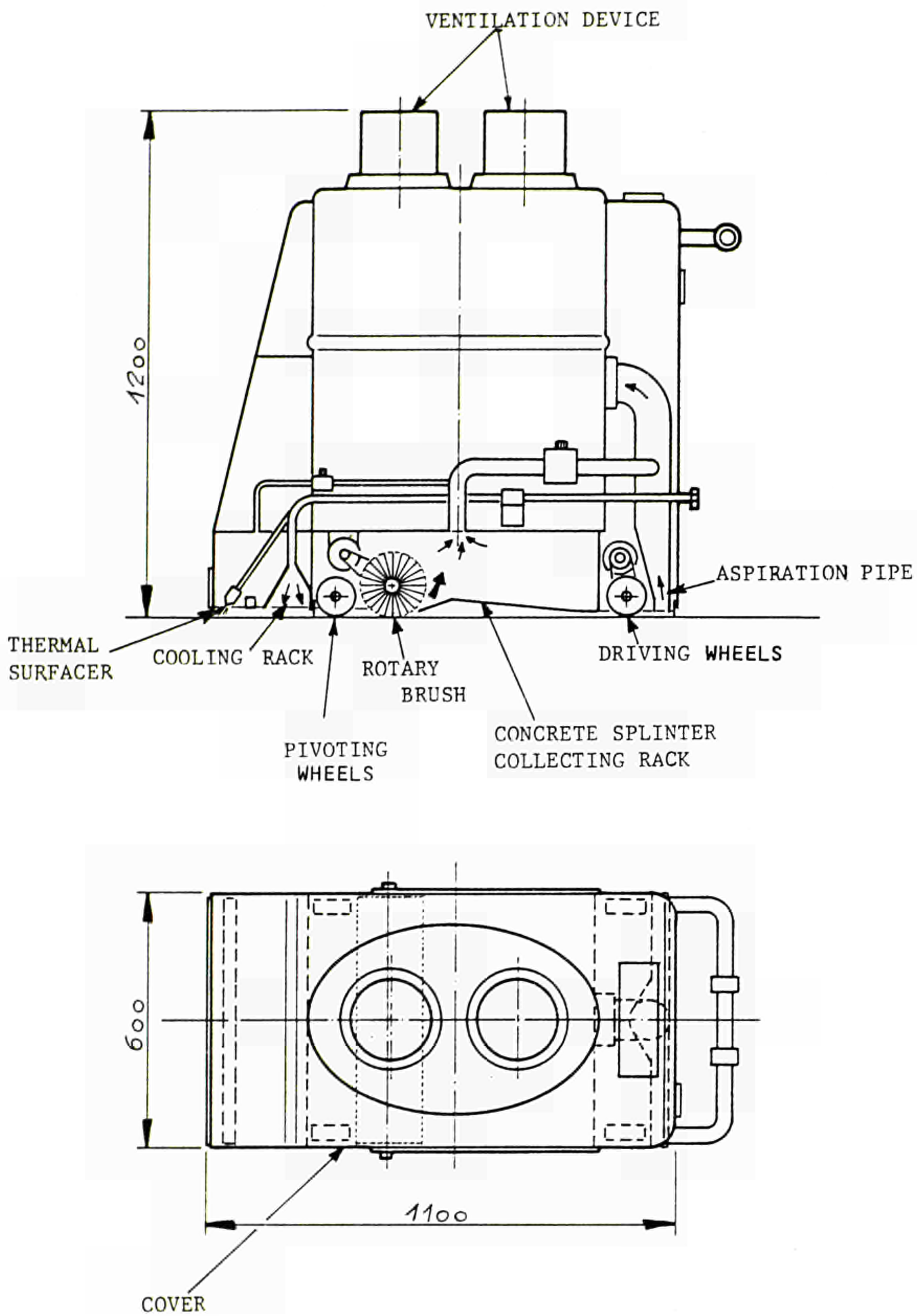


Figure 2 : Industrial prototype (project)

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 has been undertaken. It was demonstrated that this technique could remove concrete by superficial erosion and weakening. The analysis of the test results is going on to determine the efficiency and the viability of the process. The next step of the research work will be the design of a pilot unit.

Progress and Results

1. Modification and adaptation of the existing test facility (B.1.)

The test rig (Figure 1), adapted for qualifying the electrocombustion technique, is composed of:

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

Two types of concrete were tested:

- A: standard concrete with rock and sand aggregates (essentially SiO_2 and CaO),
- B: heavy concrete with baryte aggregates.

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

Two types of process gas were selected for the experiments:

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

High heat fluxes (about 1.6 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.

2. Experimental results (B.2.)

Figure 2 shows the erosion depth measurements as a function of the thermal exposure time for different experiments. The obtained results, although dispersed, are in agreement with those given by H. Sutherland /1/ and J. Muir /2/, as presented on Figure 2.

The superficial spalling of concrete, which may be the cause of the dispersion of results, can be explained as follows: the aggregate particles located close to the surface being exposed to important thermal shocks due to the strong energy supply react by changing the intercrystalline structure.

During heating, the concrete is loosing on its surface small pieces of material by following combined actions:

- inter-crystalline forces due to the effect of temperature and thermal flux;
- thermo-mechanical stresses induced by temperature gradients;
- due to dehydration and decarbonation on the surface and in the bulk of material, H_2O and CO_2 gases are set free, thus reducing adhesion and material strength.

The degraded concrete particles are removed from the surface and are falling by gravity into the containment without having reached the point of fusion. No vitrified particles have been observed (Figure 3).

The propane plasma + air configuration (1) seems to be more promising than the air plasma configuration (2), for the following reasons:

- low level of NO_x: 20 to 40 ppm for (1) - 1500 to 1700 ppm for (2);
- higher thermal efficiency of the impinging jet: large degradation of the concrete.

For a thermal exposure time of 120 s, a maximum erosion depth of about 5 mm could be obtained with a plasma (propane + air) jet, the diameter of the heated and treated surface varying from 70 to 110 mm.

On the basis of the first results of the analysis (aerosol emission and CsCl concentrations), it is expected that about 10 mm of concrete could be removed: 5 mm by thermal exposure and 5 mm by scraping, the heated concrete zone becoming brittle due to numerous cracks resulting from the thermal shocks.

References

/1/ H. SUTHERLAND - L. KENT - "Erosion rate measurements using an acoustic technique" - Rev. Sci. Instrum. 48, 1010 (1977)

/2/ J. MUIR - "Response of concrete exposed to a high heat flux on one surface". SAND77 - 1467, Sandia National Laboratories.

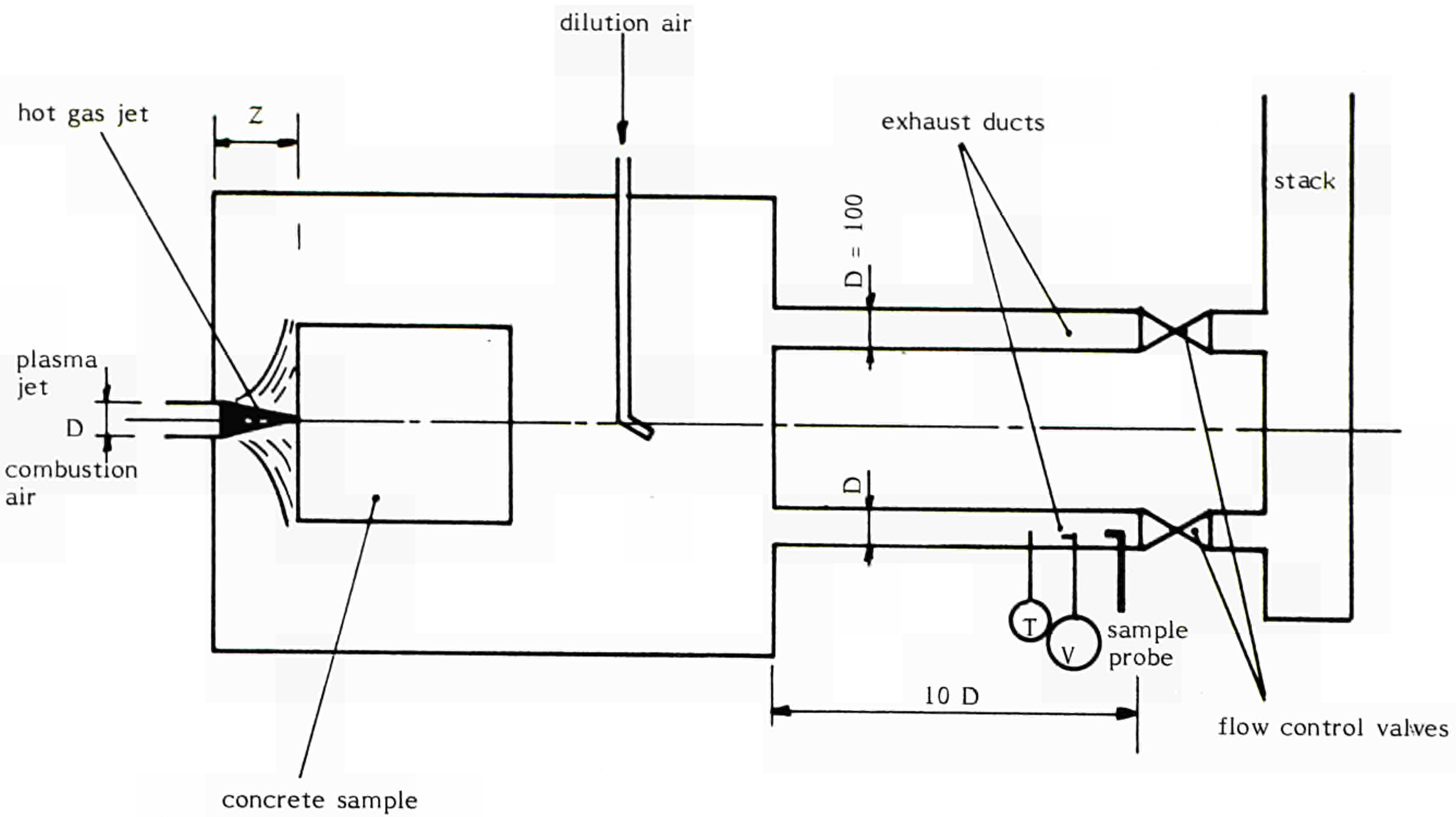


Figure 1 : Schematic diagram of the test facility

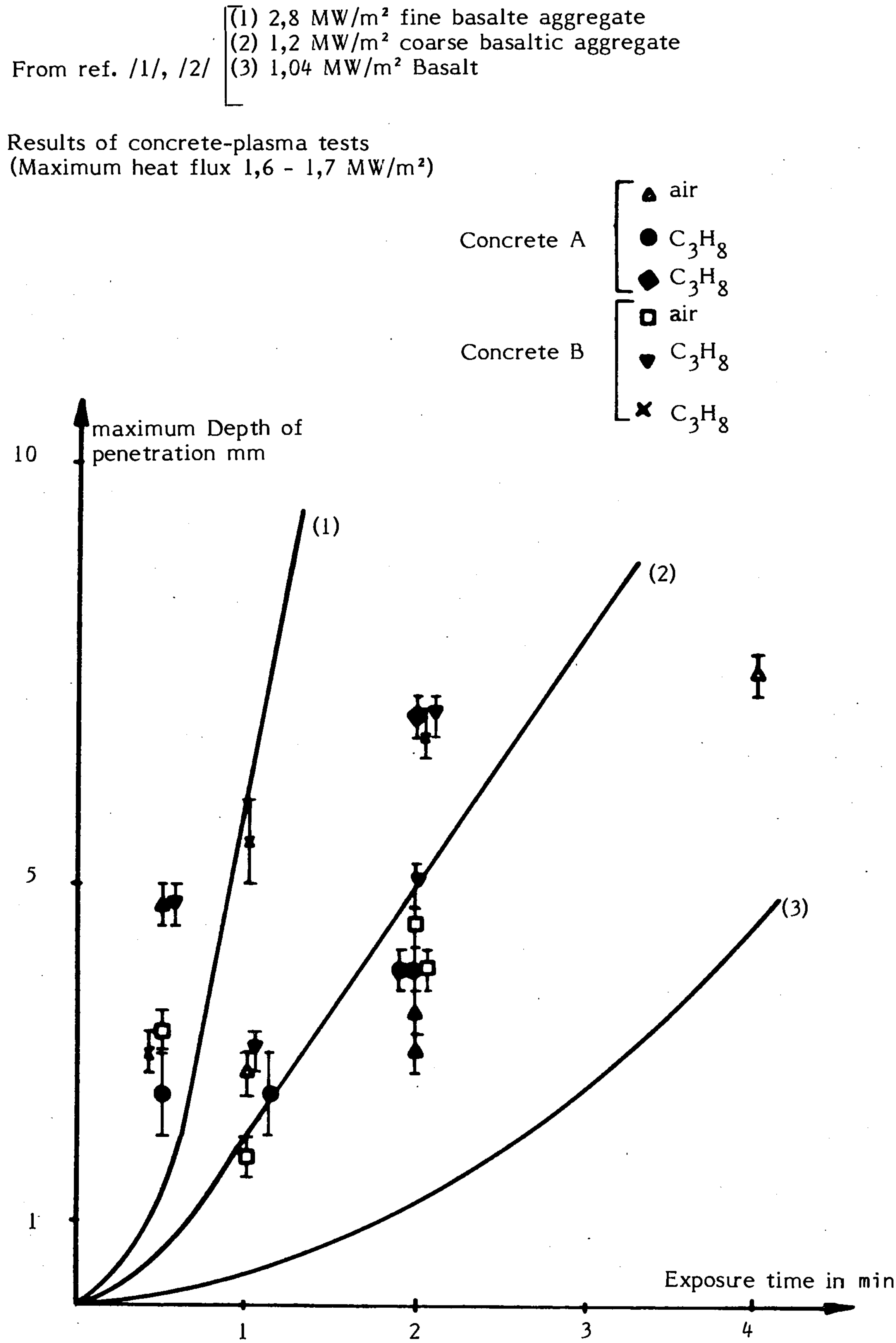


Figure 2: Erosion depth as a function of the thermal exposure time

Concrete A

Concrete B

Figure 3 : Concrete samples after heating
(exposure time : 120 s, propane plasma+air)

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

A prototype pool, 5 m deep, has been designed for testing a leak-proof electropolishing head which was designed and manufactured. This electropolishing head has been tested out of water in a closed circuit with excellent results.

The next step will be an automated process underwater.

Progress and Results

1. Development of the closed electropolishing system (B.1. and B.2.)

A closed circuit electropolishing equipment was developed, including a sucking pump, a filter, a vacuum tank and the electropolishing head (Figure 1). The electropolishing head was designed and manufactured with an electropolishing chamber and spillage chamber working under vacuum in order to avoid any leak from the head to the pool (Figure 2). This head was tested on a vertical stainless steel plate, without applied electrical current and using water instead of electrolyte, for checking the hydraulic circuit, eliminating leaks and securing the best sealing of the head.

Then the closed circuit was tested using electrolyte and applying current to observe and optimise the following features:

- flow distribution of electrolyte,
- temperature (possible degradation of electrolyte),
- proper work of spillage chamber,
- seals are not overstressed,
- complete rinsing of electrolyte chamber and control required,
- correct work even in welding points or surface liner imperfections (leaks evaluation),
- electric isolation of connection,
- flow from pool water (leak rate).

A batch electropolishing process was selected, including rinsing with demineralised water before the electropolishing head moves to a next polishing operation. The remote control of the process is centralised in a control panel.

Several electrolytes were tested in the laboratory. The best results were obtained with a mixture of phosphoric acid, sulphuric acid and water, reaching roughness values of Ra = 0.1 μm: the best polishing is obtained. With this electrolyte, the process is fast, and the problems of possible leaks have been solved.

In order to reproduce real conditions for cold tests of the electropolishing system, a simulated pool has been built (2.4x1.7x5 m). The wall of the pool for test is made of AISI 316 and the other walls are made of AISI 304 (Figure 3). At approximately every meter, there are sample connections and at the upper part of the wall to be tested, there is a 0.4x0.8 m metacrilate window to observe the electropolishing head working. Eight slight glasses are distributed along the opposite wall for observation purposes.

The system being developed to operate under water is shown in Figure 4.

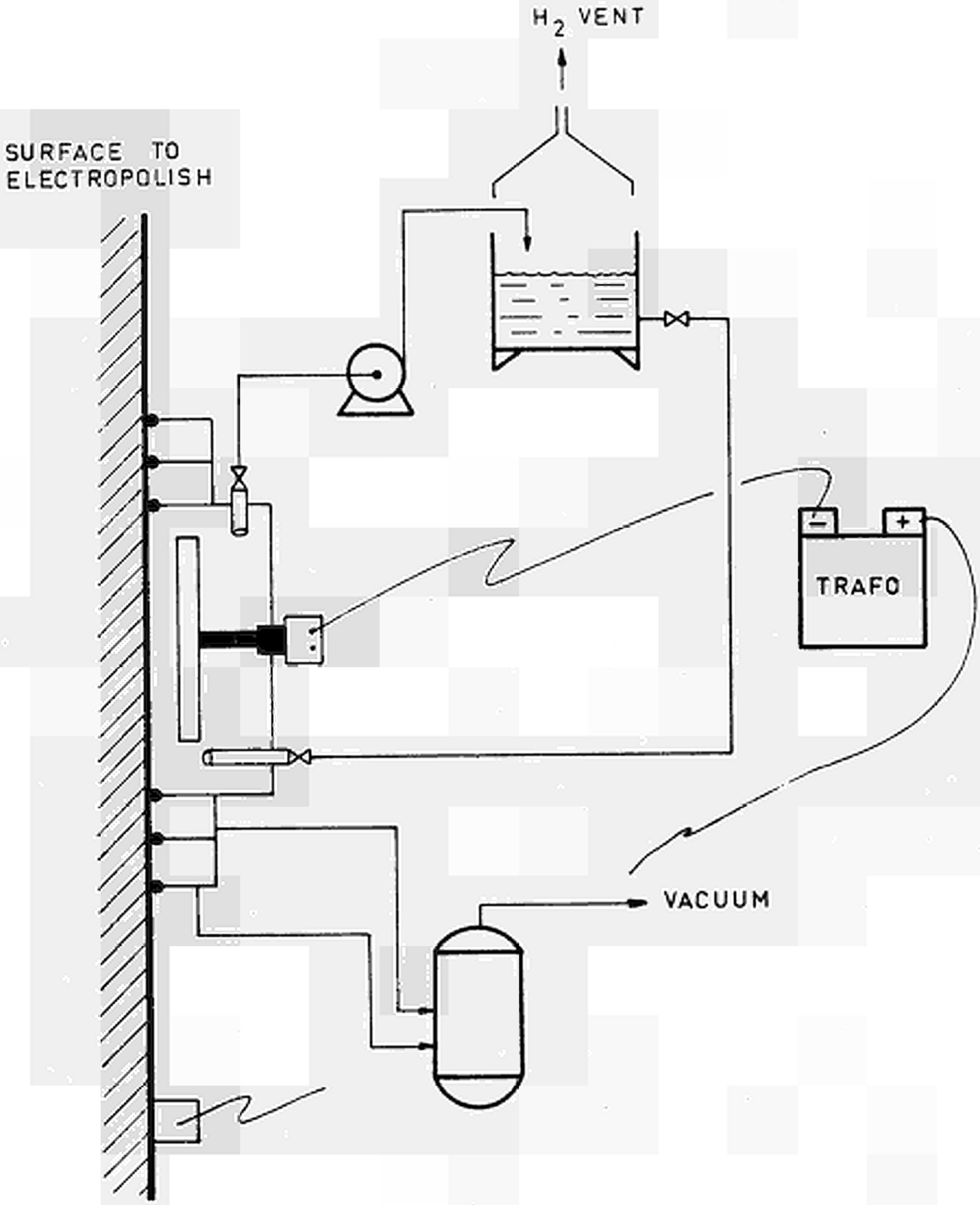


Figure 1 - Surface Electropolishing System

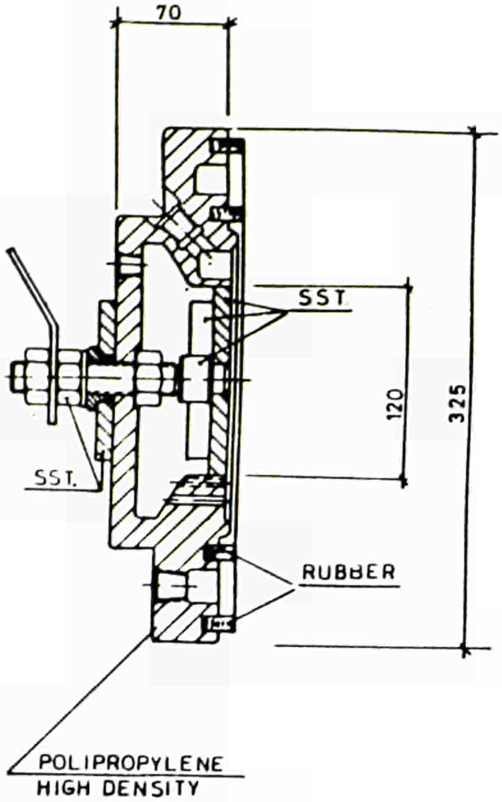


Figure 2 - Electropolishing Head

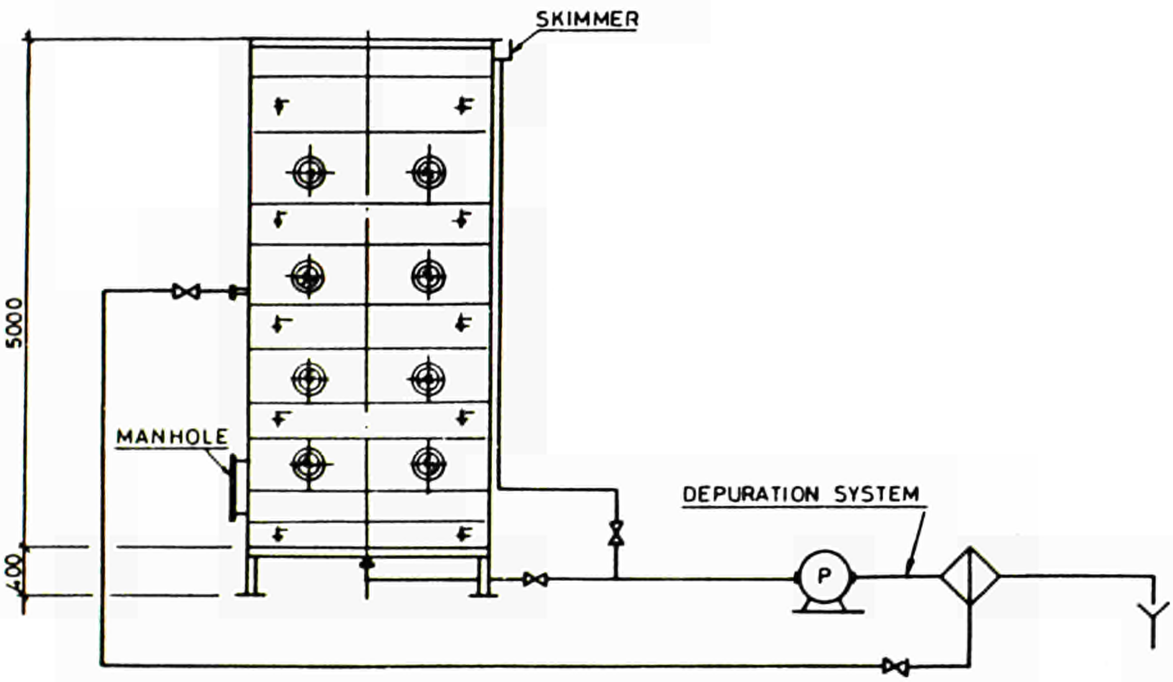


Figure 3 - Pool for underwater testing

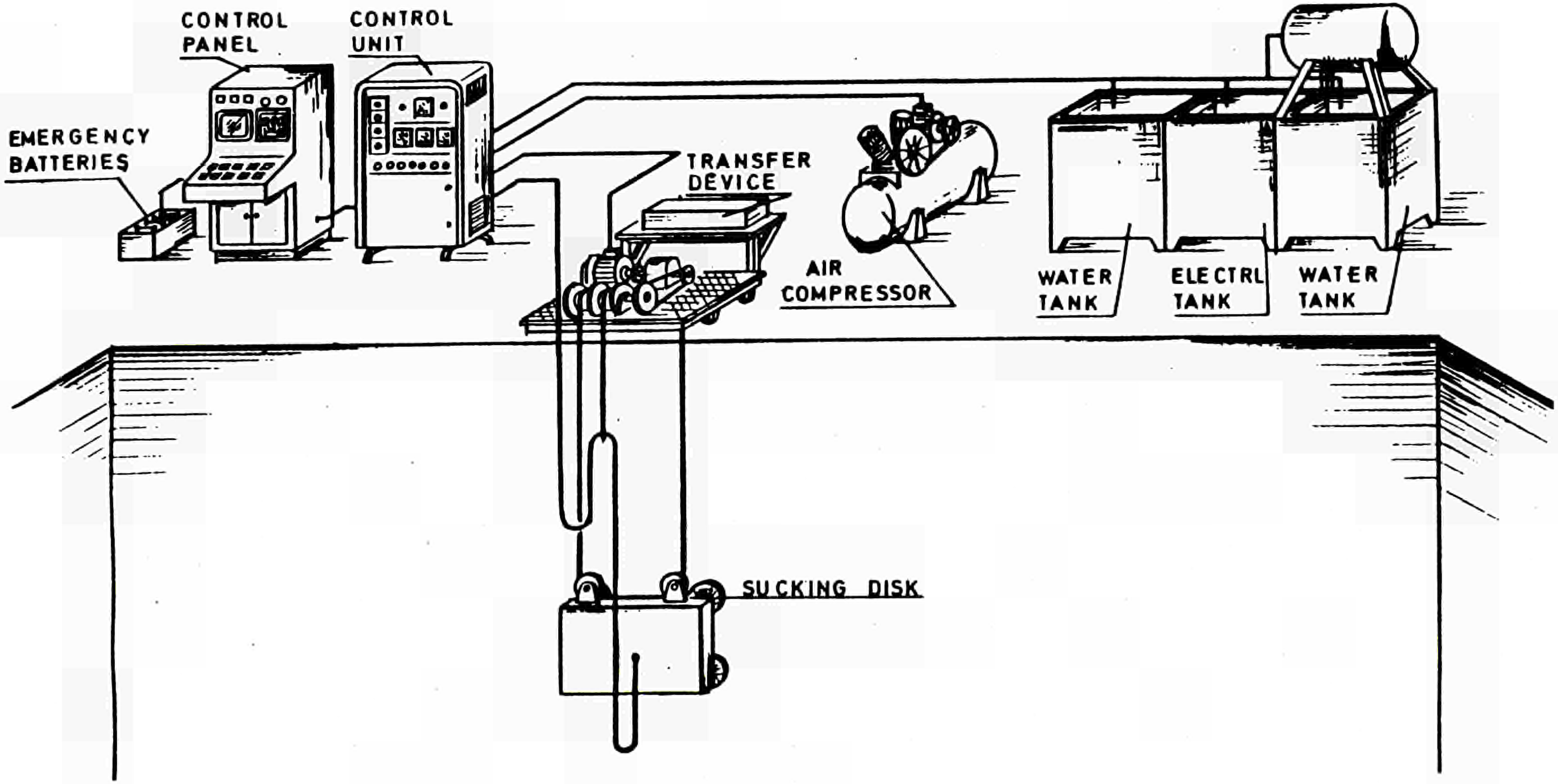


Figure 4 - Underwater Electropolishing System

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

A prototype equipment for vibratory decontamination with abrasive media was designed and manufactured to investigate the frequencies and abrasives that can be used. Non-contaminated specimens have been tested in this prototype to determine the abrasions and to design a decontamination system compatible with the vibratory device and the selected abrasives.

Progress and Results

1. Design and construction of a vibrodecon prototype (B.1. and B.2.)

A vibratory machine has been designed and constructed (Figure 1) including a rig-shaped tube, with an excentric motor as vibration equipment that provides a uniform vibratory finishing action, with additional helical movement as the parts and medium advance from starting point to a continuous process.

Also, a frequency regulator has been adapted to the motor in order to select frequencies between 0 and 3000 vibrations per minute. The best results, so far, were obtained with are 1500 vibrations per minute. The best choice of the medium to be used in a vibratory operation depends to a great extent on the results required, the physical characteristics of the parts processed, and the process equipment used.

The prototype includes supplementary equipment to wash, degrease and decontaminate the abrasives and the contaminated pieces by a solvent. The selected solvent system is a closed freon distillation unit (Figure 2). This system has the following main advantages:
- stable, reusable solvent,
- easy to decontaminate by distillation or filtration,
- inert solvent, does not react chemically.

The contaminated sludge is recovered in the distillation still and in the filtration unit.

The decontamination system has been designed, manufactured and installed to the vibratory equipment.

2. Selection of abrasives and testing on inactive samples (B.1. and B.3.)

Various types of abrasive were tested in the vibratory machine, in dry conditions on painted steel samples (50x30x6 mm and 100x45x10 mm).

Two groups of abrasives were selected:
- abrasives normally used in the blasting machine, such as metal particulate, sand and glass microbeads;
- abrasives of different sizes (3 to 15 mm) and geometries (Figure 3), such as ceramics and aluminium oxide.

The best results were obtained with abrasives of the second group, in particular with ceramic abrasives of angular form. After 30 minutes of operation at 900 vibrations per minute, the paint film completely disappears.

Figure 1: Vibratory machine

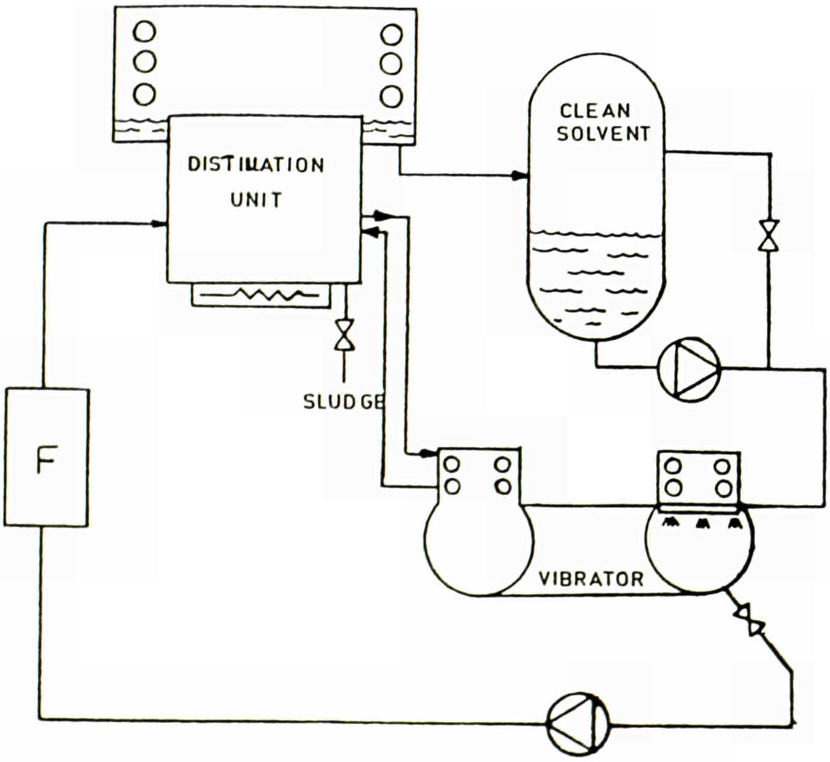


Figure 2: Vibratory decontamination system

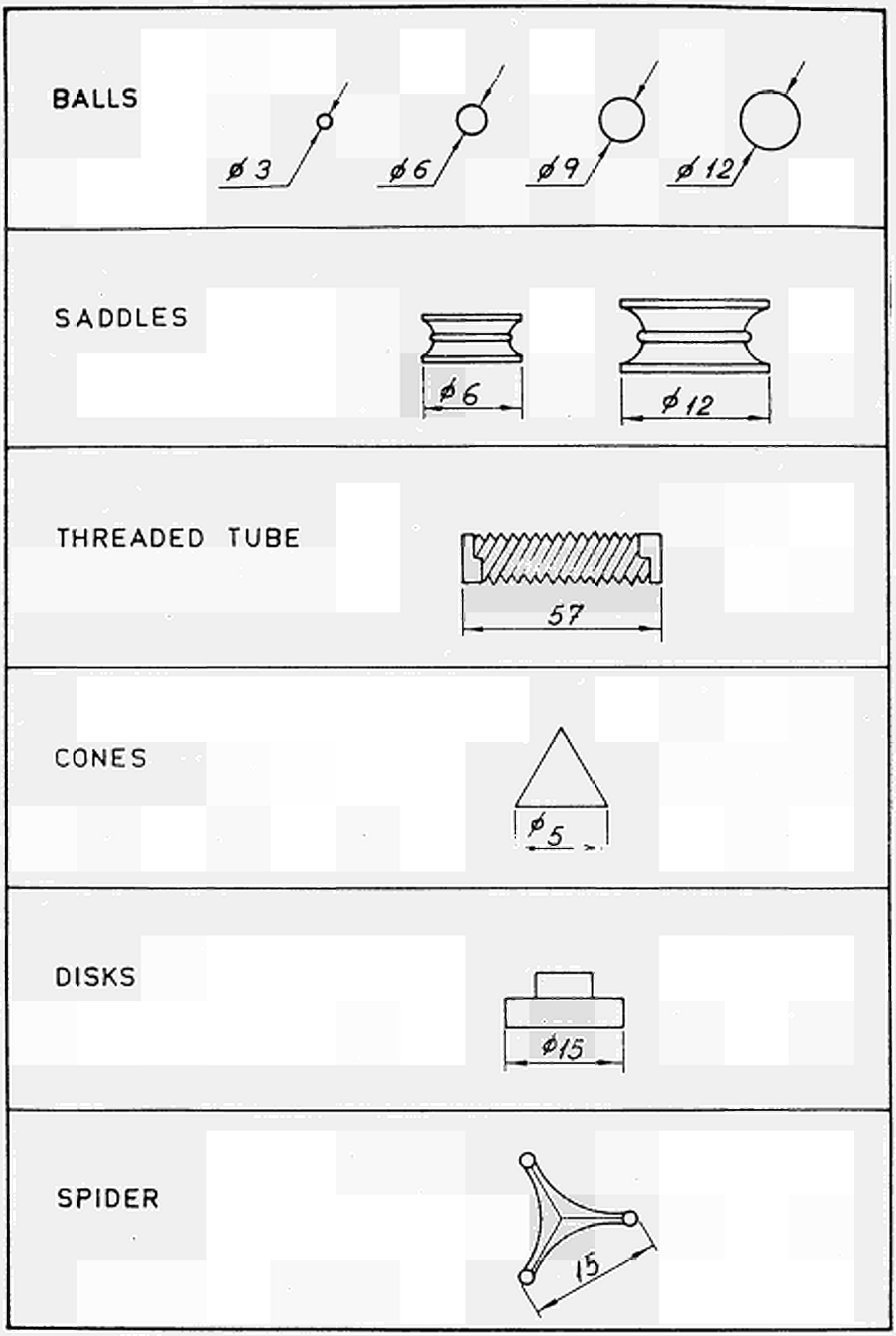


Figure 3: Abrasives of different sizes and geometries

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

Sixteen research contracts relating to Project N°3 were being executed in 1987, including three new contracts concluded in 1987. The execution of one contract was already completed in 1986. Besides, three contracts were still at the stage of negotiation at the end of the year.

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

Work has continued in the ventilation/filtration rig in the laboratory. Following earlier trials on the electrostatic precipitator (ESP), as described in the 1986 Annual Progress Report, further experiments were carried out on the vibratory cleaning system. The experiments showed that vibrating the plates gave about 90 per cent restoration of the ESP efficiency.

Following the ESP experiments, the oxypropane cutting rig was replaced by a 400W carbon-dioxide laser. The objectives of the laser experiments are (a) to characterise the fume generated by cutting glove box and active cell components, and (b) to test the effectiveness of various prefilter devices. The oxypropane cutting rig has been installed in the full-scale HERO facility. A prototype of a high efficiency particulate air filter has been built by colleagues at Harwell. The filter medium is cleaned by pulses of compressed air in a manner similar to the cartridge filter reported in the 1986 Annual Report of the Commission /1/.

Studies continue into how to remove the radioactive burden from an ESP. An ESP cabinet has been purchased and is being modified to investigate a 'boxing-out' technique to remove the contaminated ESP plates. The study has concentrated on keeping operator dose rates and airborne active dust to a minimum in the working area.

Progress and Results

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

Initial results have been obtained of the aerosol generated by cutting 3mm thick stainless steel sheets with a carbon dioxide laser. The particulate size range was measured using a TSI differential mobility particle sizer. Approximately 60 per cent of the particles were in the range 0.04 µm - 0.1 µm in diameter. A range of 0.886 µm to 0.017 µm was recorded.

2. Construction of a filtering rig (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 laser remains stationary while the target material is moved on an indexing frame. In the present arrangement the ventilation/filtration rig has an ESP prefilter to collect the fume generated by cutting 3mm thick stainless steel sheets. When the tests have been concluded on the ESP a high gradient magnetic separation prefilter (HGMS) will be installed and tested.

Results so far have been encouraging. The ESP worked at an efficiency of 85 to 98 per cent during the experiment. This corresponds with results obtained using the oxypropane torch (85 to 99 percent efficiency).

The remote indexing frame and oxypropane torch has been installed in the full size HERO Development facility (flow throughput 15,000 m^3/hr). Due to an equipment failure the planned trials with a prototype high efficiency particulate air (HEPA) filter with a reverse air-pulse cleaning system have been delayed until April/May 1988.

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

Preliminary experiments were conducted on the cleaning system of the ESP installed in the cutting and ventilation/filtration facility. Following these trials and the experiments conducted with CEA Saclay in the HERO facility described in the joint UKAEA/CEA report/2/ it was decided to investigate the ESP cleaning system further. Fume was generated for a total of six hours over a two day period. Mild steel plate 80mm thick was cut using the oxypropane torch with powder injection to increase fume production (Figure 1). The ESP prefilter unit is capable of handling a throughput of 5000 m^3/hr split evenly between four compartments each containing a coarse prefilter, ioniser section, collector plate assembly and after filter (Figure 2). The cleaning system consists of eight pneumatic vibrators, one connected to each ioniser and collector plate. The vibrators are operated by compressed air reducing from 0.6 MN/m^2 to zero over 90 seconds. This procedure shakes dust off the cell plates. The dust falls into a hopper below the ESP.

A total light scattering photometer measured the efficiency of the ESP. The ESP operated at a maximum of greater than 99 per cent and minimum of 85 per cent efficiency. Figure 3 shows the change in ESP efficiency with time. The erratic readings are due to the fluctuations in fume generation over short periods of time. This in turn affects the discharge voltages on the ioniser and collector plates and hence influences the efficiency of the ESP. The total weight of dust removed was 1.5kg. Table 1 shows the mass of dust collected in the hopper after each cleaning operation. It was concluded that the vibratory cleaning system was successful in restoring the efficiency of the ESP and hence increases the length of time between manual cleaning of the plates.

The voltages of the ESP plates were measured to compare with the efficiency (see Figure 4). It was concluded that the voltage measurements related to the efficiency but were erratic due to localised dust bridging across the plates and fluctuating fume generation. Consequently it is recommended that a voltage chart recorder should be used to indicate the voltage trend as a pointer as to when the ESP cells require cleaning.

Work continues to develop a technique for the safe removal of the contaminated plates for an ESP. An ESP cabinet has been purchased for this purpose. The technique of removing the plates by 'bagging-out' in polythene has proved unsuitable due to the awkward shape and sharp edges of the components to be handled. Consequently, efforts have concentrated on designing a procedure to 'box-out' the ESP components (Figure 5). Detailed drawings have been prepared for the modifications required on the ESP cabinet. The dust burden collected in the hopper is ducted away by vacuum This vacuum is also used to provide a depression in the ESP cabinet while removing the plates. In effect the ESP cabinet becomes a glove box with a continuous depression to prevent the escape to the HEPA filter bank or to atmosphere of disturbed dust while the main ventilation bypasses the ESP.

References

/1/ The Community's R & D Programme on Decommissioning Nuclear Installations, Second Annual Progress Report (1986), pp. 43-49, EUR 11112.

/2/ 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 of dust collected after cleaning of the ESP

No. of times vibrated	Mass of dust collected (g)	Cutting time (min)	Rate of dust collection (at ESP) (g/min)
1	142.3	13	10.9
1	542.7	53	10.2
2	364.9	29	12.6
2	68.7*	59	1.2
2	302.9	36	8.4

*cutting without powder injection

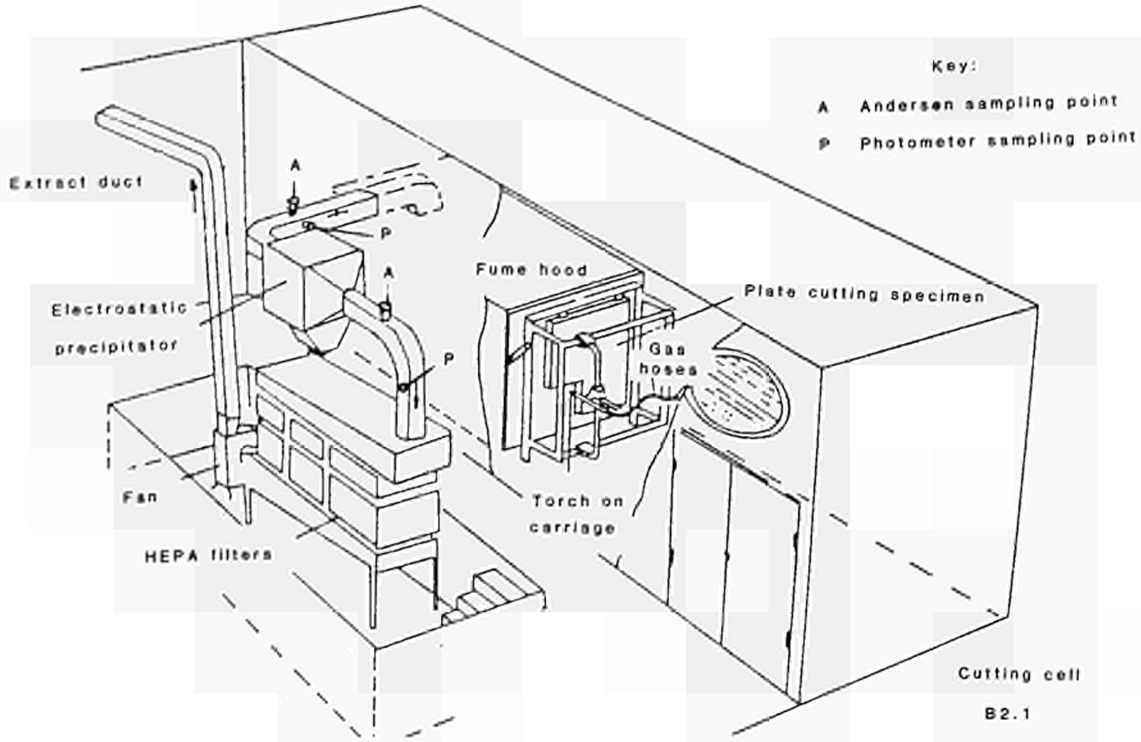


Figure 1: Flame cutting and fume generation rig

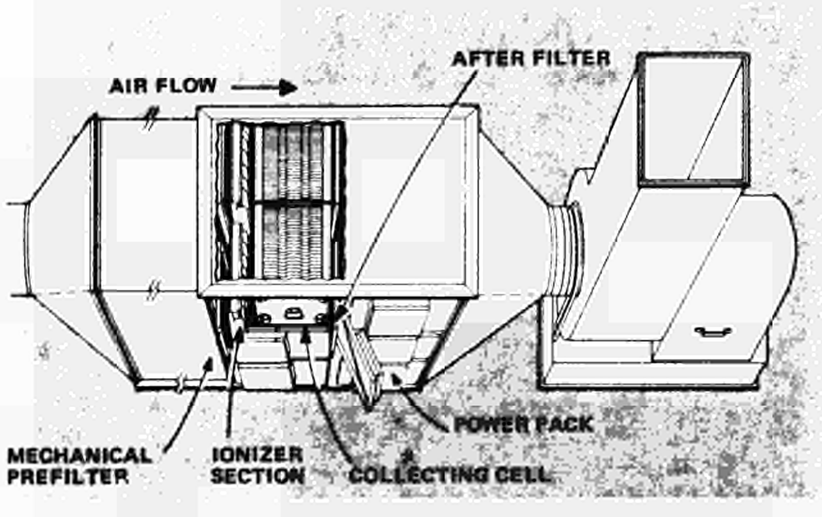


Figure 2. Key components of a modular ESP

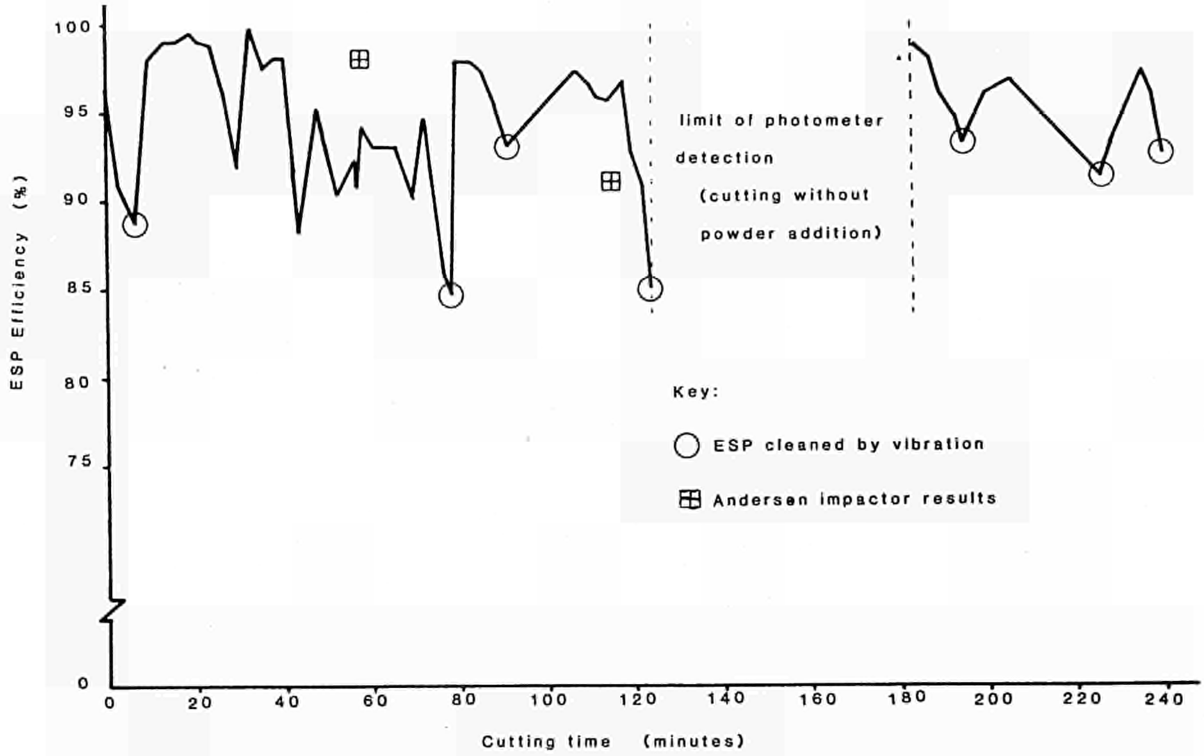


Figure 3. Efficiency of the electrostatic precipitator with time (photometer results)

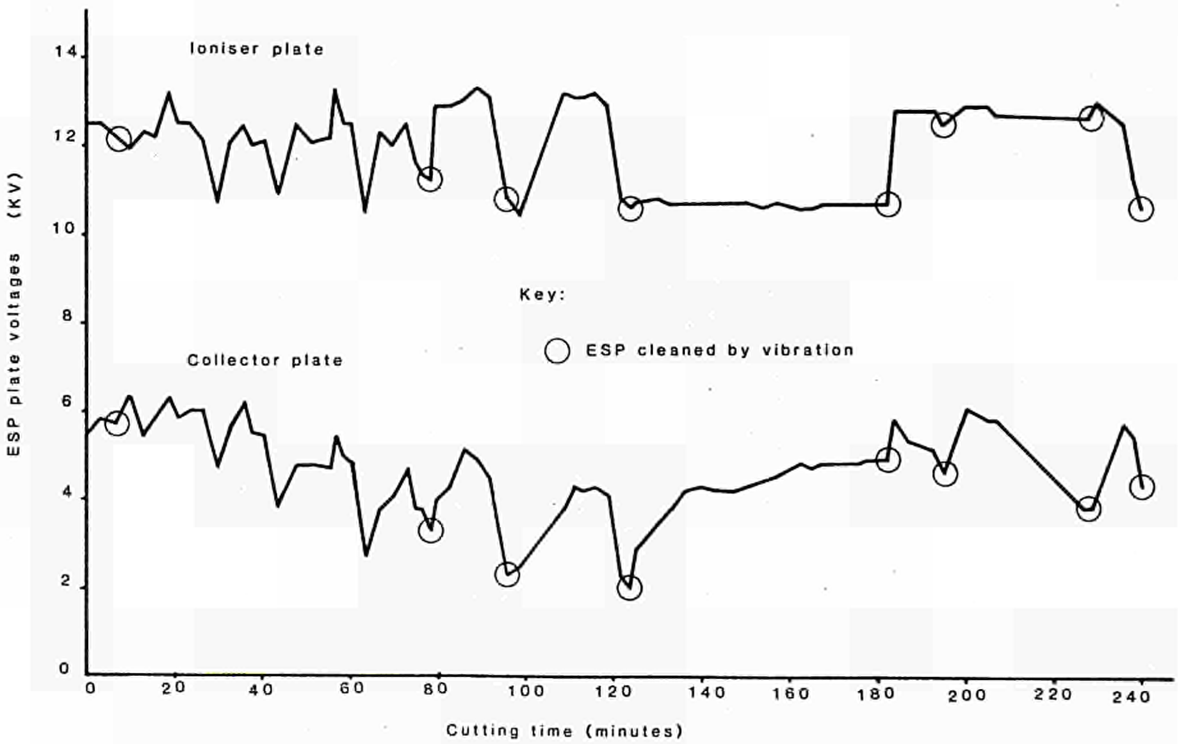


Figure 4. Changes in the ESP plate voltages with dust build-up

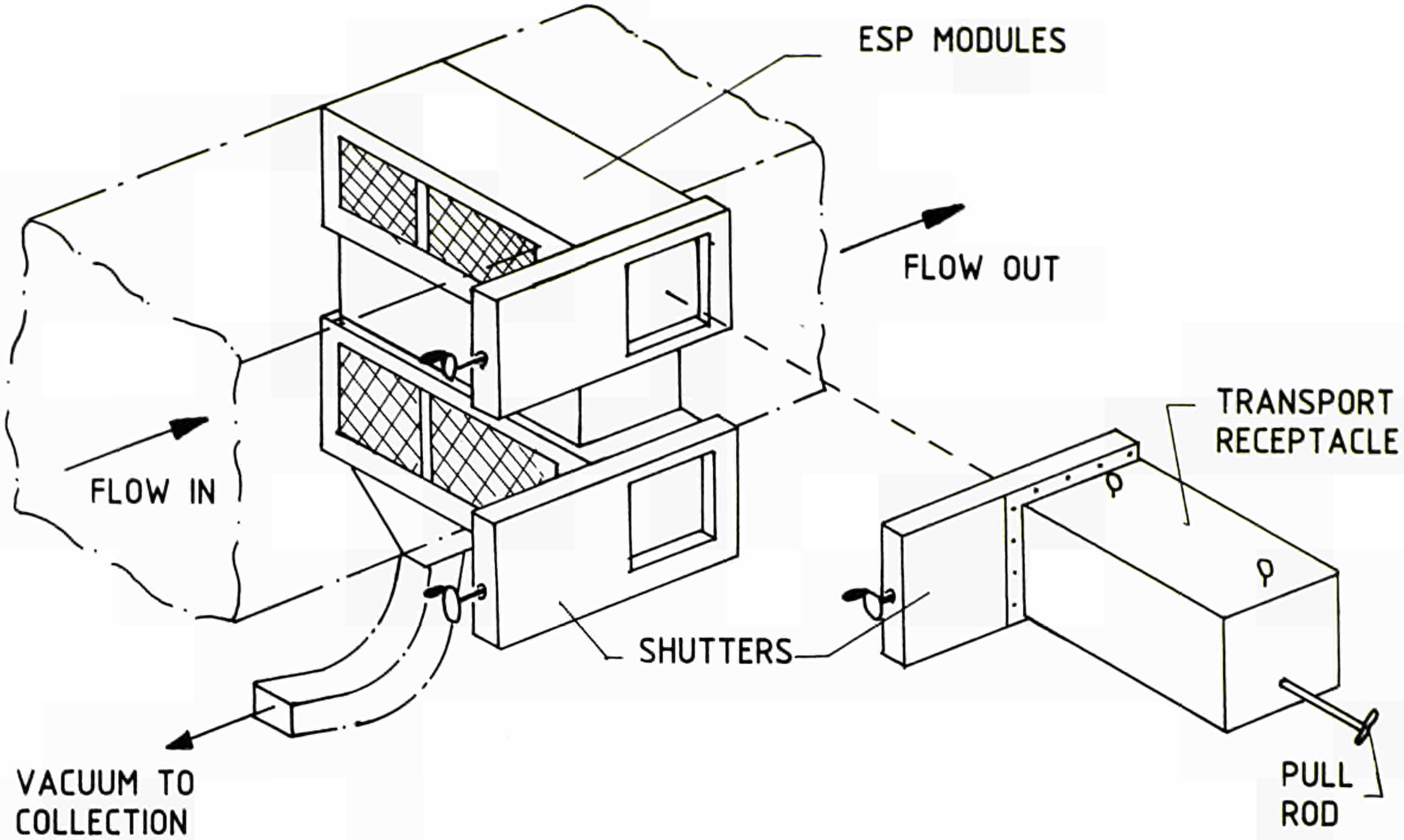


FIGURE 5: SCHEMATIC OF AN ESP FOR REMOTE HANDLING

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 - June 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

The experiments made emphasized the advantages to have a captation device at the aerosol generating source (working step B2) and to put optimized prefilters upstream of the HEPA filters of the nuclear facilities (working step B4).

An aerosol captation device at the emission source has been thus designed and some tests showed its interest by the containment and cleaning functions associated with a possible control function.

The declogging of an electrostatic prefilter by pneumatic vibrations is very dependent on several parameters.

The influence of water depth on secondary emissions produced by underwater plasma torch cutting (working step B6) has been studied in co-operation with Heriot-Watt University (contract n° FI 1D-0008).

Experiments in co-operation with CEA/CEN Cadarache (contract n° FI 1D-0037) have been made about the underwater plasma torch cutting of radioactive materials coming from the dismantling of nuclear facilities, it was thus actual cutting in nuclear environment (working step B5).

Progress and Results

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

The experiments at Windscale in co-operation with UKAEA (contract n°FI 1D-0006) summarized in the second annual progress report (year 1986) have been described in detail in a common report /1/. These experiments showed that the ESP can be an adequate prefilter for the dismantling of the WAGR reactor but some studies are necessary to know the best procedure for the declogging of such a prefilter with radioactive dust inside.

The tests on the declogging of an ESP by pneumatic vibrations emphasized that the declogging efficiency was very dependent on the nature of the particles, the deposited mass and the process of generation.

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

A device (for the captation of the particles at the aerosol generating source) has been conceived from the following concepts :

- captation at the source of the particles produced by cutting tools (i.e. surrounding as much as possible the working area, creating sufficient captation speed, placing possibly the device such as to use the natural convection phenomena and as close as possible to the generating source),
- cleaning of the secondary emissions inside the cell with reduction of waste volume,
- ventilation shutdown (air supply), the internal ventilation of the device maintaining the pressure drop of the cell by air exhaust flow equal to cell leak flow and holding the containment of the particles produced by cutting tools.

The tests put in evidence that the use of such a device with optimized conditions could allow :

- the reduction of waste volume by trapping radioactive particles at the emission source thus increasing the time life of HEPA filters,
- the reduction of radioactive deposit on the cell walls and inside the exhaust ventilation ducts,
- the diminution of decontamination and maintenance operations,
- the decrease of exposure time of workers and thus the collective dose.

3. Underwater plasma torch cutting

3.1. Experiments in cooperation with Heriot-Watt University (Edinburgh) (working step B6)

The experiments have been projected in order to :

- characterize the aerosols evolved as a function of water depth
- establish a balance for the distribution of all secondary solid wastes
- determine the concentrations and quantities of gas evolved.

The aerosol concentration is decreasing very rapidly with water depth until 3.55 m, the slope of the decrease being less strong between 3.55 m and 9.55 m (figure 1).

The water acts like a filter.

The mass mean diameter is located between 0.16 µm and 0.23 µm and the results obtained don't allow to see a sensitive trend with water depth (table 1).

Furthermore, there is an enrichment of chromium in aerosols (more volatile than Fe and Ni).

NO concentrations decrease with water depth whereas we note an increase in the concentration of H_2 especially at 9.55 m water depth for which previous measurements suggest that this increase is not due only to the stronger current used (figures 2 and 3).

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

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

3.2. Experiments in cooperation with CEA/CEN Cadarache (B.5.)

Several radioactive plates coming from the dismantling of two nuclear facilities (TRITON and RAPSODIE) have been cut by underwater plasma torch at Pegase (CEN Cadarache) with selected captation and prefiltration devices for the aerosol secondary emissions.

The tests on underwater plasma torch cutting of contaminated and activated plates (∿ 0.7 Bq/g and 1900 Bq/g) emphasize the following points :

- the aerosol concentration in the exhaust duct (flow : 1000 m^3/h) varies from 1 to 4 mg/m^3, the volumic activity of ^{137}Cs varying from 380 to 900 Bq/m^3
- the mass mean diameter of aerosols is about 0.2 µm with σg (standard geometric deviation) = 2.5. The figure 4 shows the size distribution in activity (^{137}Cs)
- the sedimented drosses and the aerosols represent respectively about 85% and 0.1% of the removed mass.

Furthermore, with the radioactive balance, we note that :

- the difference of the behaviour of the various radioisotopes (^{137}Cs ^{60}Co, ^{54}Mn) with an enrichment of the caesium in the aerosols and an possible ^{137}Cs decontamination of the cutting edges (width = 5 mm), the caesium being more volatile that the other radioelements.

A captation at the aerosol generating source was made with the inlet tube placed so as to create a good mixture in the cell surrounding the tank and with the exhaust tube placed just above the water in order to collect the aerosols as close as possible of the emission source. This installation allows to avoid the dissemination of contamination inside the cell and to obtain air change rates just above the tank of 1000 h^{-1}.

The ESP put in the exhaust duct had an efficiency superior to 90% for the radioisotopes involved, mainly carried by submicronic particles.

References

/1/ WILSON, K., BISHOP, A. (UKAEA)
PILOT, G., VENDEL, J. and LE GARRERES, I. (CEA)
Characterization and pre-filtration techniques for aerosols from oxy-propane torch cutting in nuclear facility decommissioning UKAEA Northern Division. Report ND-R-1533(W) (August 1987)

Table I: Mass mean diameter (μm) of aerosols produced by the cutting of stainless steel plate (e = 38 mm) by underwater plasma torch. Influence of water depth

Measurements	Water depth (m)			
	0.08	1.06	3.55	9.55
Total mass	0.198	0.158	0.191	0.200
Fe	0.229	0.161	0.187	0.183

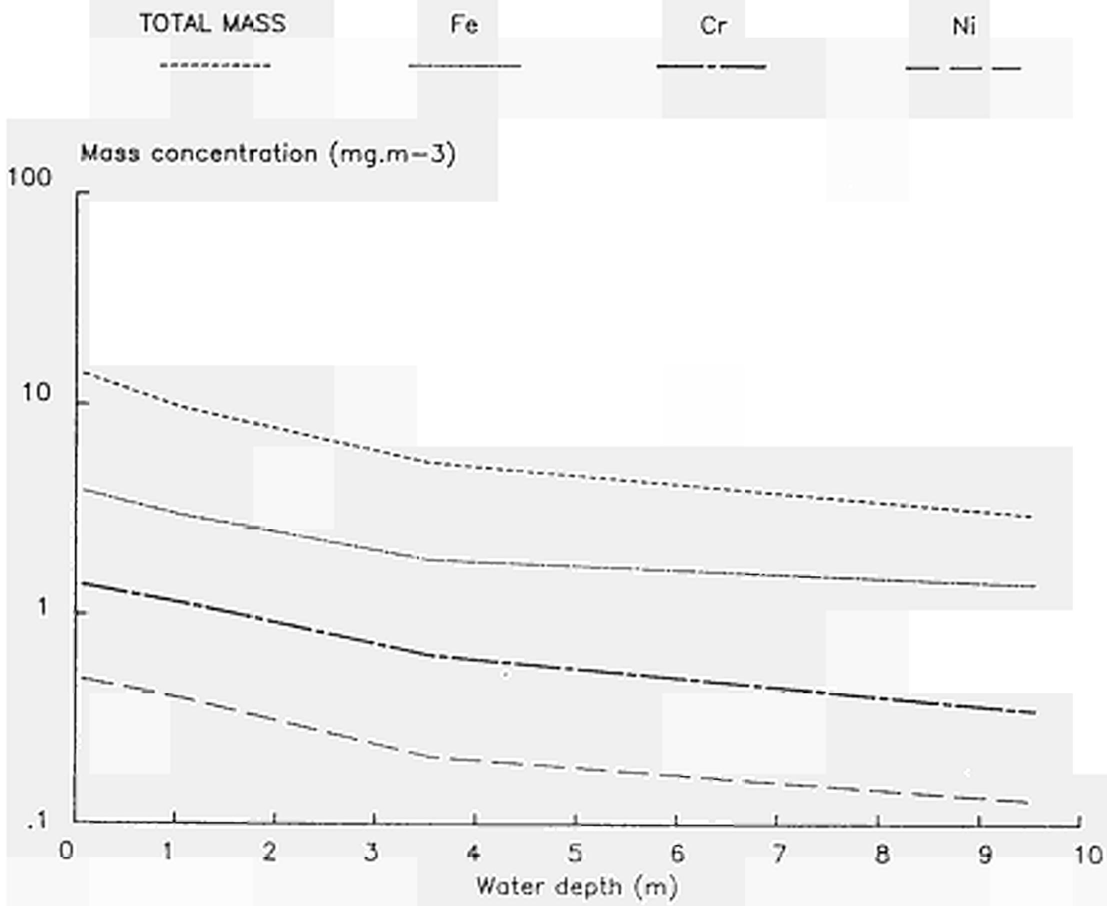


Figure 1: Influence of the water depth on aerosol mass concentration for underwater plasma torch cutting of stainless steel plate (e = 38 mm)

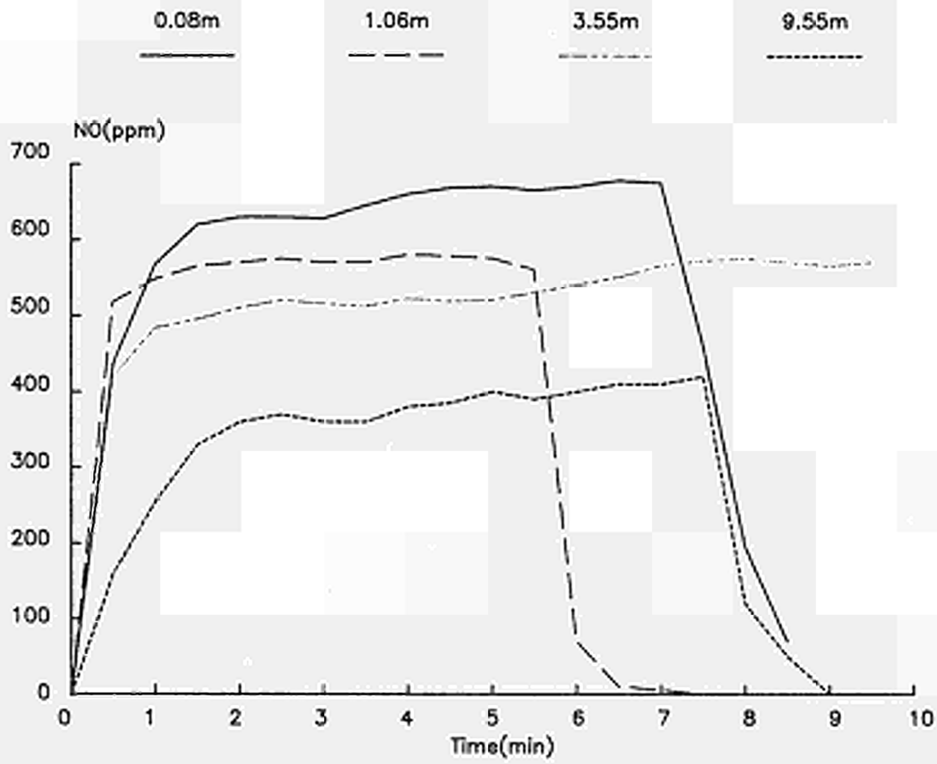


Figure 2: NO concentrations in the exhaust duct (dilution : 22)
Influence of water depth for underwater plasma torch cutting of stainless steel plate (e = 38 mm)

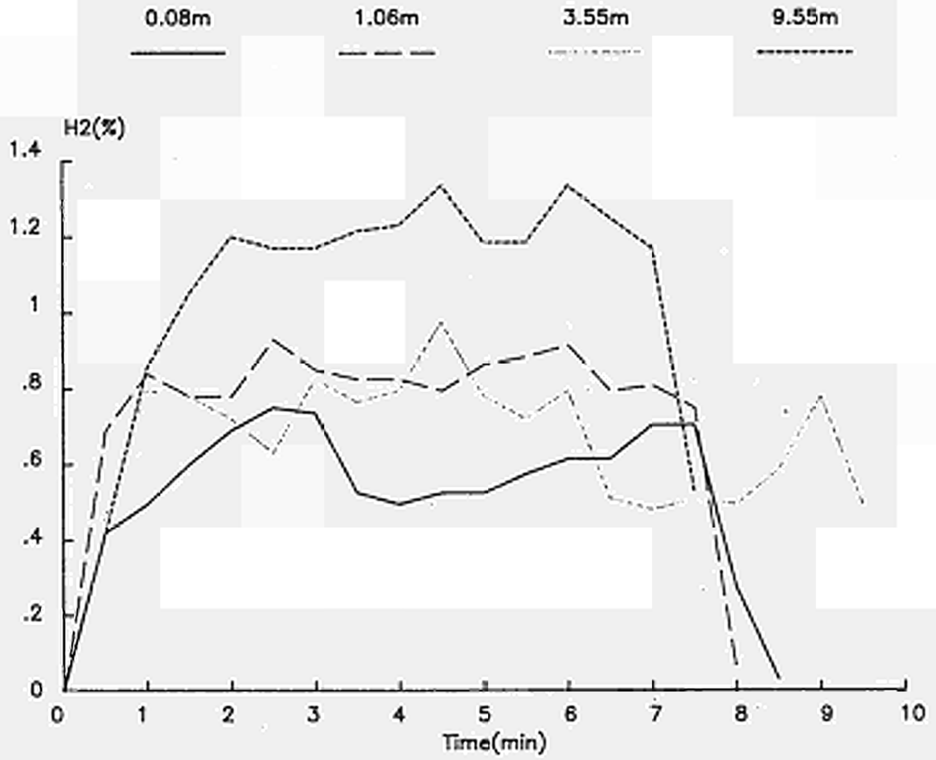


Figure 3: H_2 concentrations in the exhaust duct (dilution : 22)
Influence of underwater plasma torch cutting of stainless steel plate (e = 38 mm)

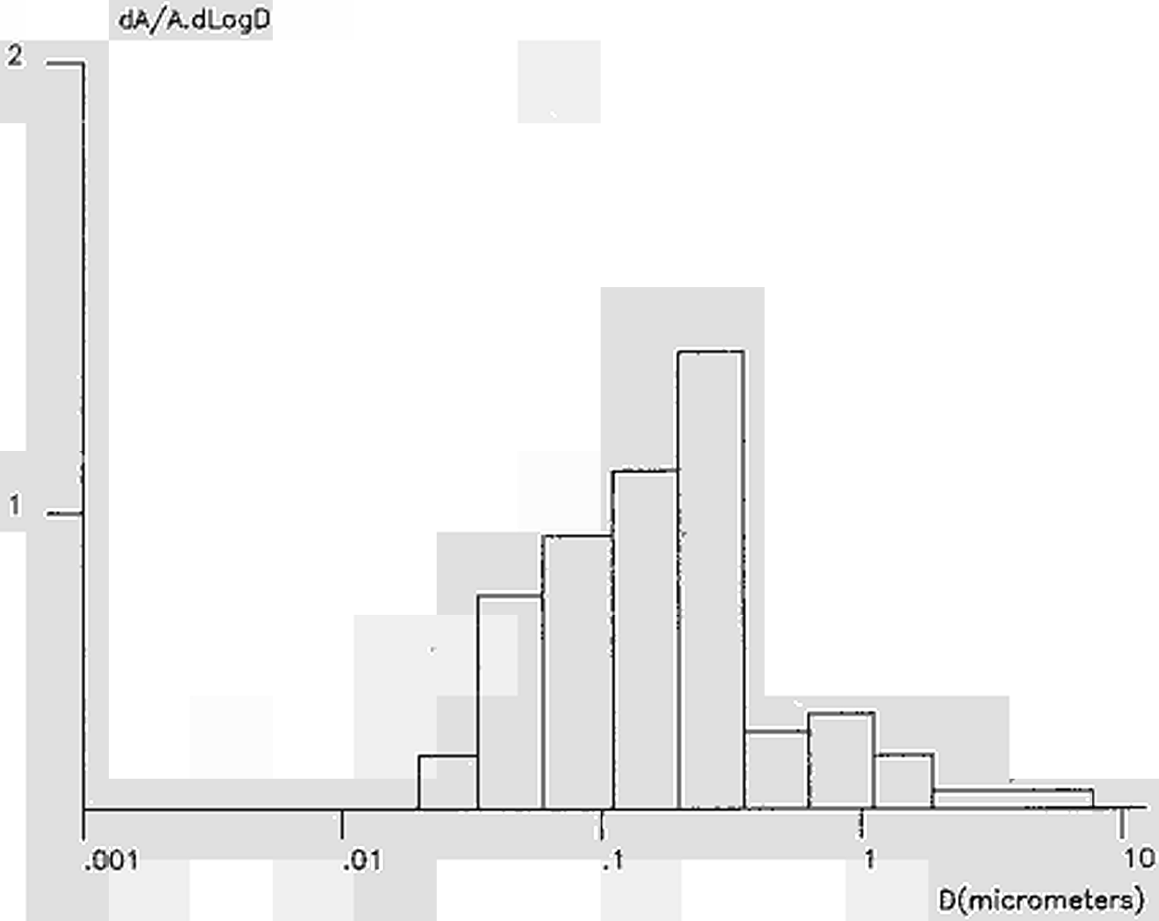


Figure 4: Radioactivity (^{137}Cs) distribution in size range of aerosols produced by underwater plasma torch cutting of RAPSODIE plates

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.

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.

C. Progress of Work and Obtained Results

Summary

Underwater plasma cutting studies have been done in the completed 0.6 m diameter by 10 m deep test vessel. Increasing current requirement and dross attachment with water depth are observed as in the earlier hyperbaric vessel runs. Rate data for filtration of fine suspended particles from water have been correlated in terms of a resistance parameter and mass concentration of particles.

A joint study with the SPIN Group of CEA Saclay on the effect of water depth on secondary emissions has been carried out. Fume particle characterization and gas analytical instrumentation from Saclay were used on the deep water test vessel. Fume particle emissions decreased with increase in water depth. The yield of fine suspended particles in the water increased with water depth but to a greater extent than expected from changes in the fume particle concentration. The mass of these water borne particles relative to their metal content suggests that they are well oxidized. Reduction in water pH was observed at all depths.

Progress and Results

1. Attached and larger dispersed dross particles (B.2, B.3, B.8, B.9)

Cuts have been done in 304 stainless steel at different water depths in the completed 0.6 m diameter x 10 m deep vessel.

As shown in Table I, attached dross increases with water depth for otherwise constant conditions. A similar trend was seen previously when varying the pressure in the hyperbaric vessel. For the deepest water (9.55 m) cut, the current and gas flow had to be increased to guarantee cutting. These changes reduced the attached dross from that at 3.55 m depth. Increasing only the gas flow rate reduced attachment at minimum water depth.

Dispersed dross shows a corresponding decrease with depth at otherwise constant conditions. It will be noted (Table I) that the weight of dispersed dross always exceeds the weight loss of the specimen. This may be due to oxidation of the dispersed material or perhaps to inclusion of water in the dispersed dross globules. There is quantitative evidence later of oxidation or other mass enhancement of the fine suspended particles. The actual weight loss in the specimen is greater here than in the previous hyperbaric vessel runs due to the relatively low cut speed used here. Size distribution of the dispersed dross particles (Figure 1) as measured by sieving shows no definite trend with depth. The apparent wide diversity in size distributions can arise from variations in the number of the relatively few large (> 20 mm) pieces of detached dross.

2. Filtration Characteristics of Particles Suspended in Water (B.7, B.9)

Further measurements have been made on filtration rates of the dilute suspensions of fine particles remaining in the water after the larger dispersed dross has settled out. These further measurements have mainly been made on the residual water in the deep test vessel. In a few runs the total water was filtered but this was rather time consuming with the available 293 mm diameter 0.45 micron membrane filter. The data presented were obtained from filtration of 25 or 50 litre samples. Data of the type shown previously /1/ have been expressed in terms of a combined cake resistance x specific volume term. Rate measurements were made at a constant pressure differential of 0.85 atmos. As for the earlier hyperbaric vessel runs /1/, no direct trend with cutting variables or water depth was evident. The resistance parameter depends rather on the mass concentration of suspended particles as shown in Figure 2. Whilst the data from the hyperbaric and deep vessels overlap there is evidence of the resistance parameter being more dependent on concentration for the deep vessel. This is most likely due to the concentrations and hence filter cake thicknesses being generally lower with the deep vessel. At a given concentration there is an uncertainty factor of about two in the value of the resistance parameter due to scatter. Some of that scatter may arise from the variety of cutting conditions covered e.g. different thicknesses, currents and speeds.

3. Ultrafine fume particles in effluent gas (B.3, B.9)

Detailed studies of the yields and characteristics of ultrafine-fume or aerosol particles have been made in a joint study with the SPIN Research Group of C.E.A. Saclay. Particle measuring and gas analysis instrumentation and sampling equipment developed by SPIN were linked to the 10 m deep test vessel as shown diagrammatically in Figure 3.

The sampling equipment and instrumentation were connected into the vessel at different heights in turn to allow sampling near the water level at all depths studied. All cuts were done in deionised water. Dry filtered nitrogen gas was added as diluent to avoid condensation in the sampling system and to avoid any possibility of explosive mixtures being formed by hydrogen in the effluent gas. To maximise the time available for particle size analysis with the DMPS instrument, a relatively low mean cutting speed, 46 mm/min, was used. Other conditions and associated results are given in the last four runs in Table 1.

The effect of water depth on the mass of aerosol in the effluent gas leaving the water is shown in Figure 4. The highest aerosol yield at 0.08 m represents about 0.026% of the total weight loss of the workpiece. Increasing the water depth from 0.08 to 9.55 m reduces the aerosol mass per unit cut length by a factor of about 5. The effectiveness of the water in terms of absolute mass of aerosol removed obviously decreases with depth. Details of particle size distributions and other particle and gas measurements made by the SPIN Group are given in the annual report for contract F11D-0007 /1/.

4. Suspended fine particles in water (B.2, B.9)

Corresponding data on the mass of fine particles remaining suspended in the water at 5-10 minutes after cutting are shown in Figure 5. The range of values at a given water depth represents the difference between water around the cutting zone and water located higher up the column. As expected, the mass of suspended particles increases with water depth. However, that increase is much greater than the corresponding fall in mass of aerosol particles. The reason for the comparatively large variation in suspended particles with depth is not clear. The higher current in the 9.55 m run will inevitably contribute. Judging from the previous runs in the hyperbaric vessel, increasing pressure does not itself promote fine suspended particles. Levels of suspended particles are higher than in those earlier runs. The slower cut speed here probably contributes to that difference.

An interesting feature of the suspended particles is that their mass is two to three times greater than their combined elemental metal content (Fe + Cr + Ni). This implies that the suspended particles comprise predominantly of oxides or other compounds rather than metals. Elemental analysis was done by the C.E.A.

5. Water quality (B.9)

Cutting was done in deionised water with an initial conductivity of about 5 uS/cm and pH about 7.4. At all depths (Table II) there was a measurable decrease in pH, presumably due to acid formation by absorption of some of the oxides of nitrogen formed in cutting. Total amount absorbed apparently increases with water depth.

References

/1/ WALDIE, B. Second Annual Progress Report (year 1986), p 58

TABLE I: Attached and Dispersed Dross per metre cut length at different water depths

Gas flow l/min	Current amps	Water Depth m	Weight Loss Kg/m	Dispersed Dross Kg/m	Attached Dross Kg/m	Attached Dross % of wt. loss
38	180	0.08	1.628	1.709	0.070	4.3
47	200	0.08	1.559	1.614	0.047	3.0
47	195	1.06	1.512	1.537	0.049	3.2
47	200	3.55	1.451	1.519	0.150	10.3
63	230	9.55	1.684	1.850	0.120	7.1

Metal 304 stainless steel Speed 46 mm/min ± 10%
Thickness 37 mm Gas Composition 60% Ar/ 40% N_2

TABLE II: Change in pH of water over cuts

Water depth m	pH Before	pH After
0.08	7.4	4.6 (3.2 near cut)
0.08	7.5	4.3
1.06	7.5	4.6
3.55	7.3	4.2
9.55	7.4	4.2

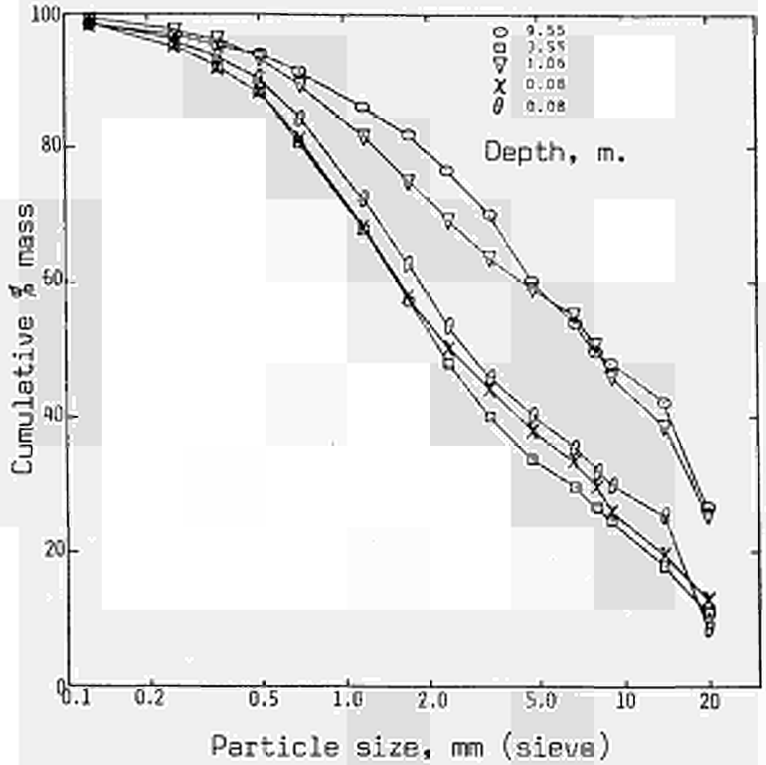


Figure 1. Size distribution of sedimented dispersed dross at different depths

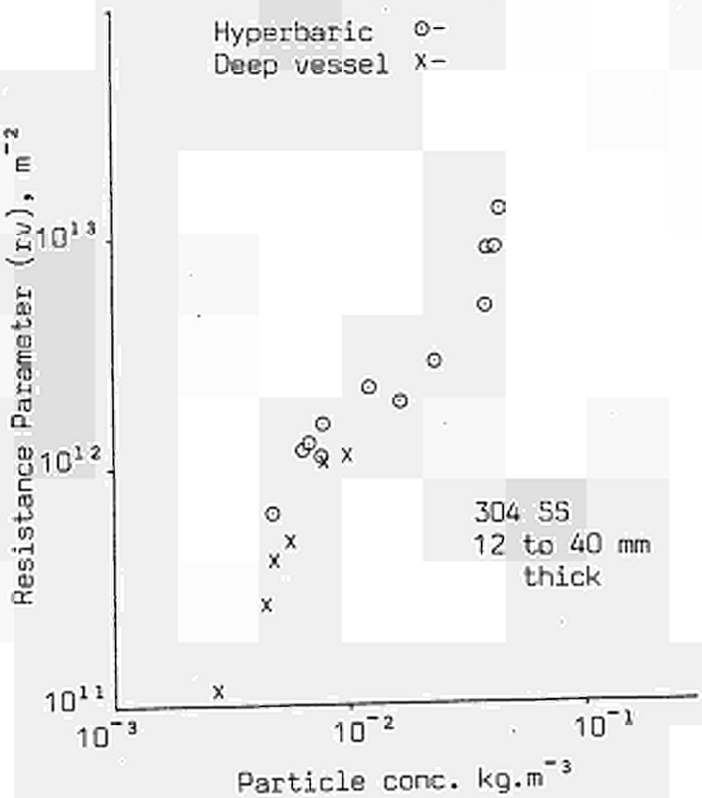


Figure 2. Filtration resistance of fine suspended particles in water

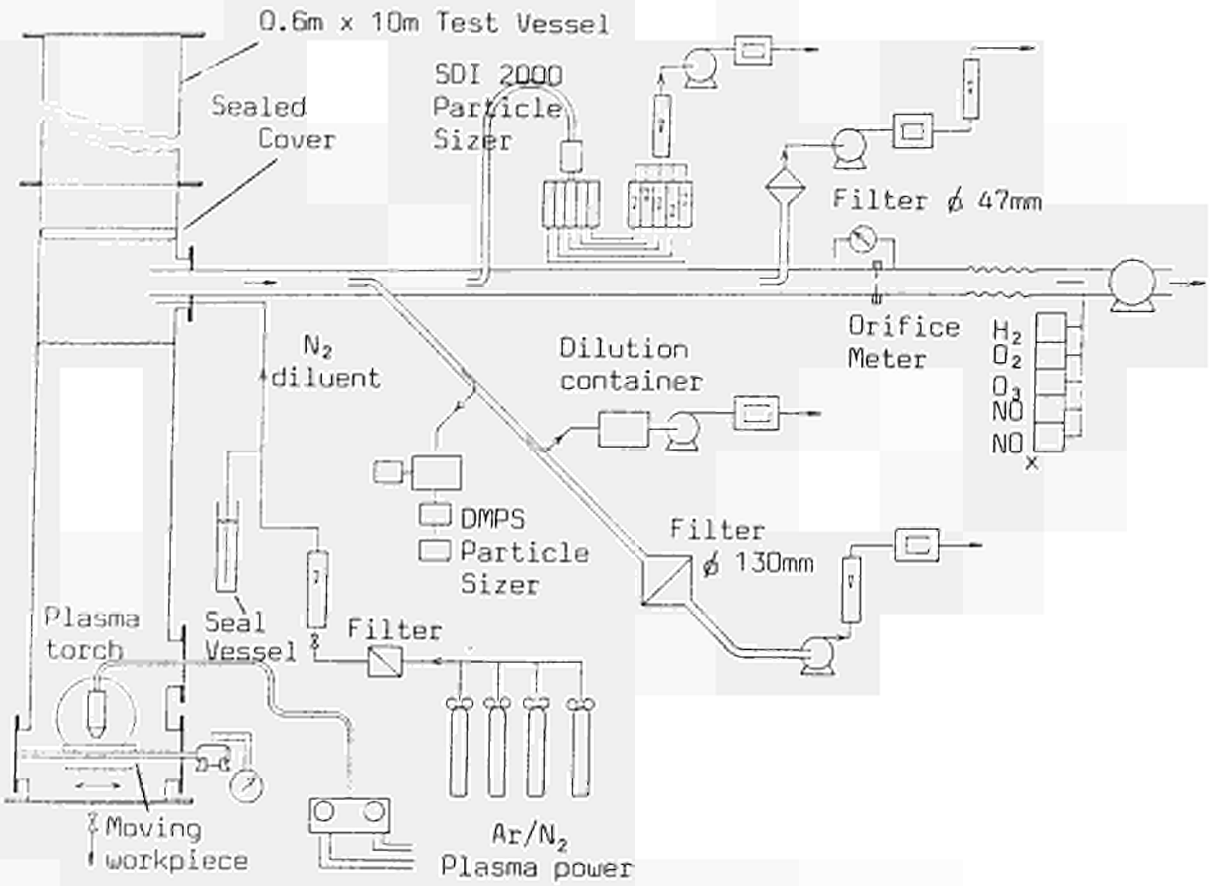


Figure 3. Sampling, fume and gas analysis instrumentation on deep water test vessel

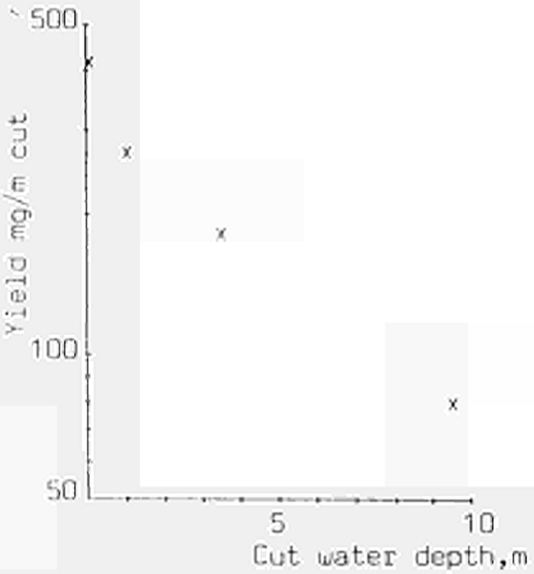


Figure 4. Effect of cut water depth on yield of fume particles in effluent gas

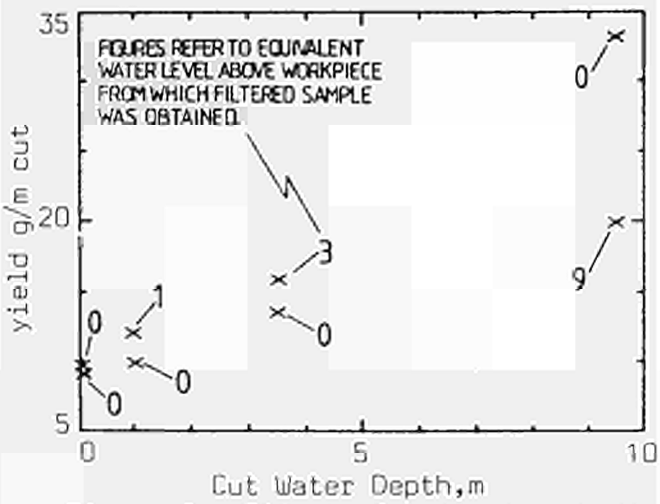


Figure 5. Effect of cut water depth on yield of fine suspended particles in water

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

Summary

Work in this contract is now completed. During this year, the research work has covered following parts of the working programme:

- Testing of the transversal cutting process for Steam Generator (SG) tubes (B.1.1.).
- Design and fabrication of a complete device for internal cutting (B.1.3.)
- Performance tests on representative Inconel tube samples (B.1.4.).

  Note: Work package B.1.5. had to be abandoned due to the non-availability of full-scale samples.
- Internal arc-saw cutting of pipes (B.2.2., B.2.3.).
- Development of an axial cutting process for SG tubes (extension of the work programme B.3.).

In this report, a summary of the obtained cutting parameters used for transversal and axial cutting of SG tubes, and transversal cutting of large pipes will be presented.

Progress and Results

1. Testing of the transversal cutting process for SG tubes (B.1.1.)

During this period, final tests have been carried out on the transversal cutting process, using Molybdene electrodes (Ø 15x1mm) and Stainless Steel (SS) tube samples of Ø 22.2x1.27mm.

The theoretical maximum number of SG tube cuts possible with the same electrode is equal to: Rus x 0.421, where the wearing ratio Rus is defined as machining section/wearing of the electrode.

- the best Rus (4.55) is obtained with a rotation speed Vr = 1000 rpm, an average intensity Imoy = 64 A, and a cutting time Tc = 90s.
- in case of work with a low intensity, for example 15 A, the best Rus (3.4) will be obtained with Vr = 2000 rpm and a cutting time Tc = 160s.

On the manually operated test bench, one Mo electrode can afford only one cut (Rus = 4.55 against 4.75 needed). But it is considered that a motorised remote controlled tool would permit two or three cuts, due to better guidance of the electrode.

This ratio could also be increased by using more appropriate materials for the electrode, as used for example in TIG welding.

Figure 1 shows a typical cutting aspect, without dross or deformation of the tube.

2. Design and fabrication of a complete device for internal cutting and tests on representative Inconel tube samples (B.1.3., B.1.4.)

Because of the shape of the water plenum in PWR Steam Generators, and due to the thickness of the tubular plate (550mm), a remote controlled tool carried by a manipulator needs to be flexible for allowing access to the peripherical tubes.

A flexible tool has been developed with its dedicated bench, for a full-scale testing on representative Inconel and SS tubes (Figure 2).

The cutting parameters for Inconel tubes were found as follows:

- rotation speed Vr = 1500 rpm,
- average intensity Imoy = 50 A,
- cutting time Tc = 100 s,
- water flow = 2.5 l/mn.

The wearing ratio of the Mo electrode decreases when cutting Inconel, as the energy consumption is higher than that of SS tubes. However, the above device allows a more regular cutting process, without peaks of intensity, as the excentration control of the electrode is much more accurate with a good centering in the tube. A regularly melted ring can be seen from the very beginning of the cut.

With very accurate parameters, only two tubes cut can be done with the same Molybdene electrode.

3. Development and testing of an axial cutting process for SG tubes (B.3.)

This process has been developed in addition to the original work programme to facilitate the extraction of the tube end, which is expended across the tubular plate thickness; this is done by making vertical kerfs along the tube to release the residual stresses.

Two cutting heads have been tested on the test bench:
- rotation of the electrode via a mechanical gear box,
- rotation of the electrode via a turbine moved by water flood.

The usefulness of the turbine cutting head appeared rather limited as the rotation speed of the electrode cannot be controlled or adjusted, which is essential for remote control.

Some tests made with Mo electrodes on SS samples have demonstrated that a kerf of 1 meter length can be executed with a single electrode, without noticing a significant wearing.

The cutting parameters with Molybdene electrodes were found as follows:
- rotation speed Vr = 1750 rpm,
- advance speed Va = 12 mm/s,
- average intensity Imoy = 50 A,
- cutting time for one 550 mm long kerf Tc = 46s,
- a constant water flood is needed on the electrode.

Figure 3 shows the typical aspect of a kerf, with dross formation along it.

4. Internal arc-saw cutting of pipes (B.2.2., B.2.3.)

Tests were made on large SS pipes Ø 190x6mm and Ø 236x6mm, on tubes Ø 89x4mm and rods Ø 22 mm, with electrodes of different size made of Cu, Steel or Mo.

A formula based on the experimental data correlates the fact that it is not possible to cut through 10mm thick pipe walls with the limited available intensity of 100 A, and even not with the design 600 A capacity cutting head used on the test bench.

This correlation is: Vd = Ilim * Cus / Se
where Vd is the cutting speed, Ilim the intensity limit of the generator, Cus is an experimental machining ratio (machining section/quantity of energy needed), and Se is the "working" section of the electrode.

For cutting through a SS pipe of Ø 81x6mm in 5 minutes, Vd should be 0.85mm/s. As Vd is directly proportional to Ilim, and as the maximum Vd obtained on the test bench was 0.19mm/s, the limit of the generator should be of about: Ilim = 100 * 0.85/0.19 = 450 A.

The cutting through of a Ø 240x6mm large pipe in an "industrially acceptable time" of 15 minutes is not possible by a carried remote controlled tool. The experimental data result in the requirement of a 1600 amps generator with a Ø 90x2mm electrode.

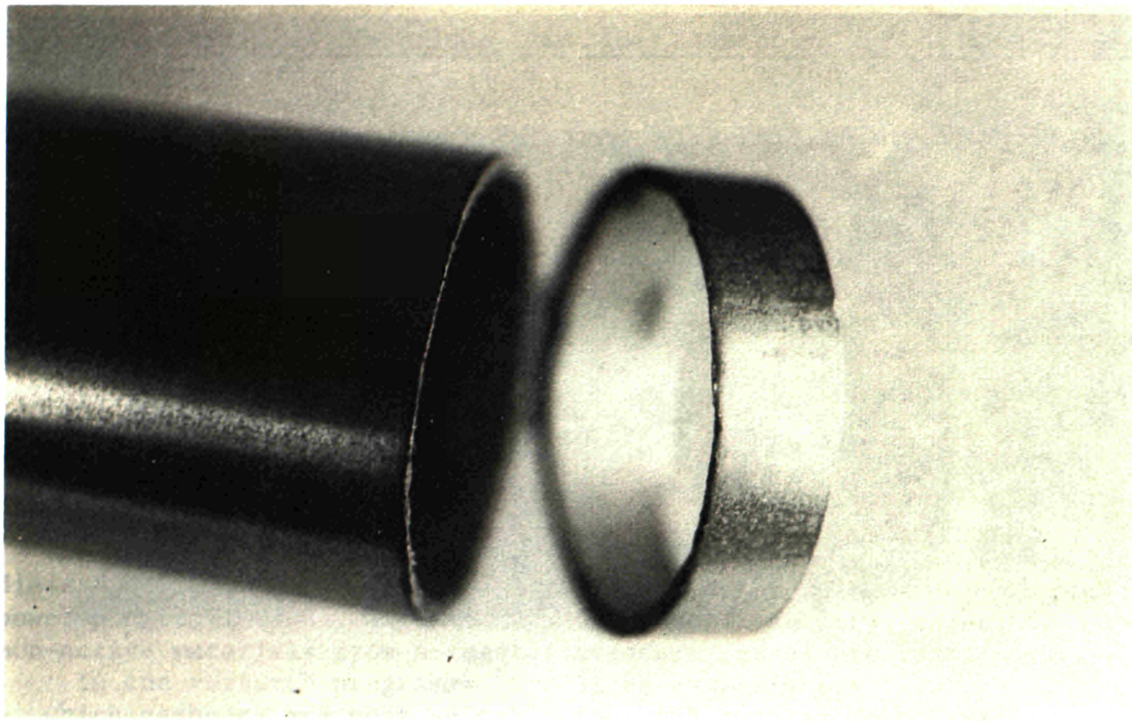


FIGURE 1 : Transversal cutting SS tube

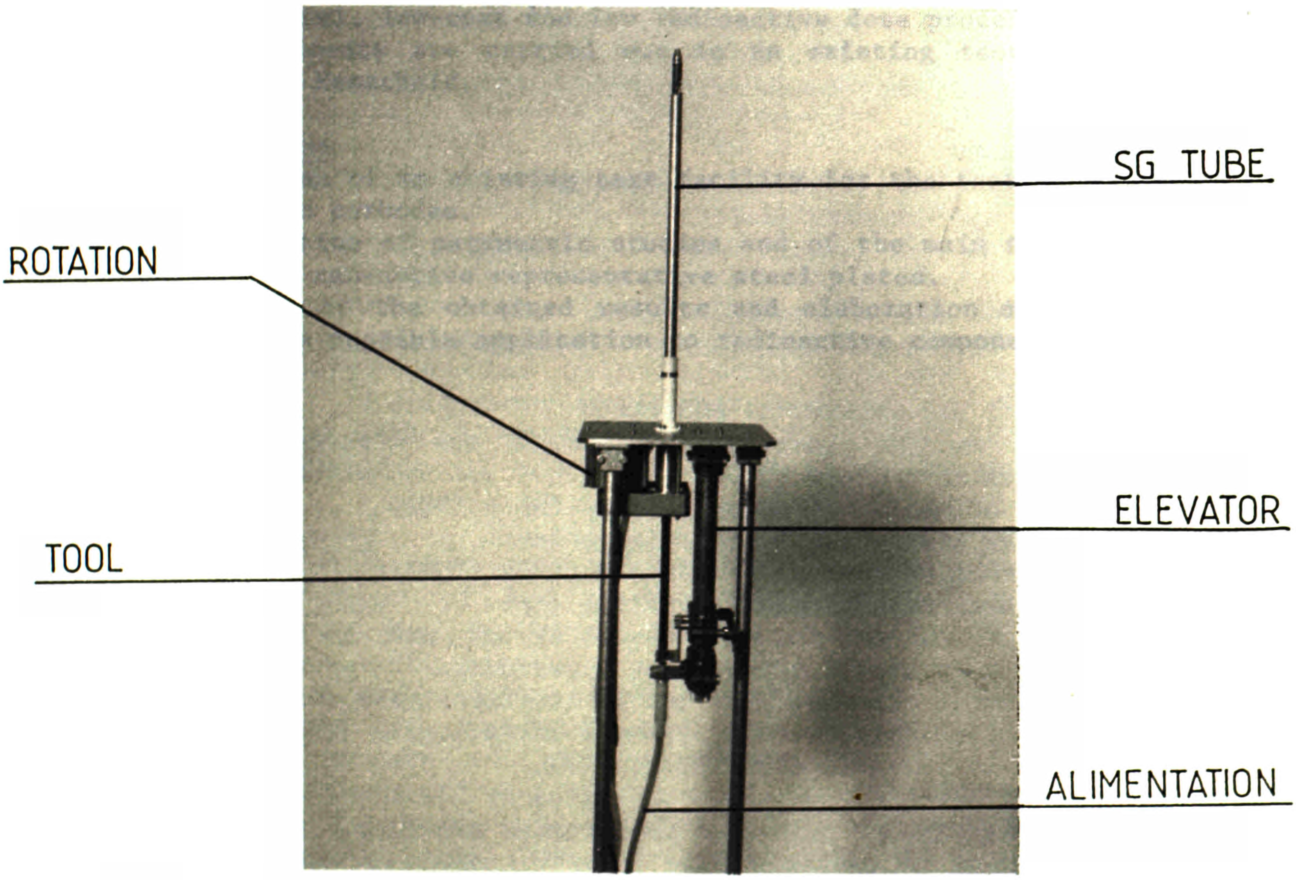


FIGURE 2 : Full scale testing bench for transversal cutting

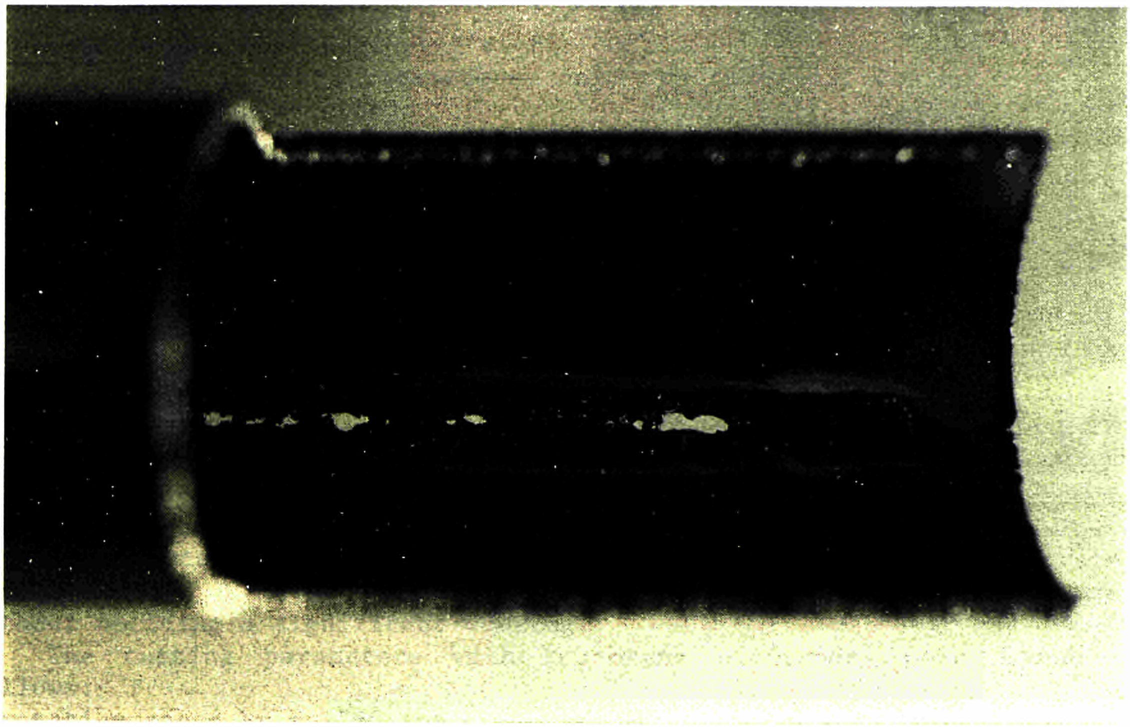


FIGURE 3 : Axial cutting SS tube

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 the 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.

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

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

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 (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

A higher concrete removing efficiency was found for simultaneous compared to sequential firing. The explosive charge mass per bore hole used was low such as to lie at the threshold of cratering/no cratering.

This result becomes even more pronounced when the blasted surface is subjected to aftertreatment by a striking tool.

Mechanical drilling of the charge holes was investigated with respect to work time, logistic requirements and remote handling capability. Core drilling is found to be favourable in all cases where reinforcement is present in the concrete.

A test setup for reduction of dust during concrete blasting has been designed.

Progress and Results

1. Initiation Mode (B.1.1)

1.1 Principle

Initiation of a large number of explosive bore hole charges is normally performed in groups together, with several groups delayed to each other by several tens of milliseconds. This delay on one hand is long enough so that the fragmentation process is concluded in one group of charges before the second group is blasted and on the other hand is still short enough such that the detonators in the second group are not yet mechanically damaged /1/.

The initiation time of the detonators within one group normally scatter within one or two milliseconds. This time scatter is of the same order of magnitude as the duration of the pressure pulses emitted by the individual charges. The duration of these pressure pulses had been found earlier during these investigations to lie between 0,5 and 1 millisecond depending on the charge mass and on the bore hole geometry. As a consequence the pressure pulses of neighbouring charges will overlap only occasionally and in an unpredictable way. It had been suggested earlier /2/ that precise timing should give a better cratering result through in - phase superposition of the individual pressure pulses.

1.2 Experiments

The experiments were performed on two identical concrete test bodies of 1,2 m diameter and 0,5 m thickness. Each test body was equipped with 7 bore holes drilled in symmetric hexagonal array.

The bore holes were loaded each with 11.5 g of PETN type explosive and tampered with a plug of rapid bonding cement. The amount of explosive was at the threshold to cause any fragmentation. In one test-body exact simultaneity was verified by applying a NONEL initiation cord of equal length to each charge. The seven free ends of the cords were connected by a PETN booster charge which was initiated by an electric detonator. The bore holes in the other test-body were fired sequentially.

1.3 Results

With one exception there was practically no cratering in the sequential firing test. Blasting of all 7 charges within < 10 microseconds simultaneously gives a different result: All 7 charges show small craters.

Following the blasting a second working step will generally be necessary in order to remove the steel reinforcement layer and the partially loosened concrete /3/. This may be performed by a combined application of cutting and striking tools. It was therefore decided to subject both test bodies to an aftertreatment by a light weight striking tool. The total energy applied with 200 strikes was about 10 kNm for each test body. While the total removed mass is low due to the minimum loading a significantly higher concrete removing efficiency is found for simultaneous blasting without and with additional mechanical aftertreatment, see Fig. 1.

2. Mechanical drilling (B.3.2)

Light weight drilling equipment can be used due to the limited hole diameter of about 40 mm. Equipment tested was manually operated, handling required one single person. Both full volume percussion drilling and core drilling was investigated. The following results were obtained:*)

- Full volume drilling is about a factor of 10 faster in concrete free of reinforcement steel.
- Full volume drilling in steel reinforced concrete was not possible with hole diameters exceeding ca. 10 mm, up to 10 mm any rebar caused excessive drilling time and wear of the drill bit.
- Scatter in hole-to-hole drilling time is of the order of a factor of 2, in exceptional cases a factor of 4 for core drilling.
- No significant reduction in drilling time with 15 % decrease in hole diameter was found.
- Drill core extraction caused no problems.

3. Capability for remote operation of drilling (B.3.2)

Table I gives an overview over the criteria for handling under remote control. As can be seen, the advantage of higher drilling speed with full volume percussion drilling is not significant in view of additional time consuming work steps and the limited applicability to areas of non reinforced concrete. Therefore core drilling is favoured. During core-drilling from a flat surface the equipment can support itself by a vacuum suction foot. Otherwise mechanical support by a robot arm during drilling will be required in cases where the wall surface is very uneven and in corners. Further investigation of functional control programs and sensor application is necessary. Adjustment of existing robot technics should include the requirements for explosive charge loading and tamping.

*) For percussion drilling results of ref. /1/ are included.

4. Dust from blasting: Effect of spraying (B.5.2)

An experimental setup to reduce the dust generated during the concrete blasting has been designed. It consists of an explosive water spray generator. A description together with the results of tests to be performed jointly with Taylor Woodrow Construction Ltd. will be given in the next annual report.

References

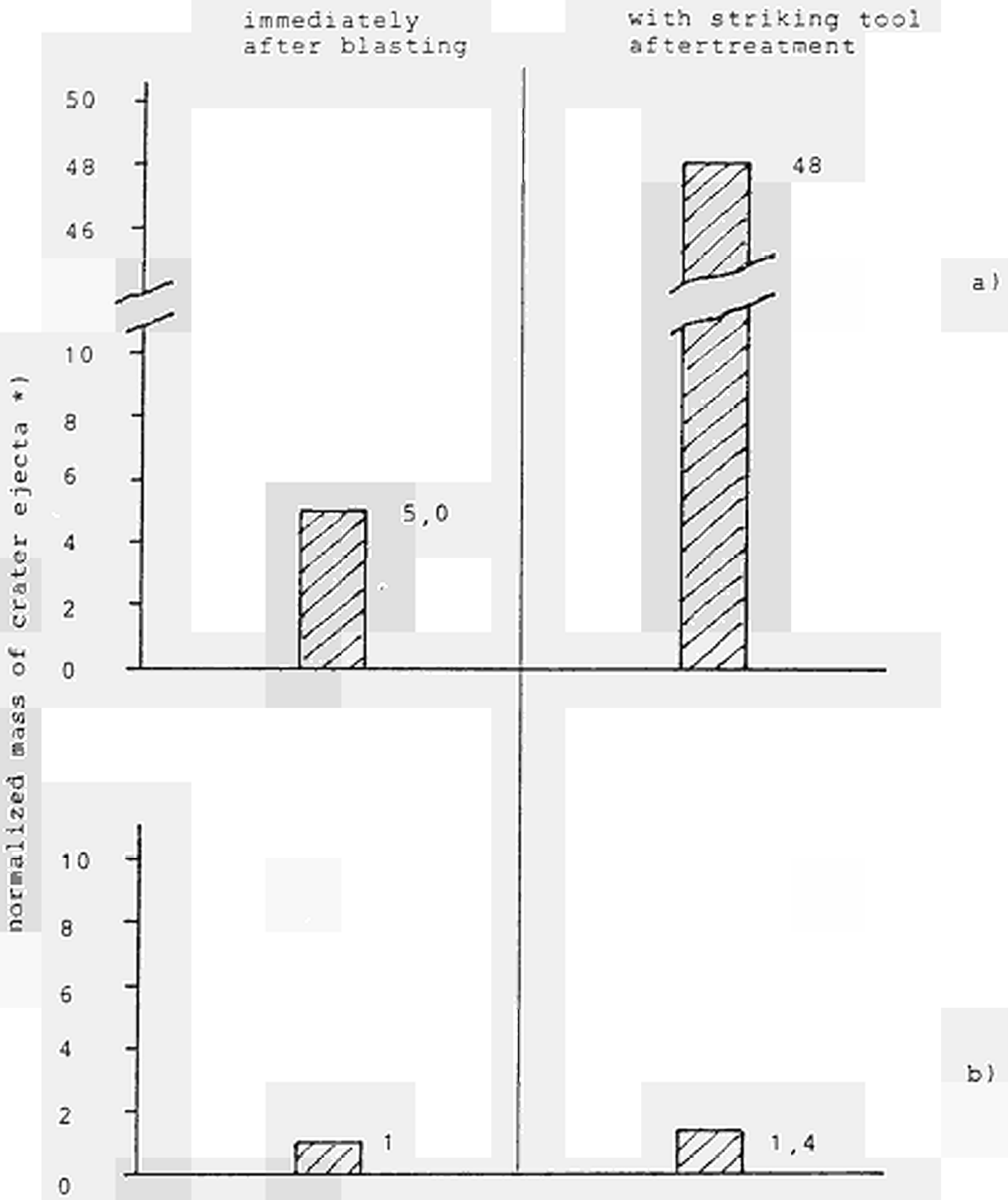
/1/ OSEBOLD, R., Abbruch von Massivbauwerken Verlagsgesellschaft R. Müller, Köln-Braunsfeld (1981)

/2/ Durchführbarkeit der Zerlegung des Biologischen Schilds mittels Bohrlochsprengtechnik (BMFT-Projekt 02 S 7031), Battelle-Report BIeV-R-65.036-4 (1983)

/3/ Beton-Zerlegeversuche in einem geschlossenen Containment, Technical Report PHDR (1986), 62-85

Table I: Criteria for remote control

<table>
<tr><th></th><th>full volume
percussion drilling</th><th>core drilling</th></tr>
<tr><td>mechanical requirements</td><td></td><td></td></tr>
<tr><td>- positioning,
re-positioning</td><td colspan="2">3-dimensional handling of 20 kg equipment
directional control
positioning accuracy 1 cm</td></tr>
<tr><td>- drill operation</td><td colspan="2">compensation for compressive force of drill</td></tr>
<tr><td></td><td>compensation for
percussion force</td><td></td></tr>
<tr><td></td><td colspan="2">compensation for momentum of drill equipment 10 kp.m</td></tr>
<tr><td></td><td>dust collection:
air flow flexible pipe
support</td><td>cooling water flow:
hose pipe support, closed
loop</td></tr>
<tr><td>- maintenance and repair</td><td colspan="2">replacement of drill bit
hose or pipe repair (leakage, clogging)
repair of drill motor
replacement of rubber seals</td></tr>
<tr><td>control requirements</td><td colspan="2">position of drill
hole depth achieved
axial drilling force
angular momentum of drilling
wear of drill bit</td></tr>
</table>



*) mass normalized to the value of sequential firing immediately after blasting

Fig. 1: Reduced mass of concrete removed without and with mechanical aftertreatment

a) simultaneous firing
b) sequential firing

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: I. Ll. Davies

A. Objectives and Scope

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

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

The present research work aims at complementing, improving and optimising the foregoing work. Extensive investigations will be executed on the adjustment of blasting parameters, material and structural effects, drilling techniques, particle distribution and on procedures for remote handling. Work is carried out jointly - based on a common and complementary work programme - with BF (contract N° FI1D-0011).

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 - (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 1987 has concentrated on work packages B1 and B2 "Adjustment of Blasting Parameters" and "Study of Material and Structural Effects". Towards the end of the year preparatory work was started on work packages B3 and B5 "Drilling and Boring of Charging Holes" and "Investigation of Generated Dust During Blasting".

Tests have been carried out to investigate the effects when explosives are placed and fired in concrete which is faced with a steel liner. These tests have demonstrated the feasibility of using explosive charges to achieve the desired results of debonding the liner and its anchoring system from the concrete. Tests have also been carried out to investigate how material may be removed at re-entrant corners, around obstacles and from curved surfaces with small radii of curvature.

Progress and Results

1. Replication of a Battelle Institute (BF) Test (B.1.3)

To assure comparability of test results produced by (BF) and Taylor Woodrow it was arranged that each contractor would repeat a test previously undertaken by the other in the preceding programme of work /1/ and /2/. To this end a test previously carried out by (BF) in which the long borehole technique was used ie where the boreholes lie in a plane parallel to the target face, was selected for replication.

The results from an initial firing indicated some tests differences in comparison with Battelle Institute's work. These appear to have arisen from charge effects and tamping procedures. It is therefore decided to repeat this test. Preparatory work has been commenced to cast and test a new model.

2. Effect of the Presence of a Liner (B.2.1)

A typical biological shield will have a steel liner on the inside surface of the containment structure to provide an impermeable membrane. In the decommissioning of such a structure it will be necessary to remove both liner and other activated parts of the containment structure. To meet this requirement investigations have been carried out to determine the efficiency with which the liner and its anchoring system may be debonded from the concrete by firing explosive charges buried in the concrete adjacent to the liner.

Preliminary tests reported on in the preceding annual progress report /3/ 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. Further firings have been carried out to optimise on the test results when using single charges and to investigate the debonding and cratering characteristics when groups of charges are fired simultaneously. Three charges spaced equidistant from each other were fired simultaneously for each multiple charge test. Charge weights and depth of burials were selected from the preceding single charge test results.

These tests demonstrated that the chosen charge separations and charge characteristics were sufficient to ensure that the combined effect of the charges produced debonding of the liner and its anchoring system over a single area of the liner (see Fig 1). Post test inspection indicated that beneath the liner there was considerable break up of the concrete. It is proposed to cut the model at a later date to facilitate the inspection of concrete break-up beneath the liner and the debonding of the anchor stud/liner connections.

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

To date, tests in this development programme have been carried out on flat test blocks or on curved surfaces having large radii of curvature. In addition cratering has taken place away from edges, sharp or re-entrant corners and from any penetrations. However in the decommissioning of nuclear facilities many occasions will arise when material will have to be removed from corners or near penetrations, gas ducts etc. Model testings have therefore been started to investigate how material may be removed at re-entrant corners, around obstacles and from curved surfaces with small radii of curavature.

Initial indications from these tests are that material could be removed from re-entrant corners by careful positioning of charges (see Fig 2). Crater characteristics, when explosives are buried in the concrete adjacent to a concave curved surface, would appear to reduce crater width as the radius of curvature of the surface is reduced. The presence of a proximate lined penetration was found to re-shape the crater contours in its vicinity.

Further tests are planned for the early part of 1988 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.

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

In the preceding programme of work/1/, it was demonstrated that shaped charges may be used to produce boreholes suitable for placing cratering charges. It was shown that a 200 gram shaped charge could be used to produce boreholes, in reinforced concrete, of diameter 33mm and 700mm long. However, the dimensions of this hole does not meet current borehole requirements for cratering charges. Preparatory work has therefore been started to modify the shaped charge design to produce a hole of the dimensions required for the explosive removal of reinforced concrete. The effectiveness of the modified shaped charge is to be demonstrated experimentally.

5. Assessment of Particle Size Distribution of Generated Dust During Blasting (B.5.1)

Preparatory work has been started to assess the particle size distribution of dust generated during blasting. For this investigation it is proposed to surround each test block with a PVC enclosure which is to help to contain the generated dust from the blasting. Samples of air and entrained particulate will be removed, using a suitable fan and ducting, for analysis. It is proposed to use a combination of an electrical aerosol analyser and a light scattering particle size analyser to study the entrained particulate. In addition it is proposed to draw a sample of air through a HEPA filter to give the total mass concentration of entrained particulate (see Fig 3).

References

/1/ FLEISCHER, C. C., Taylor Woodrow Construction Ltd, CEC Report No EUR 9862 EN (1985). "A Study of Explosive Demolition Techniques for Heavy Reinforced and Prestressed Concrete Structures".

/2/ FREUND, H. U. et al, Battelle - Bericht BIeV-R-65.036-4 (1983). "Durchfuhrbarkeit der Zerlegung des Biologischen Schilds mittels Bohrlochsprengtechnik".

/3/ CEC-EUR 11112 The Community's research and development programme on decommissioning of nuclear installations. Second annual progress report (year 1986).

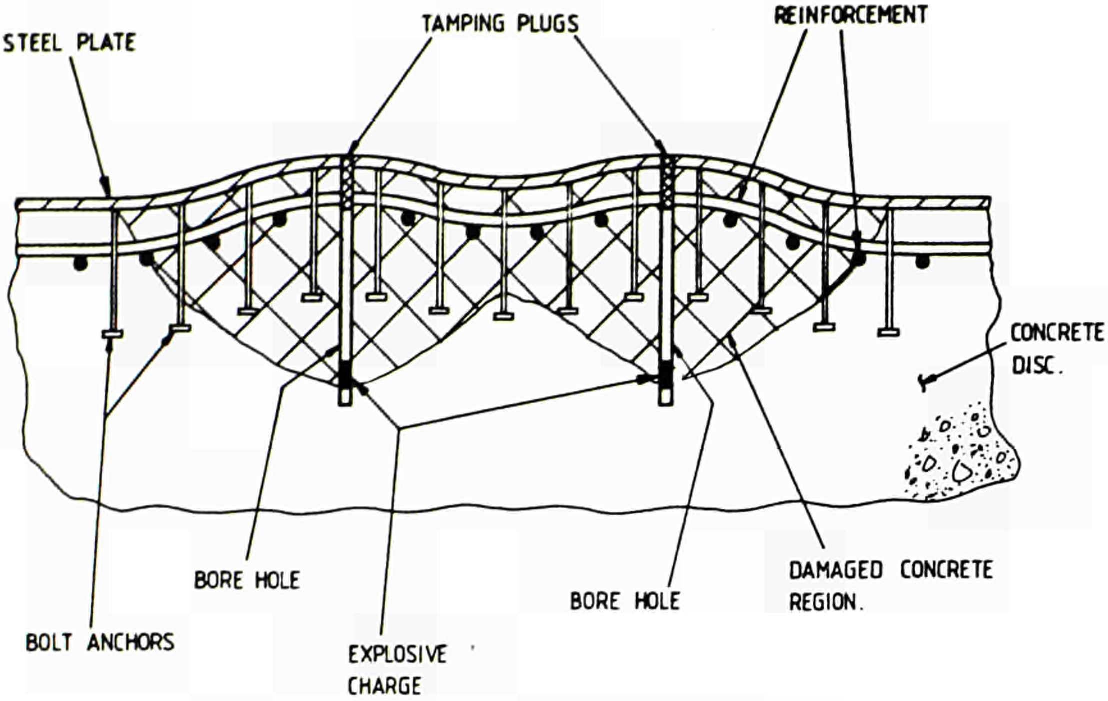


Fig. 1. Sketch of simulation of a steel lined concrete face after firing of multiple charges.

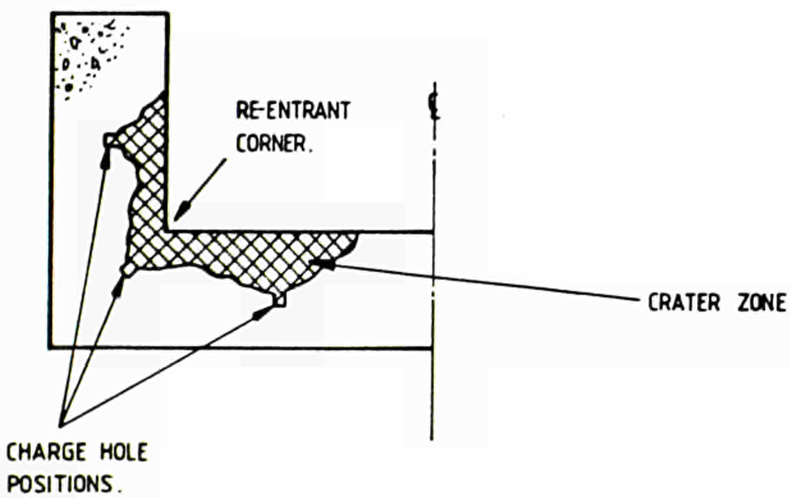


Fig. 2. Section showing removal of concrete from a re-entrant corner.

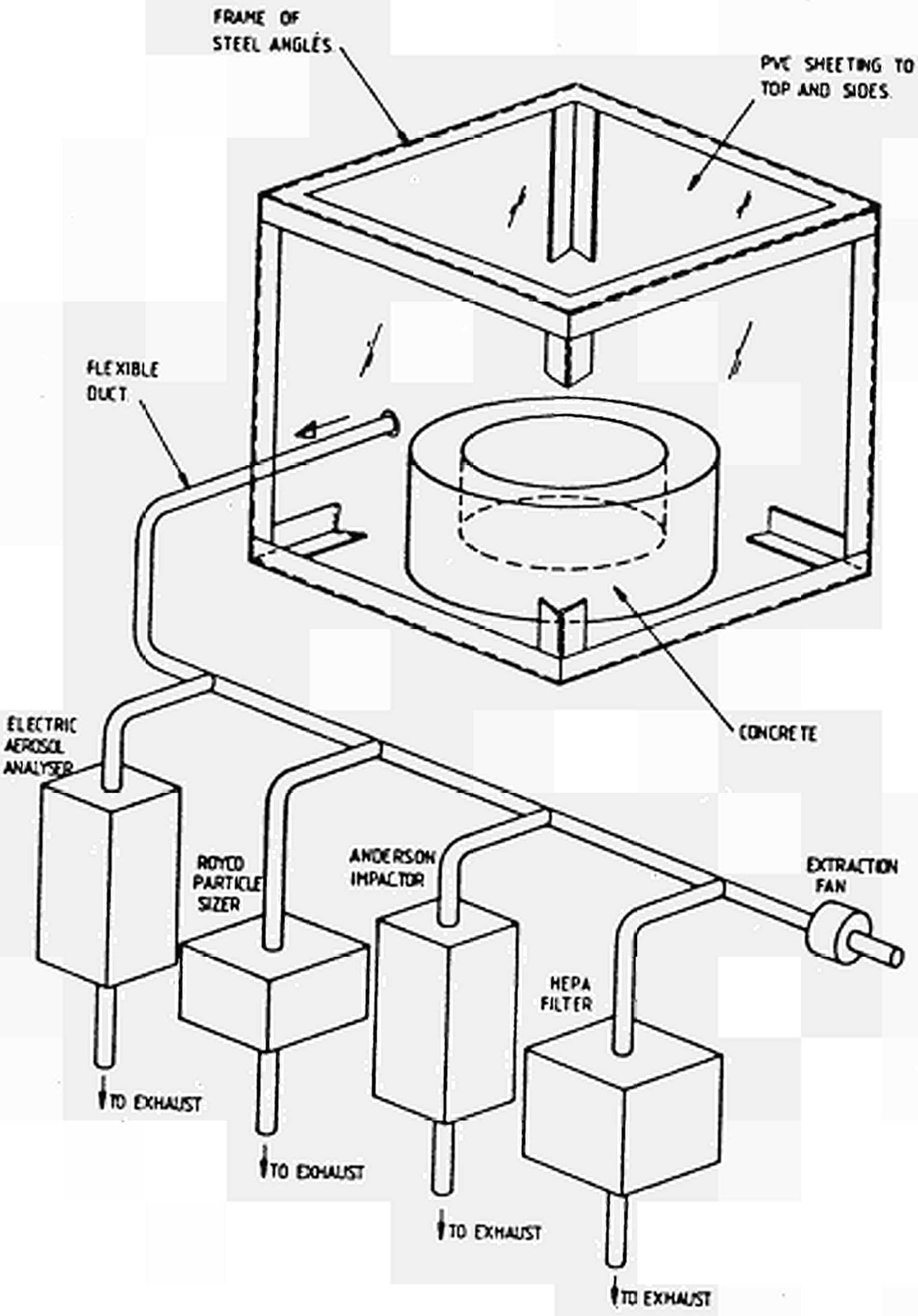


Fig. 3. Sketch of test set-up for particle size distribution analysis.

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.

B. Work Programme

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

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

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

C. Progress of Work and Obtained Results

Summary

The main facts during the present year (1987) have been :
- Reception of the mechanical arm and remote control.
- Installation of the arm on the hot cell mock-up.
- Preliminary laser cutting tests.

These various phases overlap several technical stages as : the tests and setting of the slave processor after interconnection between the romote control and the arm, and the coupling with the CO2 laser.

After these stages, at the end of the year, we have performed the first laser cutting on plane metal-sheet.

Progress and Results

1. Functional testing (B.1.)

The laser robot ROLD has been delivered at CEN-Saclay in february 1987. The mechanical design had been commited to BARRAS-Provence company.

The articulated arm was immediately settled on the hot cell mock-up, in order to achieve the mechanical and electrical reception tests of the various components : gap control of the joints, test of the transmission, motors, brakes, feedback components, resolvers, tachometers. The motion of the joints have been checked (without slave command).

The remote control has been connected to the robot. The heart of the system includes an INTEL 8086 microprocessor in standard multi-bus I network. The operating system is RMx86. This industrial control cabinet (COMMERCY) has been modified in order to be adapted to the specifications of the laser robot.

We have progressively tested the various functionalities of the digital circuits of the encoder cards (signal position).

Speed variators have been adjusted using a classical method (response to a speed step).

We developped a special process which enables us to adjust joint by joint the digital slave processor. The collected data : position gain and speed gain have been integrated in the software of the slave command of the robot.

The functionalities of the software trajectory, which have been tested use a coordinates transformer.

This software includes :
- Initialization (original position)
- Software limits (angular limitation)
- Recording of teaching points of the trajectory
- Teaching parameters of the laser process (laser power, speed, opening of the shutter, gas valve)

Static repeatability tests of the robot show that expected performances are obtained (+/- 0.074 mm).

2. CO2 Laser Connection to the Arm (B.2.)

An optical path has been established between the CO2 laser and the top of the wall of the hot cell mock-up. The length of the path is about 10 meters. This line is composed by two mirrors (gold platted transmitted through metallic tubes.

The eight reflecting mirrors which are inserted inside the mechanical structure of the arm, have been oriented.

The particular technology used allows a quick replacement of the mirrors and do not affect the optical alignment.

3. Laser Cutting Tests (B.3.)

The first tests of cutting with teaching trajectories began with plane metal sheet samples. These tests have been made in several zones of the working volume of the robot arm (Figure 1).

The preliminary cuts show a kerf width of 0.3 mm with a roughness of Ra = 10 µM. Sample thickness is 2 mm.

For demonstration tests, we have chosen two kinds of samples : pipes with various diameter and mechanical structures of the ceiling of the hot cell. The maximum thickness of these samples will not exceed 6 mm.

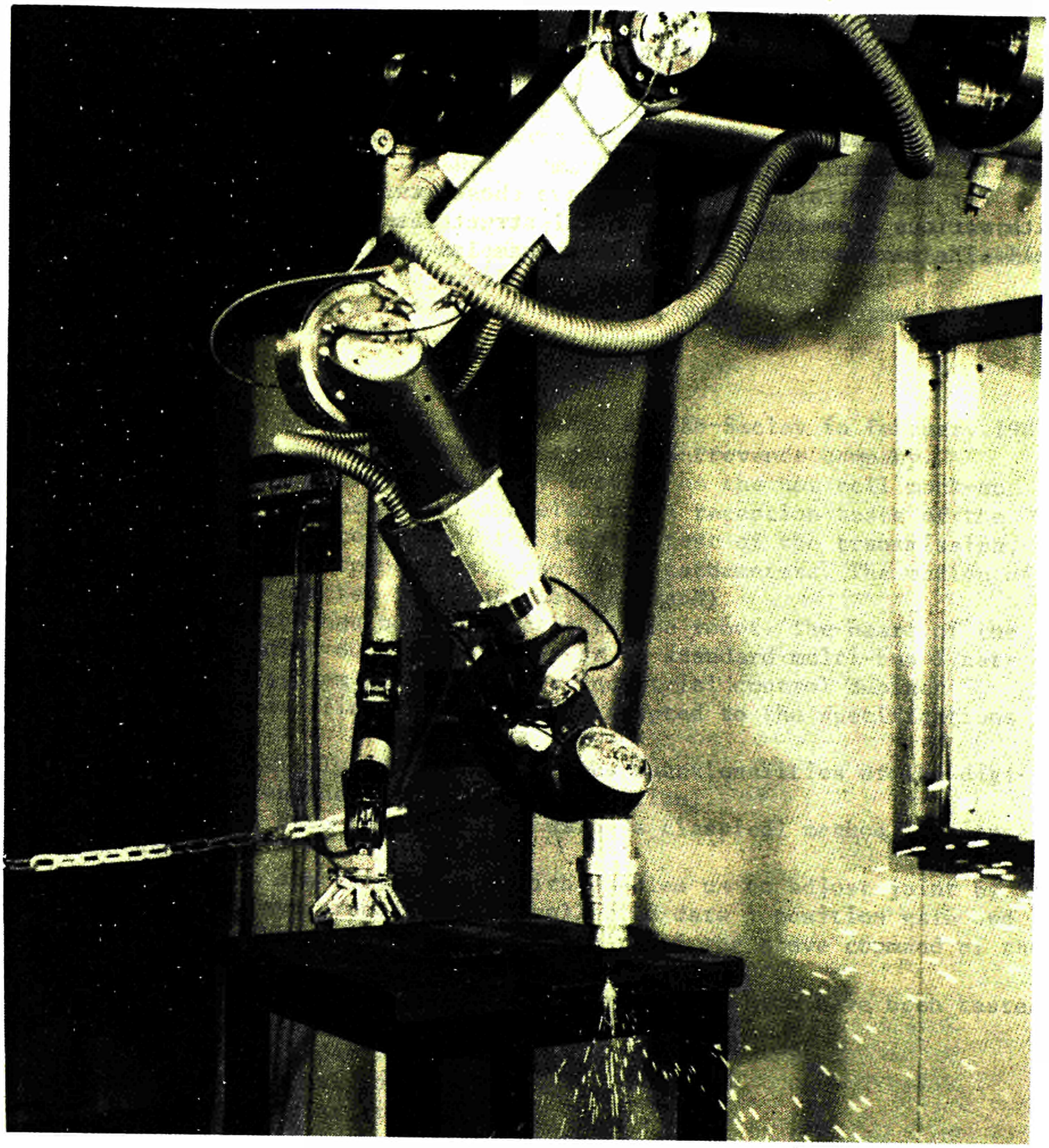


FIGURE 1 - ROLD - Dismantling Laser Robot
First Cutting Test

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 - December 1988
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. <u>Progress of Work and Obtained Results</u>

Summary

The experimental study and the related analysis on the granulometry of the aerosol generated during the laser beam cutting of metal samples have been completed and led to the conclusion described below.

The tests on laser beam set-up either in order to study beam propagation or to identify the cutting capability far from the source with high power laser beam, have been completed.

Some experimental tests were carried on to measure the beam characteristics along the path and the air velocity inside the pipes which enclose the radiation.

Using appropriate focalizing elements it was possible to cut metal samples of Fe 42C and AISI 304 with a thickness up to 45 mm with 4 kW laser power.

One component of the Garigliano plant (the steam drum) has been chosen as a reference component to verify the practical applicability of the laser cutting tecnology to the decommissioning.

The related drawings have been collected and the study is now in progress.

Progress and Results

1. Aerosol characterization (B.1.)

The main results obtained during 1987 can be summarized as follows:

The study on the granulometry of the aerosols generated during the laser beam cutting of metal samples, puts in evidence the presence of two fractions: the first one, which corresponds to more than 2/3 of the total aerosol mass, is composed by aggregates of many, very small, particles, the second one is composed by solid spherical particles with diameters larger than some microns.

The mean diameter, obtained by means of impactors, is mainly due to the aggregates and depends upon the initial concentration and upon the time.

The change in chemical composition of the aerosols with respect to the sample material (enrichment in Ni and reduction in Cr) has been found exclusively in the aggregates (in case of use of O_2 as support gas).

The total mass of the generated aerosols, in case of use of O_2 as cutting gas, corresponds to 1/10 of the mass lost by the sample and therefore, taking account of the chemical composition, the metal mass associated to the aerosol is of the order 1/20 of the mass lost by the sample.

In case of use of air as cutting gas the above quantities are reduced by a factor of two.

The examination of the various cutting parameters puts in evidence the importance of the cutting velocity, of the sample thickness and of the support gas as regards the amount of generated aerosol.

The use of air instead of O_2 leads to a remarkable reduction in amount of aerosol, however, the cutting efficiency obtained with O_2 has been reached, in case of use of air as support gas, only by doubling the laser power.

2. <u>Assessment of the applicability of remote operation laser cutting to specific Garigliano BWR components (B.2)</u>

In order to understand the thermal blooming effect, due to the presence of particles of aerosols along the laser beam path, where the radiation is not collimated, some conclusive theoretical studies have been carried on.

The experimental set-up has been completed including metallic pipes for gas flow control and for operator safety.

In figure 1 and figure 2 respectively a general view of the system and the cutting station near the Spectra-physics 5 kW laser source are shown.

Water cooled copper mirrors and a Zn Se lens have been utilized for beam propagation and directing, as focalizing elements with appropriate focal length.

It has been necessary also to work with a final optic's protection against the particulate produced during the cutting process and, consequently, a particular nozzle has been developed.

The air velocity in the transversal section of the pipe has been measured with a hot film anemometer in order to control its own uniformity during the flow variation.

The radiation attenuation after propagation and the beam diameter have been measured.

The experimental beam characteristics were slightly different in comparison with the calculated ones.

Therefore it was necessary to rebuild some components of the cutting system.

Laser cutting tests on metal samples have been carried out on mild steel Fe 42C and stainless steel AISI 304 with thicknesses between 15 and 45 mm and with a laser power of 4 kW.

The study of a remote control system aimed at checking the applicability of the laser beam to the decommissioning of one component of the Garigliano plant will be performed using the steam drum as the reference component.

The main mechanical drawings of this component have been collected and the study is started.

Figure 1: General view of the experimental fixture

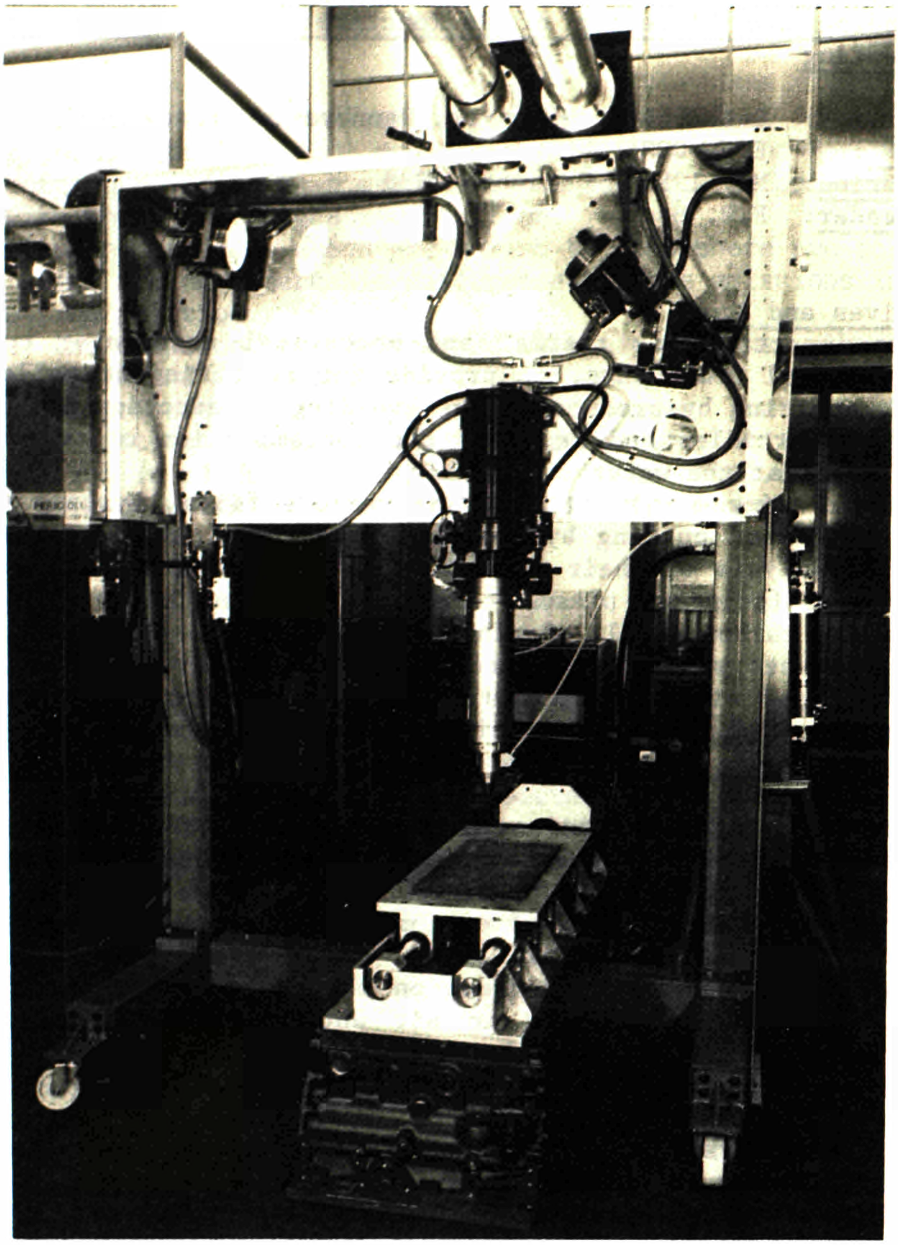


Figure 2: Cutting station

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 parameters of underwater plasma arc cutting have been found out for cutting stainless steel (1.4301 equiv. to AISI 304) in the horizontal and overhead cutting position in a waterdepth of 100 mm. The parameters are compared with those of the gravity position.
The particles and aerosols which are coming to atmosphere are collected and analysed. There are no remarkable differences to the emissions in normal gravity cutting position.
The mass of the suspended and sedimented particles in the water is analysed for cutting different materials.
The underwater grinding technique has been optimized. The parameters were found out for cutting stainless steel and aluminium. The result is that the rate of abrasion is rather low compared with those of aluminium.
A dust and aerosol emission could not be measured even with a condensation nucleous counter.

Progress and Results

1. Determination of the cutting parameters (B.2.)

The parameters of underwater plasma arc cutting of stainless steel have been found out for the horizontal and overhead cutting position. It was the aim to find a combination of cutting parameters which gave a low aerosol emission and a good lifetime of the torch. The torch was mounted as close as possible to the vertical z-axis of the manipulator for these experiments (see APR 1986 for a detailed description of the test facility). That makes sure that most of the rising gas bubbles are coming into the suction hood for a good collection efficiency.
The parameters are compared with those of the normal gravity position. The results are given in table I.
This table shows the cutting parameters for the material thicknesses of 20, 40, and 60 mm, each in the gravity position (g), the horizontal position (h) and the overhead position (o). The main parameters are the electrical data, voltage and current, the diameter of the nozzle, the feed respectivly the cutting speed and the emitted aerosol mass in gramm per meter length of cutting. There are no serious differences between these three cutting positions especially when compared with the gravity position.
The handling of the plasma cutting process in the overhead position gives no problems. The main advantage is that the overhead position allows a greater distance between nozzle and workpiece so that there are no stray arcs between nozzle and workpiece.

The cutting or grinding with an abrasive wheel, driven by compressed air, produces a rather small seam and a very smooth cutting edge compared with the plasma technique. But it needs a very solid construction of the manipulator because of the large cutting forces.
The used grinding tool has a nominal power output of 1.2 kW. It is driven directly without any gearbox. The diameter of a new disc is 178 mm, the thickness is 3.5 mm. The depth of water is about 500 mm to make sure that the disc is completly under water.
Table II gives the cutting parameters for cutting stainless steel and aluminium at atmosphere and under water. Obviously there is a great difference between the speed of rotation at atmosphere and under water because of the great loss of power in the water.

The rate of abrasion is an indirect definition of the cutting speed. It was impossible to realize this extremely low cutting speed needed. The resulting cutting speed when cutting a sheet of stainless steel of 10 mm thickness is about 0.4 mm/min, when cutting aluminium of the same thickness it is about 45 mm/min.
The result of the aerosol measurements by underwater grinding of stainless steel and aluminium is that an emission of particles could not be measured even with a condensation nucleous counter.

2. Analysis of the cutting by-products (B.3.)

Underwater plasma arc cutting produces particles in a wide size range. The small ones are rising to the atmosphere as aerosols. The others are staying in the water and are showing a certain behavior depending on their size and density. During the cutting process there is a cloud of particles around the place of cut. The large particles settle down to the bottom of the basin at once. Most of the suspended particles will agglomerate with increasing of time after the end of cutting and then they will settle down. The cutting experiments for analysing the suspended and sedimented particles were done in a small bassin to rise the concentration of soluted substances. All tests were done in normal drinking water of a waterdepth of 100 mm. The water was filtered by a 0.2 µm filter cartridge before use. The bassin was cleaned completly after every cutting test and filled with fresh water again.
The following materials were used for the cutting tests:

- stainless steel (1.4301 equiv. to AISI 304)
- mild steel (St 37 equiv. to Fe-360 B)
- cladded carbon steel
- aluminium

The used cutting parameters for mild steel, cladded carbon steel and aluminium are given in table III. For stainless steel see table I.
The measurement of the pH - degree and the conductivity gave the results shown in table IV. In general there is a little decreasing of the pH - degree and a rising of the conductivity.
The analysis of the suspended particles followed immediately after cutting to avoid the agglomeration of the particles. There is a high level of suspended material when you cut aluminium compared with the other materials. The reason for the differences between the results of stainless steel and mild steel is the high dross attachment when cutting mild steel.

Figure 1 and Figure 2 are showing the results of the sieve size analysis of the sedimented dross, picked up from the bottom of the cutting tank.

Table I: Cutting parameters
for underwater plasma arc cutting of stainless steel

Underwater Plasma Arc Cutting - Cutting Parameters									
Material: stainless steel (1.4301)						Waterdepth: 100 mm			
Parameter	Material thickness								
	20 mm			40 mm			60 mm		
Position:	g	h	o	g	h	o	g	h	o
Current: (A)	250	250	250	420	400	400	450	450	450
Voltage: (V)	120	130	130	130	130	130	130	130	130
Nozzle: (mm)	2.5	2.5	2.5	3.5	3.5	3.5	3.5	3.5	3.5
Feed: (mm/min)	700	680	700	330	300	300	200	180	180
Aerosol mass: (g/m)	0.2	0.3	0.3	0.5	0.6	0.6	1.3	1.3	1.4

Table II: Cutting Parameters
for underwater grinding of stainless steel and aluminium

GRINDING - Cutting Parameters								
Material		Press. (bar)	Flow rate (nm^3/h)	Rotation (1/min) max.	min	Feed (mm/min)	Infeed (mm)	Abrasion (mm^3/min)
stainless steel	atm.	6.7	73.5	7880	6880	50	1	150
	under water	6.5	72.3	2450	2200	10	0.5	15
aluminium	atm.	5.8	86.7	8100	5100	500	2.3	3450
	under water	6.3	75.6	2500	2200	500	1	1500

Table III: Cutting Parameters - plasma arc cutting

Thickness (mm)	Nozzle (mm)	Current (A)	Voltage (V)	Cutting Speed (mm/min)
Material: mild steel (St 37)				
20	2.5	250	120	380
40	3.5	420	130	300
60	3.5	420	130	150
Material: cladded carbon steel				
18	2.5	250	120	600
Material: aluminium				
10	2.5	250	125	500
20	2.5	250	120	330
25	2.5	250	120	300
52	3.5	420	130	300

Table IV: pH degree, conductivity before and after plasma arc cutting under water. Sedimented and suspended particles.

Thickness (mm)	pH degree before	pH degree after	conductivity (µS) before	conductivity (µS) after	sedimented (g/m)	suspended (g/m)
Material: stainless steel (1.4301)						
20	7.90	7.46	420	500	849	1.0
40	7.90	7.49	430	600	2300	12.0
60	7.97	7.54	460	610	3160	5.9
Material: mild steel (St 37)						
20	8.05	7.82	450	520	606	2.5
40	8.01	7.74	450	530	617	5.2
60	7.95	7.62	450	520	1569	3.0
Material: cladded carbon steel						
18	7.86	7.73	520	560	600	7.7
Material: aluminium						
10	7.91	7.75	450	490	109	12.8
20	8.01	7.53	450	570	336	23.8
25	8.01	7.57	450	620	431	51.2
52	7.91	7.46	450	550	694	22.5

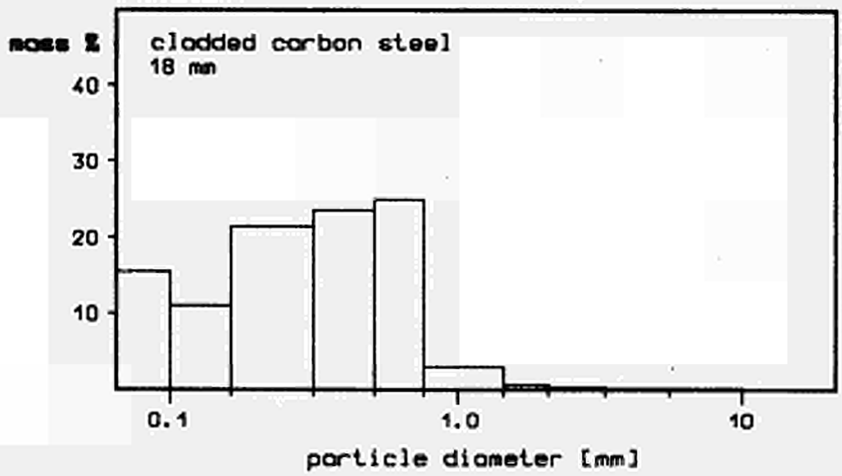


Fig. 1: Sieve analysis of the settled kerf material of cladded carbon steel

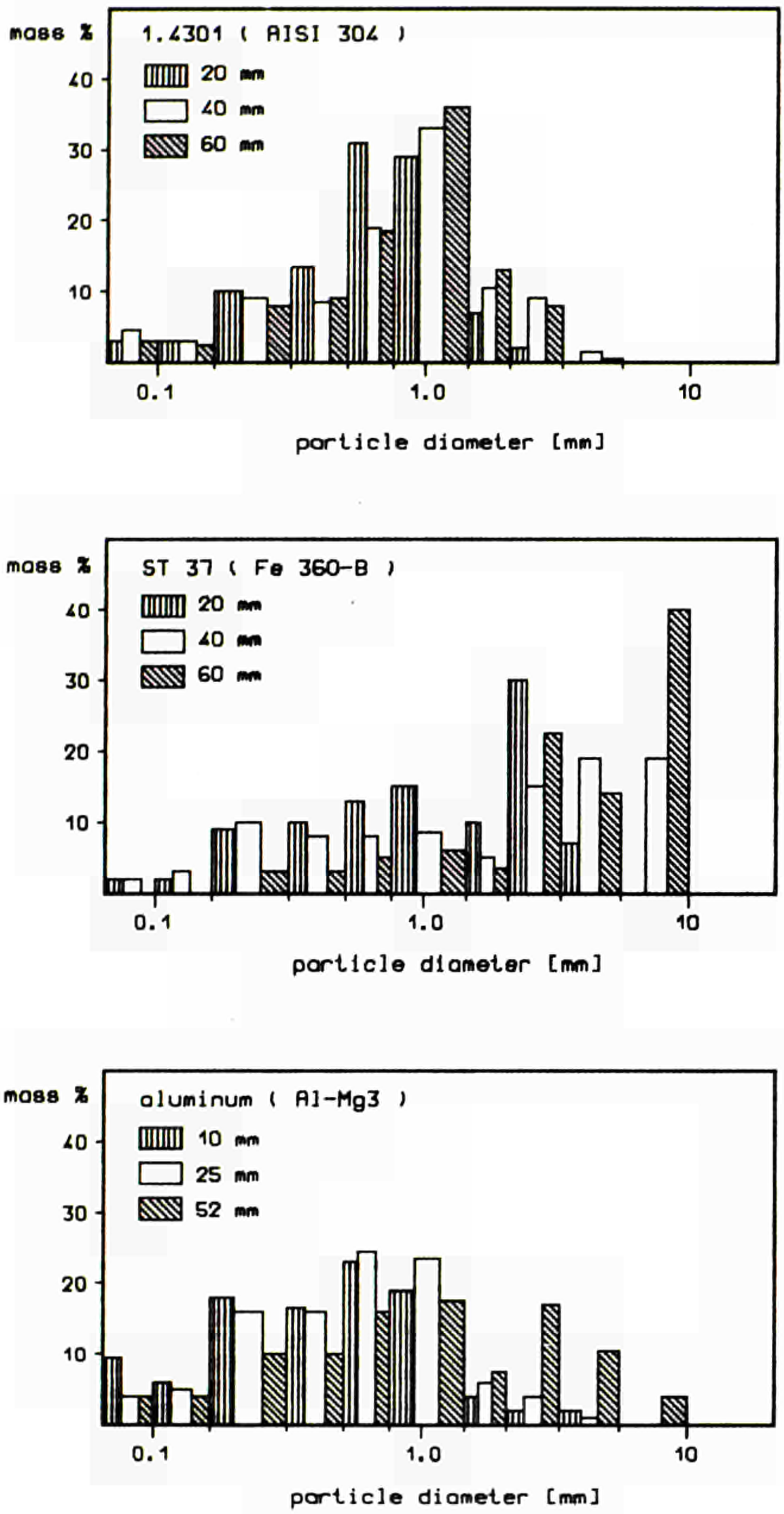


Fig. 2: Sieve size analysis of the settled kerf material

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 - January 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.

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. 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 1987, was connected with three main aims, in relation to the topic of work programme package B1, B4, B5 due to testing opportunities.
Cuttings of mockups of reactor complex shapes, such as connecting vessel shell, variable thickness tank, thermal shield, constitute the first part of our work.
The second one concerns, during the cutting of stainless steel parts, the concentration measurement of the different gases produced, the granulometric distribution, the nature, the concentration measurement of the emitted aerosols.
The third part deals with cuttings on radioactive materials and enabled to determine the activity balance of secondary emissions, the estimate of cutting edges effects, the proved presence of caesium enrichment in aerosols, the interest of a correct collecting device at the source and at last the efficiency study of an electrostatic prefiltration.

. Progress and Results

1. Cutting of mockups of complex shapes (B.1.)

A reactor connecting vessel shell, 4 mm thick, was cut (see Fig.1). This test enabled to show the feasability and repetitiveness without visibility of the nozzle cutting by the interior. The torch axis was put perpendicularly to the nozzle axis.
The second application was the cutting of a reactor tank sheet, whose thickness ranges from 15 to 50 mm. A window was cut straight into the sheet (see Fig.2). The last point concerns a thermal shield made up of five sheets of different thicknesses and 10 mm apart. The cutting was possible on a thickness of 62 mm corresponding to 32 mm of steel (see Fig.3).

2. Analysis of the aerosols and gases (B.4.)

The following results were obtained after cutting stainless steel (Z6 CNDT 17.12) sheets, 1500 mm long and 70 mm thick. The research by neutronic activation of the main elements contained in the aerosols was made.
The aerosols have a log normal granulometric distribution centred to an average diameter of 0,1 µm. The mass concentrations in the duct are twenty to thirty times lower during the under water cutting in relation to the open air cutting (see Fig.4). The aerosols chemical composition is close to the one of cut steel (see Fig.5).
Furthermore, as far as gases are concerned, with a plasma gas such as argon, one gets hydrogen emissions included between 14 and 20 % in volume. The nitrogen monoxide concentrations are included between 14 and 20 ppm at the source. The ozone concentrations are of about 10^{-3}ppm.

3. Cutting on active, contaminated materials (B.5.)

In order to validate in actual conditions the results of the above chapter 2, some radioactive samplers were cut. These samplers were taken from the Triton reactor liner and from the primary circuit pipes of the Rapsodie reactor.

The aerosols measurements were obtained owing to the experimental device shown in figure 6. This experimental device allows to measure the concentration (sampling filters) and the size distribution (initial and diffusional spectrometer)of the aerosols produced.

The measurements of the ventilation flow rates, the renewal rate and the optimised position of the inlet and exhaust ducts (for correct captation at the source) inside the containment were obtained owing to the experimental device shown in figure 7.

The helium tracing by injection allows to evaluate :

- The ventilation flow rate by injection in the exhaust duct.
- The renewal rate of the containment by injection in the inlet duct.
- The renewal rate just above the water and thus the efficiency of the captation device by injection at the edge of the experimental tank.

The under water cutting by plasma arc torch on radioactive sheets ($\simeq$ 0,7 Bq g-1 and 1900 Bq g^{-1}) enabled to underline the following points :

- The aerosols concentration in the exhaust duct, with a flowrate of 1000 m^3 h^{-1} varies from 1 to 4 $mg.m^{-3}$.
- The volume activity in 137 C_s varies from 380 to 900 $Bq.m^{-3}$.
- The aerodynamic mass median diameter of the aerosols is of about 0,2 μm.
- The deposited slags and aerosols represent about 85% and 0,1% of the removed mass.
- The radioactive balance showed clearly :
  - . The transfer of an important part of activity into water during immersion.
  - . The difference of behaviour of the various radioisotopes with an important cesium enrichment in the aerosols.
  - . The ratio between volatile form and aerosols in cesium is less than 1% at the level of the sampling point upstream of the prefilter.
  - . An effect on cesium of the thermally affected zones by the cutting.
  - . The interest of a well designed collecting device at the source, which avoids the scattering of secondary emissions inside the containment.
  - . The efficiency of the electrostatic filter used
    The latter is superior to 90% for the radioisotopes which are involved.

This type of prefilter ensures the protection of the HEPA filters put downstream on the installation.

Note: All dimensions are in mm.

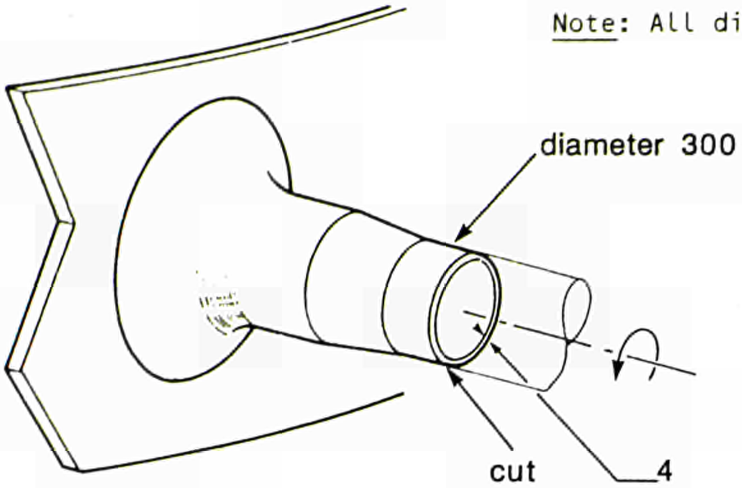


Figure 1 - HORIZONTAL CONNECTION ON VESSEL

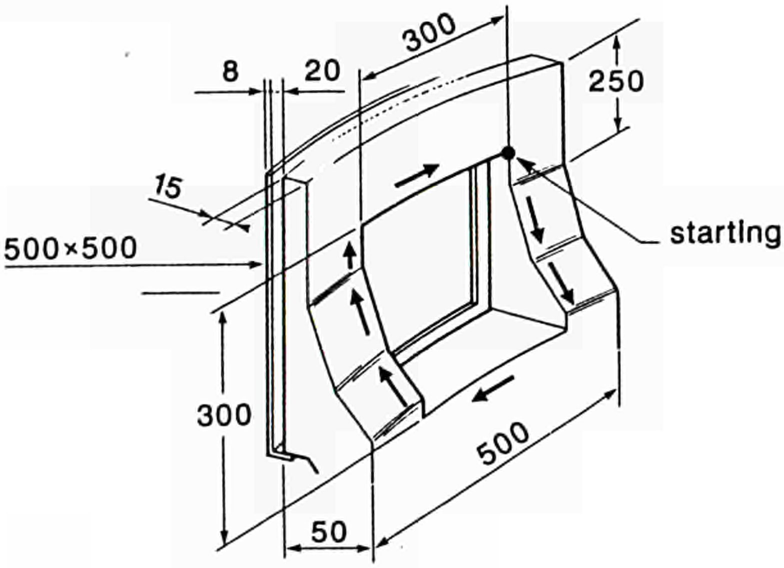


Figure 2 - VESSEL WALL WITH VARIABLE THICKNESS

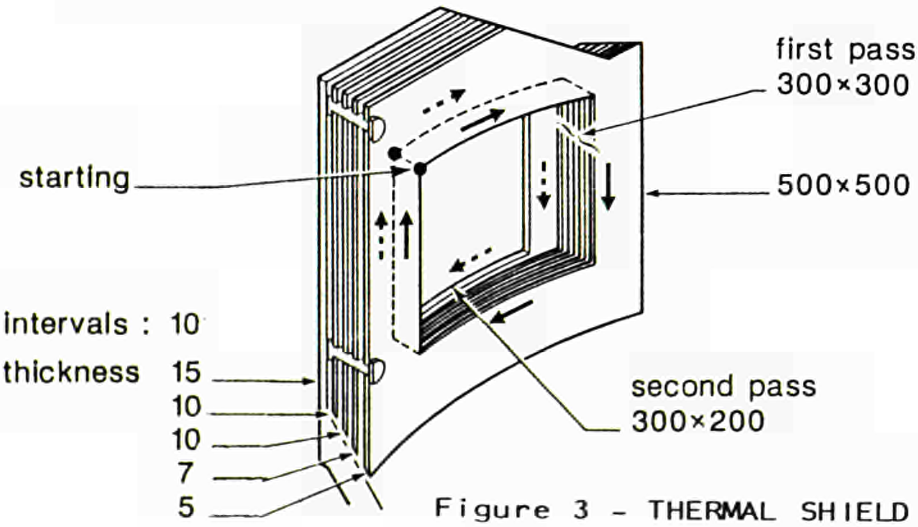


Figure 3 - THERMAL SHIELD ASSEMBLY

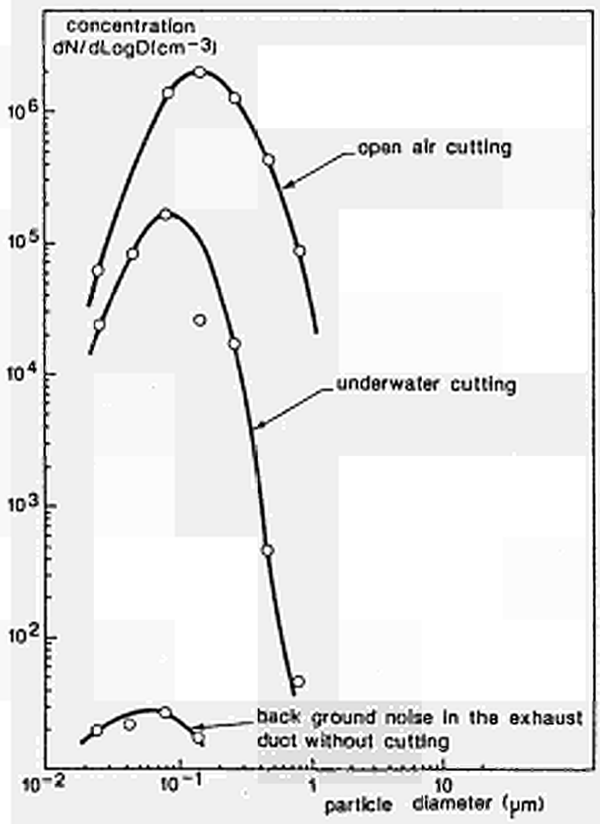


Figure 4 - GRANULOMETRIC DISTRIBUTION IN THE EXHAUST DUCT

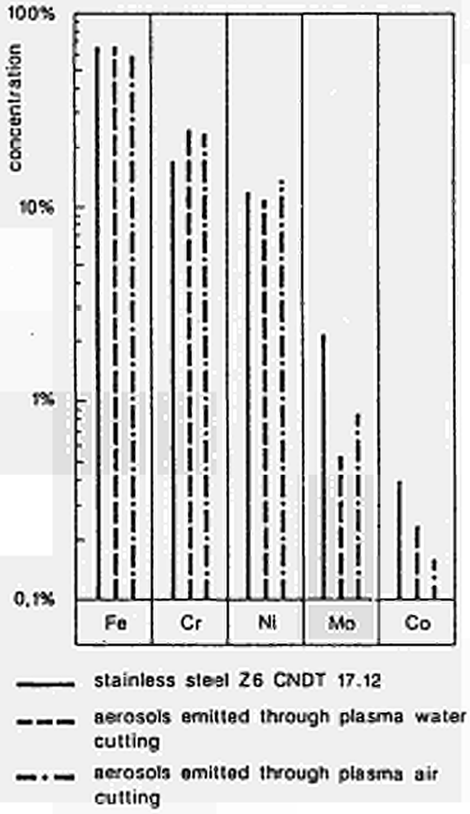


Figure 5 - CONCENTRATION OF THE EMITTED AEROSOLS

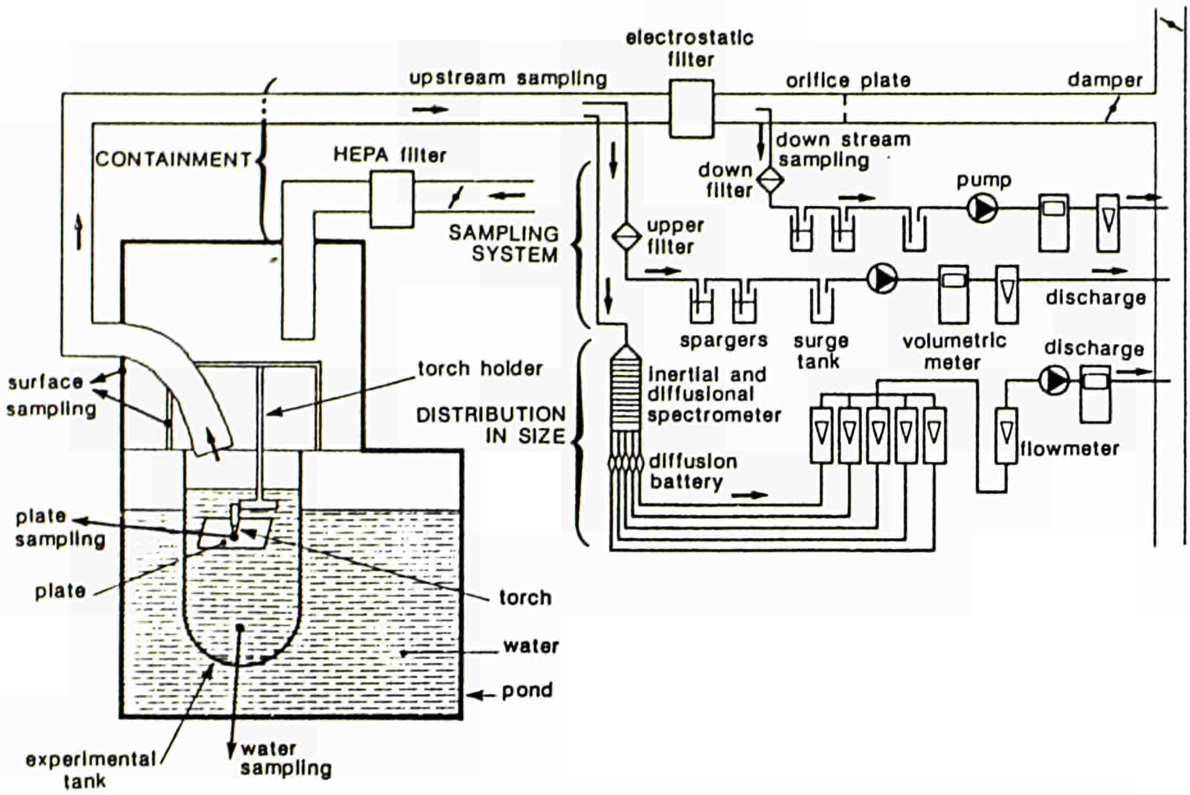


Figure 6 – DIAGRAM OF THE EXPERIMENTAL ASSEMBLY FOR AEROSOLS MEASUREMENTS

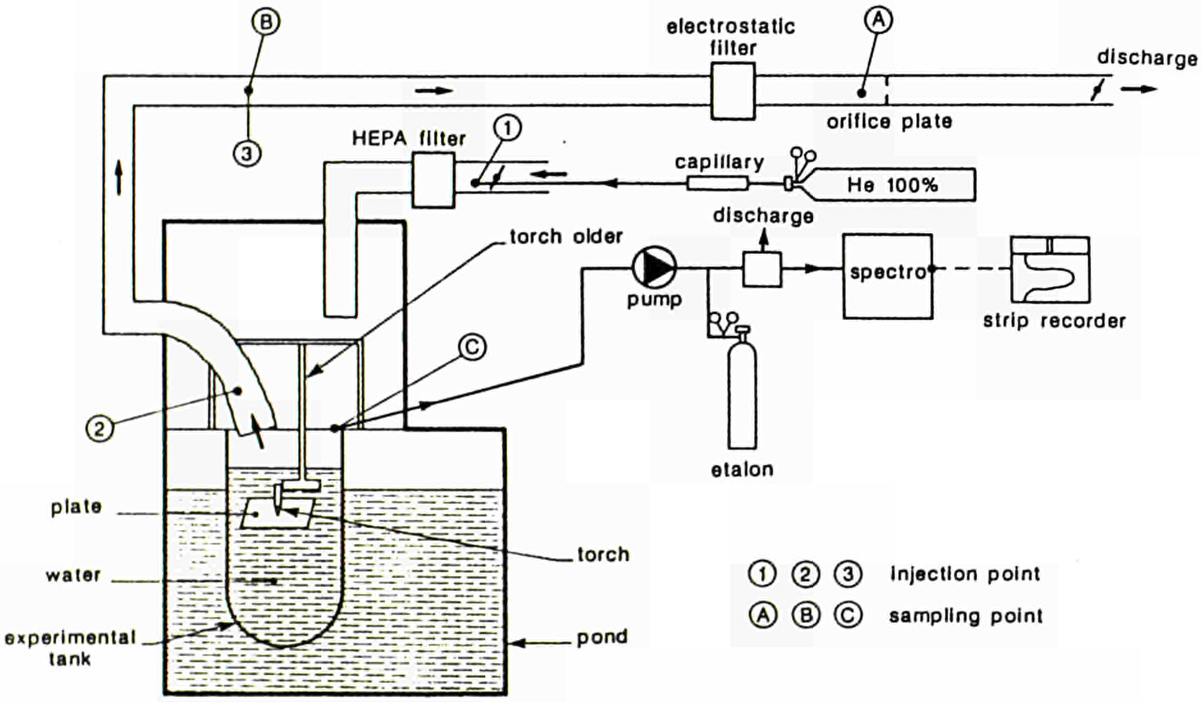


Figure 7 – DIAGRAM OF THE EXPERIMENTAL ASSEMBLY FOR HELIUM TRACER

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 and Obtained Results

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

An extensive survey of commercially available manipulator and control systems has been completed. In parallel with this a design specification has been written outlining the potential requirements of the manipulator.

Arrangements have been made for the hire of the most suitable system in conjunction with a related training course. A good working relationship has been established with the manipulator supplier regarding the development of their product.

An outline test programme has defined major areas for development. The main problems arising in the areas of material, component selection, and restrictions on the overall dimensions of the manipulator arm.

Progress and Results

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

A survey of all the major manipulators and relevant control systems has been completed outlining the product capabilities and facilities currently available from the industrial manipulator The most prominent of these are as follows:

The physical restraints on the manipulator design may require the motors and gearboxes to be mounted within the arm, hence producing a 'clear' profile. This has been taken into consideration for the hire of a suitable manipulator as well as the need to be able to reach anywhere within a hemispherical operating envelope.

Typical speed of movements in a nuclear environment are less than required for general industrial applications. The end effector velocities required in general industry are in the region of 2 m/s compared with the 0.1 m/s required in the nuclear applications. The maximum accelerations are reduced accordingly which minimises problems of high inertia forces associated with industrial manipulators. Due to man control, limited number of repeatable operations, and risks of collision in nuclear applications, high speeds should not be required.

Industrial manipulators are used for a variety of tasks, some of which have a very high degree of accuracy and repeatability (of the order of $\pm$0.1mm). This degree of accuracy should not be necessary for a nuclear manipulator required to perform cutting, grinding and general cleaning where an accuracy of $\pm$ 1mm should be adequate.

Many materials and components cannot be used in a radiation environment. This is due to the molecular structure of plastic, rubber, grease and glass degrading within a radiation field, some metal alloys are also affected. This will affect the power and control cables, lubricants and components of the motor shaft encoders.

The control system required for a computer controlled, remote manipulator in a nuclear envoironment should be similar to that of an industrial robot to minimise costs. The major deviations in design of the control system being related to the protection against radiation damage and provisions of unique control facilities such as real time joystick control. Radiation protection of the control system is provided in two ways. Firstly the control unit is to be situated outside the radiation environment, secondly replacement of unsuitable components and materials, such as encoders and cabling with nuclear "hard" equivalents.

The manipulator is required to operate in several modes; mainly manual control with multi axes joysticks, requiring maintenance of tool geometry to the workface. The control system will have to be positioned out of the radiation environment, this will require a length of control cable of 80 to 100m. This will be facilitated by a reeling drum feed system.

2. SETTING UP MANIPULATOR SPECIFICATIONS (B.2)

The outline specification for a fully programmable nuclear manipulator has been defined. A criteria of this study is that the control system should be a standard commercially available unit as designed for use with manufacturing robots. The mechanical arm would, by necessity, be of a design to cope with the conditions likely to arise, but commercially available components are to be used and where possible would complement the control system. This will achieve the objectives of this study in using commercially available equipment to produce a manipulator at low cost.

The factors that have been considered in the manipulator specification are gained from experience gained on the WAGR decommissioning project, and other nuclear power and fuel reprocessing projects. The expected radiation doses are Beta and Gamma Radiation at 1 GY/hour with the total dose of the project at $3x10^3$ GY. It is anticipated that in other projects the requirements could be upto 10^6 GY and may also involve Alpha contamination.

The manipulator could receive contamination from:- direct radiation from spent fuel, activated equipment and from particles created from the machining involved in decommissioning.

The operating conditions expected would be at 40-150°C at a pressure of slightly below atmospheric (-50mm H_2O).

Decontamination of the manipulator is an important factor, this will be facilitated by gaitering, minimising of particle traps, swabbing, electro polishing and ultrasonic techniques.

Consideration is being given to a payload of 20 to 40 Kg (depending on reach) for the manipulator to be effective as it is anticipated proven hand held tooling as employed on existing manipulators will be used.

In decommissioning, the manipulators overall size has to be considered, due to the limitations of access to the reactor core. This will generally be specific to an individual project and the design of the manipulator will be tailored to suit.

It is required that the risk and consequences of collision in nuclear applications are small, thus the maximum tool velocity will be 0.1 m/s.

The manipulator will be part of a system that is capable of positioning the end effector at the work face. Reach requirements will depend largely on the specific project, but for the sake of this study and to conform to existing equipment, a reach of 2.5m seems appropriate.

To reach every part of the working envelope and to maintain tool attitude, movements in a full six axes will be necessary. These compromise three primary axes giving real world geometry and three secondary axes for the wrist/tool motions. The most appropriate configuration would be the "elbow and arm" as shown in fig. 1.

It should be possible for the robot to have a repeatability in the order of ± 1mm, which would be sufficient for tool operations of welding, disccutting and gas cutting.

For ease of maintenance the manipulator should facilitate the quick exchange of sub assemblies, thus reducing the man dose uptake. This could be feasible with a modular assembly of the manipulator arm.

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

A suitable manipulator and control system has been hired. A training course in programming has been attended and a test schedule of work will be undertaken shortly so that the greatest operational experience can be gained within the hire period.

4. PREPARATION OF AN APPROPRIATE PRELIMINARY MANIPULATOR DESIGN (B.4)

The specific requirements for a nuclear manipulator have been considered and have highlighted areas for further investigation. The main problems are with material and component selection within the control system, namely the glass discs in the encoders, the shielding for the cabling and the distance control signals have to be transmitted. These problems can be overcome with the use of nuclear "hard" components and with suitable amplification and filtering of control signals.

Decontamination of the manipulator is an important aspect, this will be achieved through suitable gaitering and removal of all particle traps with particular attention being paid to the wrist joints.

5. ASSESSMENT AND MODIFICATION OF CUTTING TOOLS. (B.5)

Initial consideration has been given to the specific tools required for decommissioning and to the particular areas that require modification, namely to the control of the tool from a remote system via a reeling drum feed system.

6. DESIGN STUDY OF REMOTE-OPERATED TOOL CHANGE FACILITIES (B.6)

Consideration has been given to remote tool change facilities. This will necessitate a library of tool locations and orientations, and a cartridge system for holding redundant tooling. Attention is being given to remote power links to the tool through slip rings and bayonet fittings.

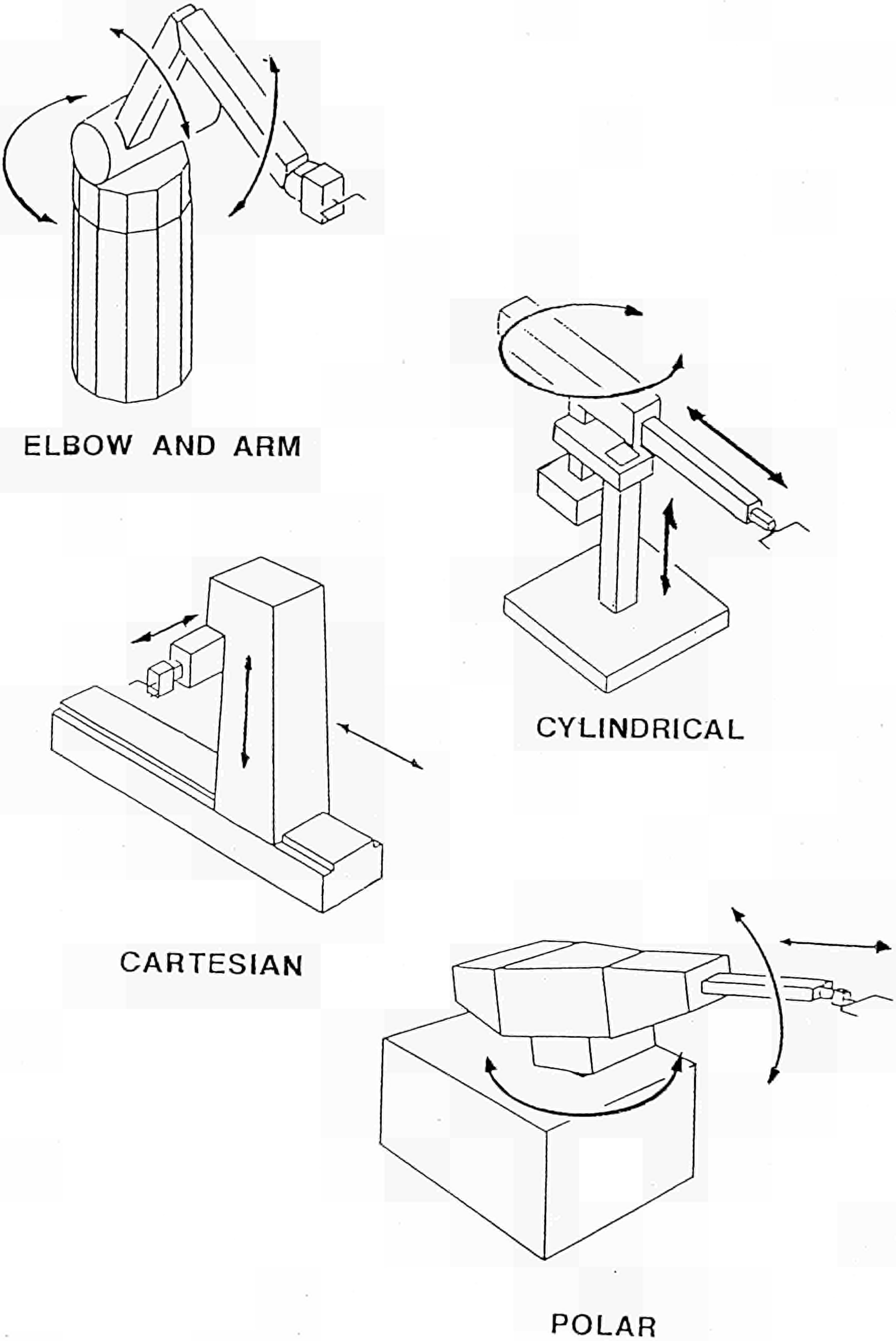


Figure 1: Types of manipulator

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 - December 1988
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

In the reporting period a modified underwater tv-camera was used for optical cutting process control. Furthermore, an image processing system with a CCD-camera was arranged and special software has been developed for workpiece recognition. The software structure is designed and some modules of system software have been developed. Inductive sensors were being tested for automatic positioning and special ultrasonic sensors for collision avoidance. Furthermore, a light barrier was selected in order to registrate the attrition of the abrasive-wheel. For cutting control, a special photoelectric sensor and a tactile sensor were being tested. The adaptive control mode was specified and the parameters of gas flow rate and rotation speed were subjects of investigations. An ultrasonic sensor for thickness measuring was being tested for the application.

Progress and Results

1. Workpiece Recognition Underwater (B.1.)

After all important aspects have been considered, the tv-camera system now performs a satisfying optical process control and workpiece recognition. Problems in observation, caused by a lot of gas bubbles leaking out of the cutting-off machine, have been solved after reconstruction and the insertion of sealing rings. In addition to the tv-camera system with a conventional image pick-up tube, a new image processing system with a CCD-camera was mounted and adjusted for workpiece recognition underwater. Fig. 1 illustrates the structure of the designed and assembled image processing system. The frame-transfer-CCD-camera with 384 (H) x 576 (V) picture elements (221184 pixels) is of a very small size (30 mm x 40 mm x 30 mm), stable image geometry, high sensitivity, it has no lag, no sticking, high immunity against vibration and shock, immunity to any disturbance from magnetic fields and practically unlimited life time and maintenance free. The CCD-camera, the synchronizing unit, the power supply and the monitor, all together build an independent system for optical process control and workpiece recognition. The extension with video memory board, image processing board and microcomputer system with peripheric components results in an image processing system where the PPI-boards are holding together all the logic necessary to be interfaced to an external CCIR video source and to sample and display the acquired image data in video realtime. The video memory has a resolution of 8 bits per pixel (256 greyscales) and a capacity of 512 Byte. Depending on the RAM available and the desired acquisition window, the PPI-1 can hold several images simultaneously for reference purpose or event recording. The PPI-2 board holds all circuitry to digitize and sample the video signal plus the necessary control for the PPI-1 board. For the camera system, a program in PASCAL and ASSEMBLER has been developed, which allows for a comfortable image processing and analysis.

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

The hardware of the designed process control system has been completed and the hardware modules were being tested. In the then following phase, the software structure was designed and the program modules were developed. Fig. 2 shows the total software structure for process control. The sensor-aided automatic positioning of the abrasive-wheel cutting-off machine is designed to be carried out by means of a closed control loop with an inductive sensor. This system will be able to position the tool in a defined preprogrammed distance to the workpiece surface. For the application of the inductive sensor, the signal responses were being investigated and the experiments with the sensors result in the following summarized characteristics: high repetition fidelity of sensor signals, small signal deviation between identical sensors, independence of sensor signal and workpiece thickness (1 mm < s < 60 mm), independence of sensor signal and workpiece surface quality (rust, colour, tinder), invariability of the sensor signal when diving in water, small temperature drift, small malfunction if material is located at the side of the sensor.

For collision avoidance, it was decided to use a special version of an ultrasonic immension technique probe. Fig. 3 illustrates the adaption of the ultrasonic sensors at the cutting-off machine and presents the working area. The control of the cutting area is performed with the highest measuring length of 180 mm. Because the actual coordinates of the tool are known, it is possible to calculate the distance but not the direction towards the collision object. To simplify signal processing, it is useful to tune the monitor aperture to the collision measuring range. Echoes inside the monitor aperture range indicate any real danger of collision by an optical or acoustic signal.

3. Control of Cutting Action (B.3.)

To assure high quality cutting, it is necessary to adapt sensors for controlling the cut and the attrition of the abrasive-wheel. A photoelectric sensor or the tactile sensor follows the abrasive-wheel, scanning the gap. On both sides of the abrasive-wheel a light-barrier is mounted (cf. Fig. 4). If the tool does not cut the workpiece as a whole, the tactile sensor gives an electric impulse for registration. To use the photoelectric type of sensor for cutting-through-control the detection range must be studied in the air and underwater by varying the sensor adjustment parameters to gain information about the optimal mounting position and application limits (cf. Fig. 5). The experiments proved a mounting height of about 10 mm above the workpiece surface to be optimal. In this case, it is possible to detect of workpiece connections in the root of the gap up to a maximum depth of about 45 mm.

If the attrition of the abrasive-wheel has reached a defined level, the process control system receives a signal from the light-barrier and after indicating this on monitor, the operator decides on the next working steps.

4. Adaptive Control of Cutting Parameters (B.4.)

The cutting-off machine needs the maximal gas flow rate to reach the idling speed which is controlled by a centrifugal governor, underwater. So, the method applied to control the air flow rate for controlling the r.p.m. of the abrasive wheel is not satisfactory (cf. Fig. 6). It is planned to regulate the feed rate of the tool by contineous measuring of the workpiece thickness. Additionally, the rotation speed must be measured and the r.p.m. will be compared to a limit. If the limiting value does not come up to the estimate, the feed is reduced or stopped until the r.p.m. steps over the defined rotation speed limit.

Tests lead the use of a sensor with a frequency of 0,8-3,0 MHZ. This type gives the most reliable signals if the ultrasonic sensor deviates from its vertical orientation. Additionally, the adjustment of the ultrasonic measuring system doesn't provide any problems. The results of the other tests permit the successful application of the chosen ultrasonic sensor for measuring tasks such as described above.

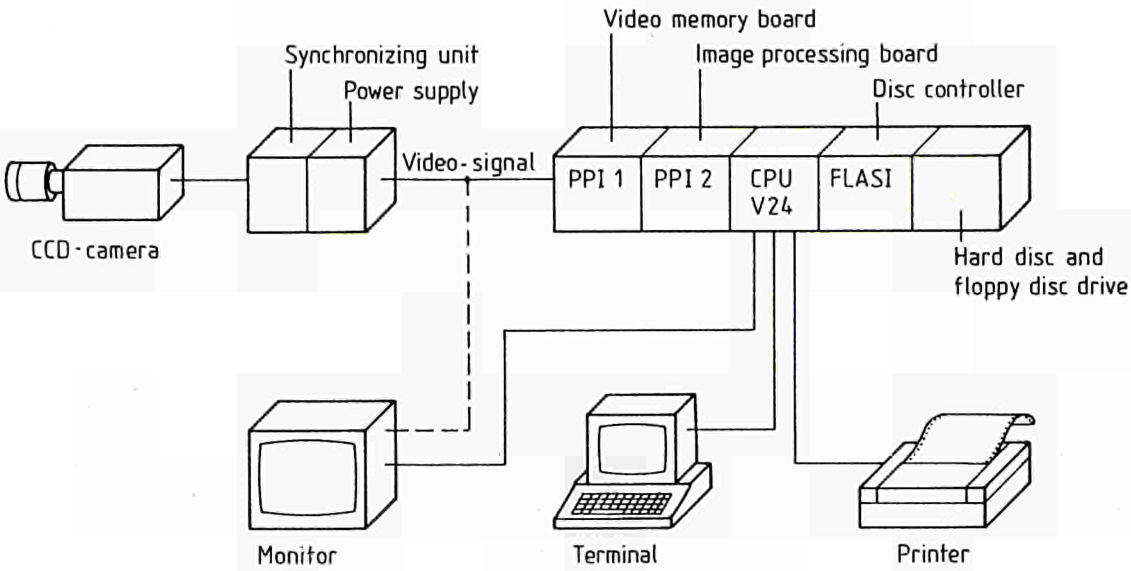


Fig. 1: Block Diagram of the Image Processing System with CCD-Camera

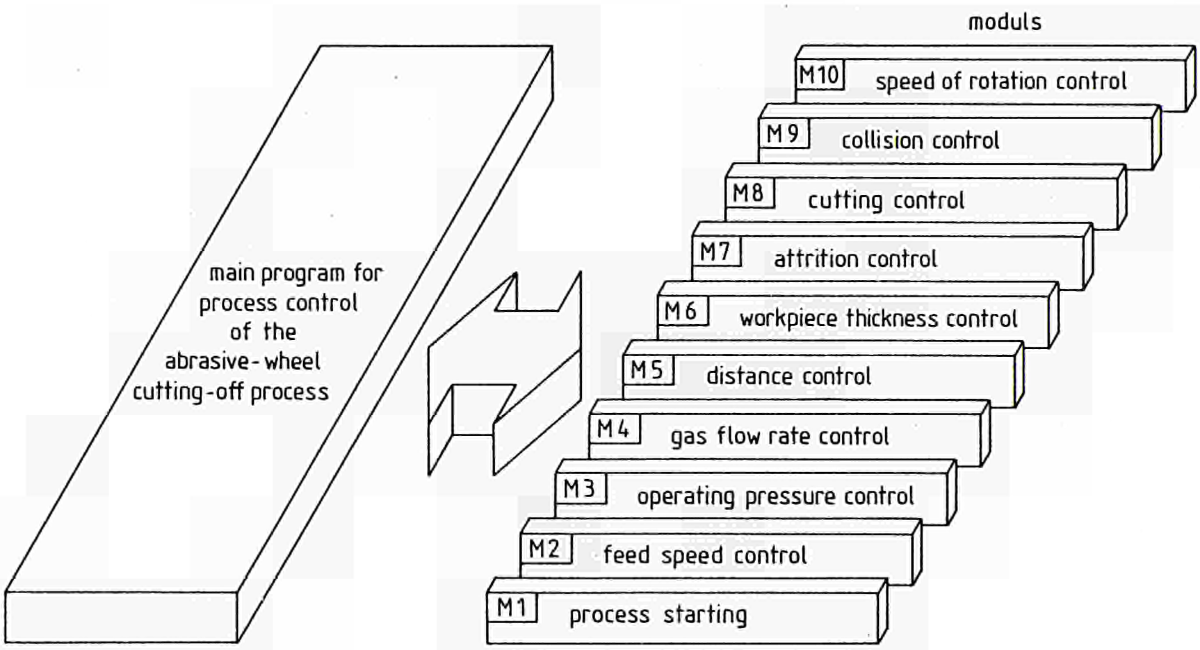


Fig. 2: Software Structure

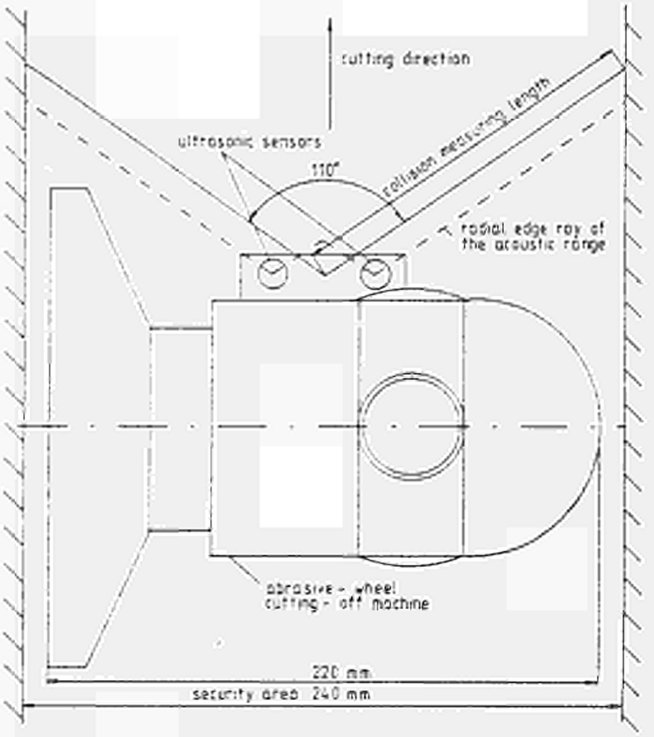


Fig. 3: Schematic Presentation of the Working Area of the US-Sensor

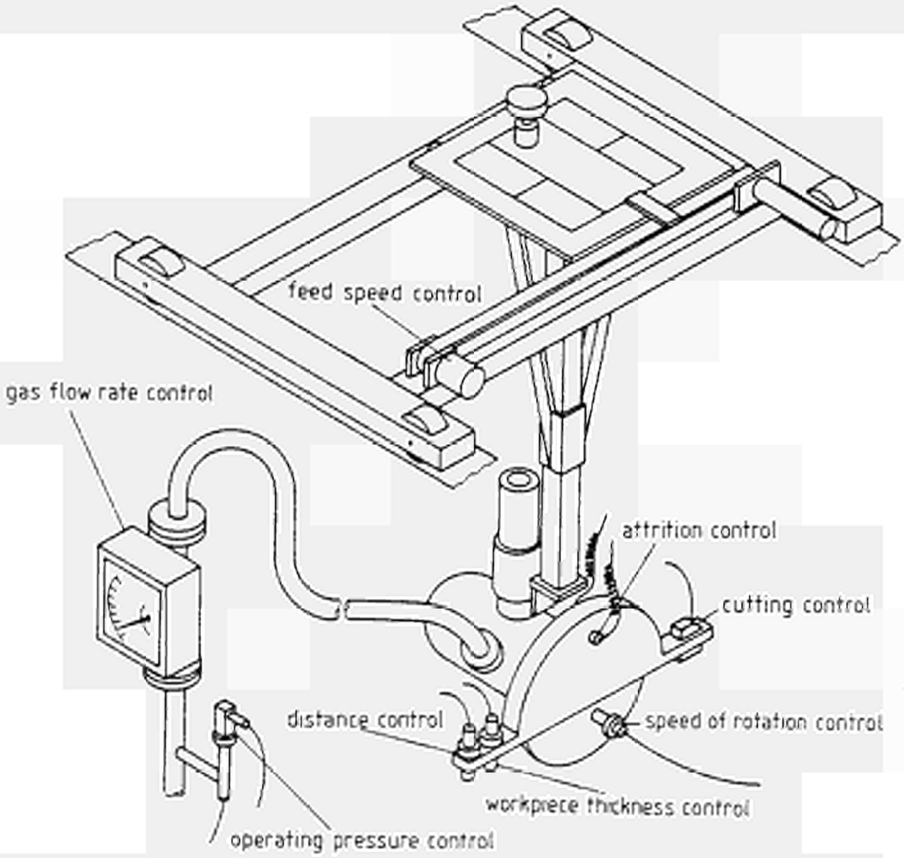


Fig. 4: Sensor Attachment for the Abrasive-Wheel Cutting-Off
Process Underwater

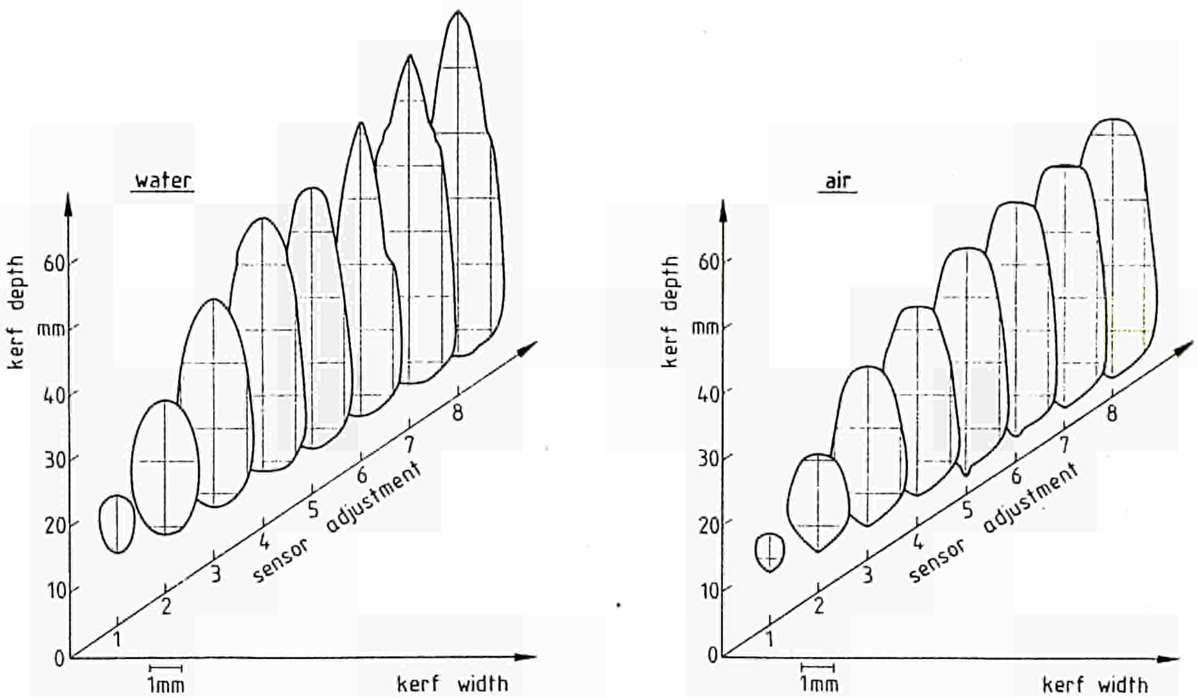


Fig. 5: Working Area of the Optical Sensor for Work Underwater and in Air

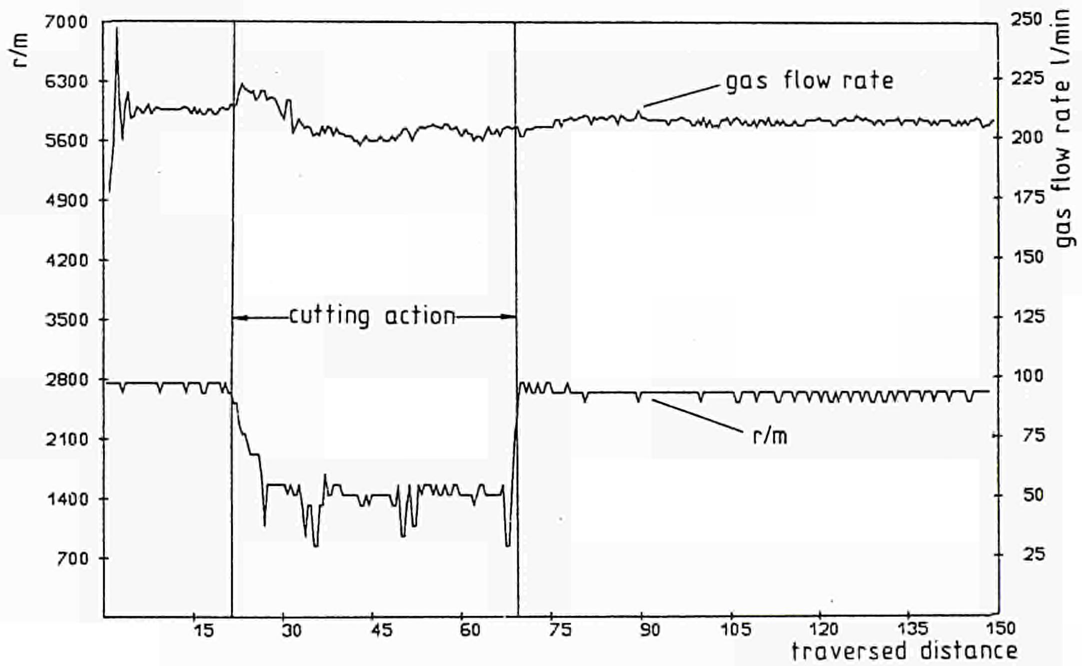


Fig. 6: Gas Flow Rate and Rotation Speed during Underwater Cutting

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

Summary

A comprehensive literature review has been undertaken which has provided useful information for the preparation of the experimental programme and equipment. Mathematical modelling of the microwave and power fields in a concrete block, both steel reinforced and unreinforced, subjected to microwave attack at two frequencies has been carried out and estimates of the likely temperature rise with time obtained.

A method of launching microwaves into concrete has been established from theoretical considerations and from the findings of the literature review. Equipment for laboratory trials has been designed and assembled using an 896 MHz, 25 kW microwave generator. Reinforced concrete blocks, 0.6 m in dimension and representing the concrete in a Magnox reactor biological shield, have been attacked at different power levels and the surface removed to the depth of the reinforcing steel (100 mm).

Outline proposals for the design of a remotely operated prototype microwave machine have been prepared.

Progress and Results

1. Literature review (B.1)

References relating to the subject fall into three main experimental areas: (1) fissuring/cracking of concrete (UK 1964-69), (2) road repairs (USA 1973-84, France 1986, USSR 1979), and (3) general fracturing/ spalling of rock and concrete (Japan 1971-87, USSR 1978, France 1973).

The principal conclusions of the review are:

a) Microwave demolition of concrete has been demonstrated on several occasions since 1964 and is clearly feasible.

b) Nearly all reported demonstrations relate to plain concrete. There is little data about either reinforced or very dense concretes.

c) Disintegration of concrete under microwave attack is generally agreed to be due to internal tensile stresses created by the generation of steam from the water present in a relatively impermeable medium.

d) Attack takes place primarily over the outer 100 mm, which is convenient for stripping off surface layers. This depth of attack will vary with the type of concrete in question, the frequency of the microwaves, the presence of reflections from reinforcement, the rate of traverse of the applicator and the applicator/concrete air gap.

e) Demolition rates of about 1 litre per kWh of applied energy have been recorded for plain concretes, and the context provided by other applications shows that this figure will represent a maximum for the more difficult case of reinforced material.

2. Theoretical analysis - power and frequency (B.2)

The microwave field and associated thermal power in a 0.6 m concrete block have been evaluated using electromagnetic theory for a plane incident wave, allowing for the possibility of a standing wave field. Both steel reinforced and unreinforced concrete have been examined, at microwave frequencies of 896 MHz and 2450 MHz.

The analysis indicates that, for unreinforced concrete, the maximum temperature during the application of microwave power occurs in the first few centimetres below the surface and the temperature profile consists of a series of hot spots which decrease in magnitude with depth. Over a short period of time, before significant thermal diffusion occurs, temperature increases approximately linearly with time and also with power. The near-surface temperature rise is more rapid at the higher microwave frequency but the peak occurs at a shallower depth.

For example, 20 kW at 896 MHz applied to a 0.6 m thick plain concrete block produces a maximum temperature after 3 minutes of 110°C at a depth of 30 mm, whereas at 2450 MHz this temperature is reached in 1-1½ minutes but at a depth of 20 mm. This suggests a higher rate of concrete removal at the higher frequency.

3. Launching technique (B.3)

Attack of the concrete surface from an oblique or from a leaky waveguide is apparent from the literature as the best means of reducing reflected power. An angle of 45° downwards towards the vertical concrete face was selected as the best compromise between the requirements of refraction and reflection at the concrete surface, the propagation of microwaves within the test cavity, and the clearing of debris.

Examination of horn configurations led to the choice of a square end to the waveguide rather than a sawn off (oblique) end or a horn of varying geometry. This provided as concentrated a plane wave as possible. Power reaching the concrete from the horn decreases rapidly with the distance of separation so that a fairly close spacing, consistent with practical manoeuvring of the concrete block, of about 5 cm was chosen. The maximum power density was about 2 kW/cm^2 per kW of applied power which is greater than has been used in most of the experiments described in the literature.

4. Effect of steel reinforcement (B.4)

The theoretical analysis (B.2) was extended to examine the effects of steel reinforcement at a depth of 100 mm below the concrete surface. Calculations indicated that reflection from the steel would increase microwave energy deposited in the surface layer of the concrete, resulting in peak temperatures being reached in about half the time of those in unreinforced concrete and at a slightly greater depth. This should result in a significantly higher rate of concrete removal. It was concluded that the amount of energy deposited in the steel would be very small compared with that deposited in the concrete and that no preferential heating would occur near the steel.

5. Laboratory trials (B.5)

The equipment consisted of a variable-power 25 kW, 896 MHz generator which launched microwaves into a 248 x 124 mm rigid rectangular waveguide approximately 3½ m long. The end of the waveguide passed into a steel box of 1.5 m in each dimension which enclosed the concrete block under test. The generator was protected against power reflection by a water circulator mounted on its launch guide.

The concrete test blocks were of 0.6 m dimension and the mix composition and steel reinforcement were representative of those in a Magnox reactor biological shield. The evaporable water content in the surface layer was 2½ to 3% at the time of testing.

Four reinforced faces on two concrete test blocks were subjected to microwave attack at different power levels. Concrete spalled from the surface at rates and depths dependent on the power level. The relatively small area of the face resulted in an unrepresentative loss of concrete at the edges if the block was attacked other than at the centre of the face. Fig 1 shows the area of concrete spalled after one attack near the centre; concrete was removed to the depth of the reinforcement, 100 mm, at the centre of the crater. Figure 2 shows the same block attacked again just below the position of the first attack; spalling is much more extensive due to failure of the block edges.

Results of the experimental work are summarised as follows:
(a) Reinforced concrete can be spalled by microwave attack at 896 MHz.
(b) The concrete cover can be removed as far as the reinforcement.
(c) At high power densities (>350 kW/m^2) a superficial spall of about 30 mm depth can be achieved.
(d) Removal efficiency when demolishing the cover is about 2 ℓ/kWh regardless of power density.
(e) Removal efficiency for a superficial spall at high power density is only about 0.2 ℓ/kWh.
(f) Power reflection problems can be reduced to an inconsequential level by adequate impedance matching and horn design.
(g) Power reflection from exposed reinforcement is sufficiently low to justify further exploration of the direct attack of underlying concrete.
(h) Power reflection changes during the traversing motions required in demolition work may be appreciable and may require on-load tuning.
(i) Explosive cracking is due to steam pressure generated at a hot spot. Doming of the surface occurs followed by tensile failure. Nut-sized fragments are ejected from the central region with considerable force. Internal temperatures of up to 300°C in the concrete are attained.
(j) Non-explosive internal cracking occurs, adjacent to the main explosion zone.
(k) The depth at which explosive cracking occurs is about 100 mm for low power densities, and the explosion is correspondingly vigorous. At the higher power densities the explosion depth is less, making possible the superficial spalling option of (c) above and also leading to the removal of cover by a series of explosions when employing high power density.

6. Prototype machine (B.6)

There are two options for a remotely-operated prototype microwave concrete demolition or stripping machine. One is to position the microwave generator, on a manipulator, directly at the work face so that the transmission distance to the device for launching the microwaves at the concrete surface (the applicator) is minimal. A major disadvantage of this method is that the generator would be in an unfavourable environment and susceptible to damage and contamination. The other option is to position the generator well away from the working area and to transmit microwaves to the applicator via a long, steerable waveguide; this is the preferred arrangement and is the one considered here.

A proposal for a prototype machine is shown in Fig 3. It represents an intermediate development stage between the small-scale static tests undertaken as part of the current programme and a fully developed machine to operate inside a reactor biological shield. Such a prototype machine would be used to investigate important aspects of microwave power transmission which have been identified as the feasibility and performance of flexible or articulated waveguides, applicator design for optimum concrete removal and minimum power reflection, and the control of reflection by on-line network analysers. It would also provide more information on realistic concrete removal rates, in various orientations, on large panels with reduced edge effects, and on concretes of different composition and embedded metal configuration. Microwave leakage and safety aspects would form an important part of the work.

The proposal illustrated consists of a remote, static microwave generator of high power (at least 60 kW) protected by a circulator, and a steerable waveguide assembly, suitably supported, consisting of rigid and flexible sections capable of transmitting microwaves to an applicator mounted on a simple three-axis manipulator. The estimated cost is £350 k.

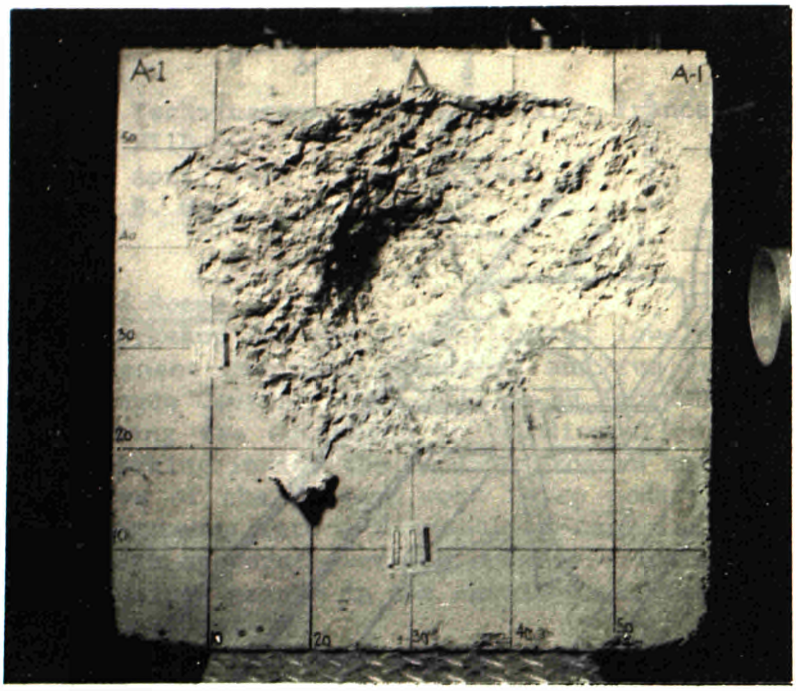


Figure 1: Damage to concrete block after first attack

Figure 2: Damage after second attack. 10 cm downward traverse.

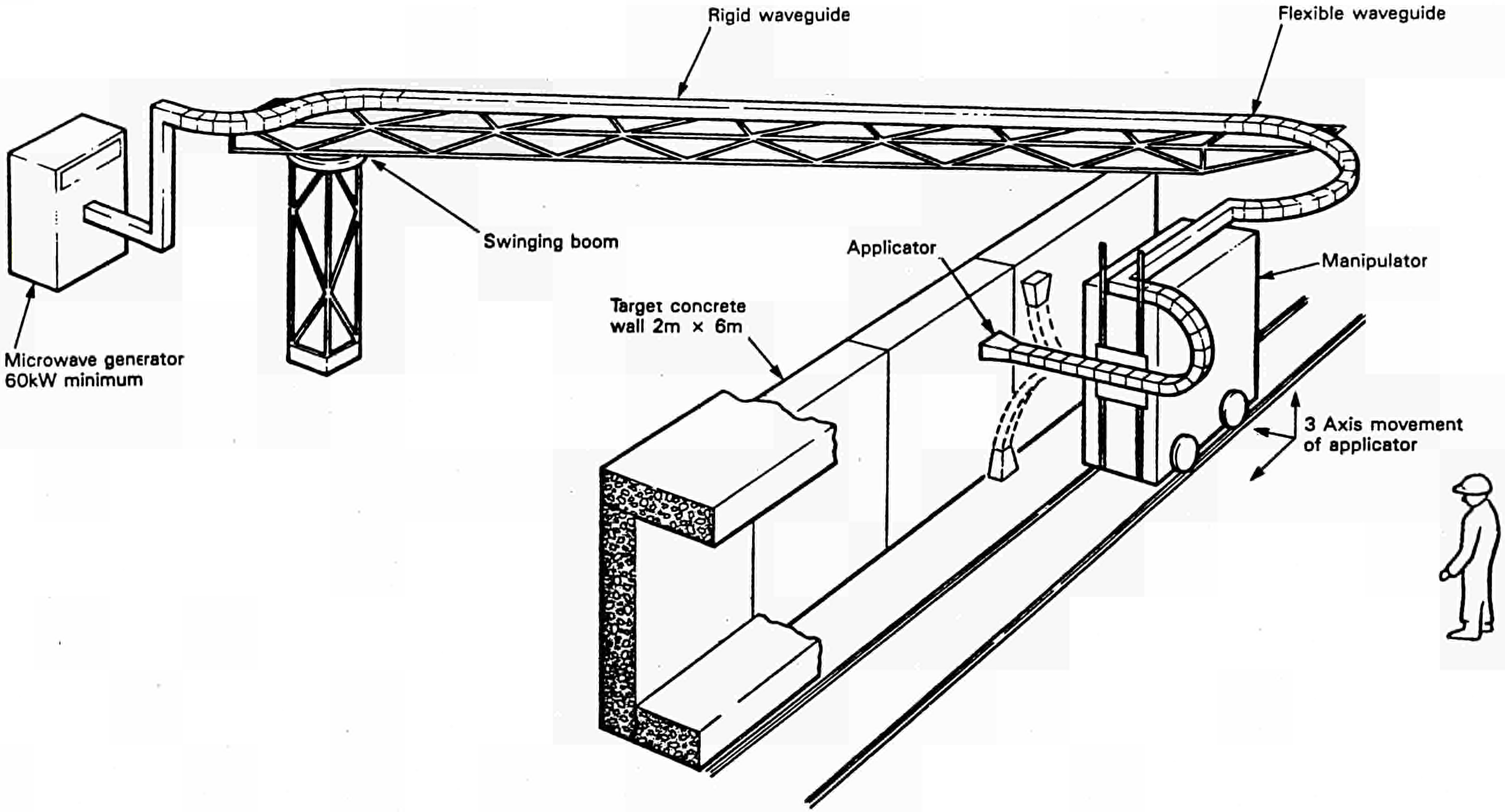


Figure 3: Proposal for prototype machine

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

SUMMARY

During 1987 the remaining detail design work relating to panels has been completed. All the panels have been manufactured and further assembled on the test cell. Mechanical tests have demonstrated the good behaviour of them.

Sealing tests for hatch panel still remain to be performed.

Progress and Results

1. Final project and realization (B.2.)

The as-built phase takes into account the improvements brought to the pre-design to cut down the construction costs or to increase the use capabilities of the new panels.

1.1. Hatch panel (Figure 1,2)

The swiveling panel should be assembled, on an optional basis, onto the standard frame panel (1600 x 1600 mm) with its opening of 1200 x 1200 mm.

The pneumatic jack and the rotating shaft support of the hatch panel are fitted on a plate which may be easily bolted on the frame panel. Tightness is ensured by gasket ; locking in the closing position is provided by two pneumatic toggle-fasteners.

Likewise the sliding model is designed for its fitting on a standard frame panel. This has not been manufactured.

1.2. Manipulator holder panel

The attachment of this panel to the support structure has been improved to get a better take-up of the horizontal axis momentum.

1.3. Wall equipment sealed transfer device (Figure 3)

The transfer device,(connecting tunnel with double sealed door) from LA CALHENE which was previously to be fitted on a special panel, will be from now fitted on a new standard frame panel (800 x 800 mm) with a 550 x 550 mm opening blanked off by a plate secured by clamps. This closing plate fitted with a special flange allows supporting the connecting tunnel.

In this configuration, one of the tunnel plugs may incorporate as an option, tight penetration connectors or wall equipment.

Transfer in glove-box is easily performed via connexion and locking on the tunnel.

Depending on the needs, the frame panel-connecting tunnel-glove box assembly ensures the following fonctions.

a - Tight transfer of equipment in both directions;

b - Removable support to tight penetrating connectors with possible interventions in glove-boxes;

c - Support to wall fitted equipment with possible interventions in glove-box, the containment continuity being ensured.

As far as cases b and c are concerned the standard plug is modified accordingly.

1.4. Biological protection (Figure 4)

The protection system provides for the addition of 4 plates of 25-mm thick lead maximum attached along the cell walls by means of lug angles bolted on the panel edges, and lead strips .

The 800 x 800 mm plates are equipped with inserts for screwing in of eyebolts.

2. Testing of panels (B.3.)

2.1. Hatch panels

The tests have proven that this panel is easy to assemble, as well as the proper operating order of the opening and closing mechanisms.

2.2. Manipulator holder panel

The mechanical tests and the use of standard telemanipulators have resulted in qualifying this panel as a new component of the modular workshop.

2.3. Wall equipment sealed transfer device

The tests have proven that the non telemanipulable plug of the connecting tunnel may easily be modified and adapted for supporting any wall equipment or providing passage for tight connectors.

2.4. Biological protection

The tests have proven that the dimensions and weight of the lead panels do not raise any major difficulty for installation. The stiffness of lug angles with cell panels is sufficient for maintaining the lead plates in position.

3. Operating evaluation of modular works (B4 phase)

The new panels are qualified from the mechanical standpoint for enlarging the use of the modular workshop to teleoperation procedures.

Checks for hatch panel sealing remain to be performed with the cell operating at negative pressure.

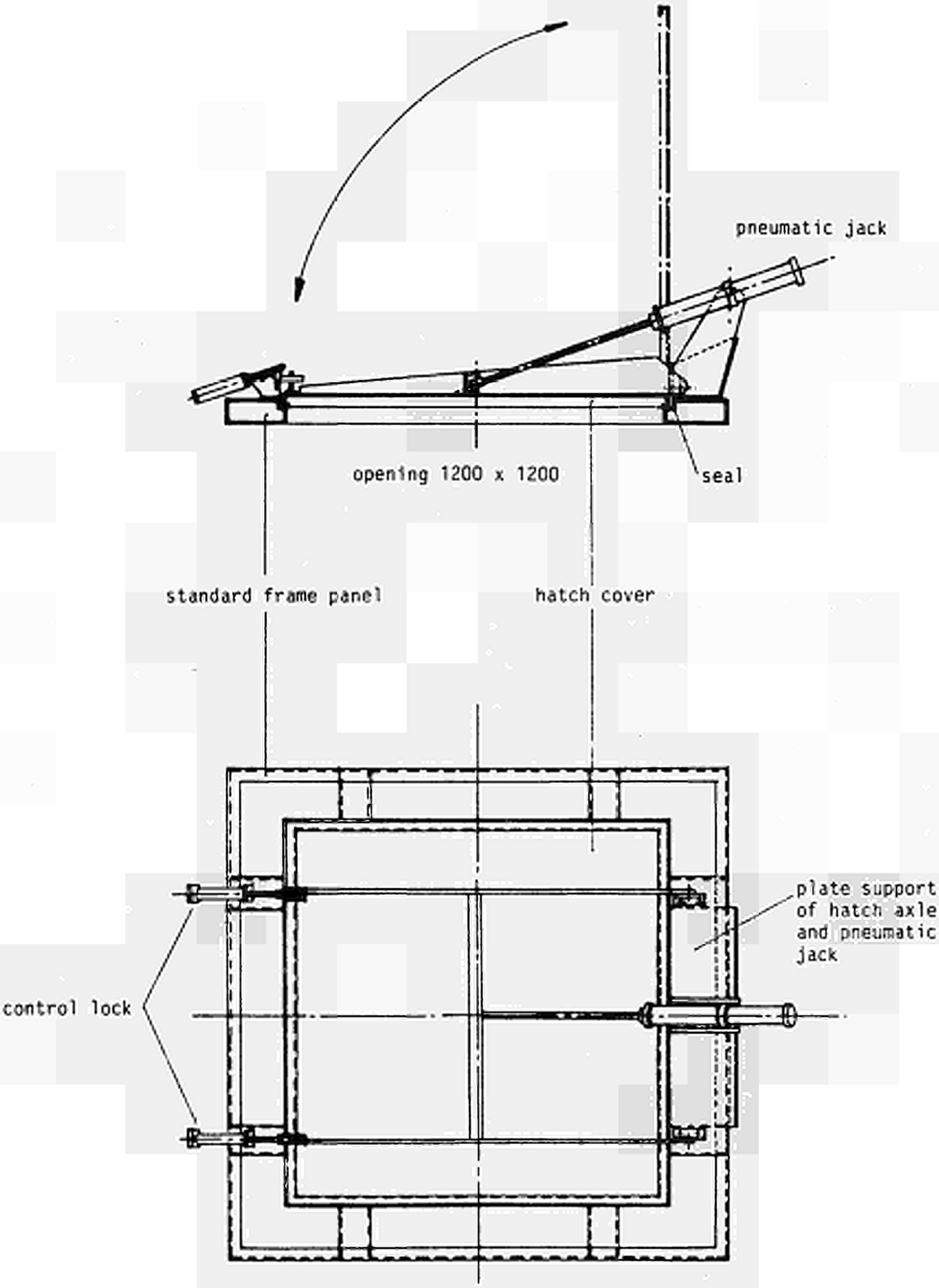


Figure 1. Swivelling panel

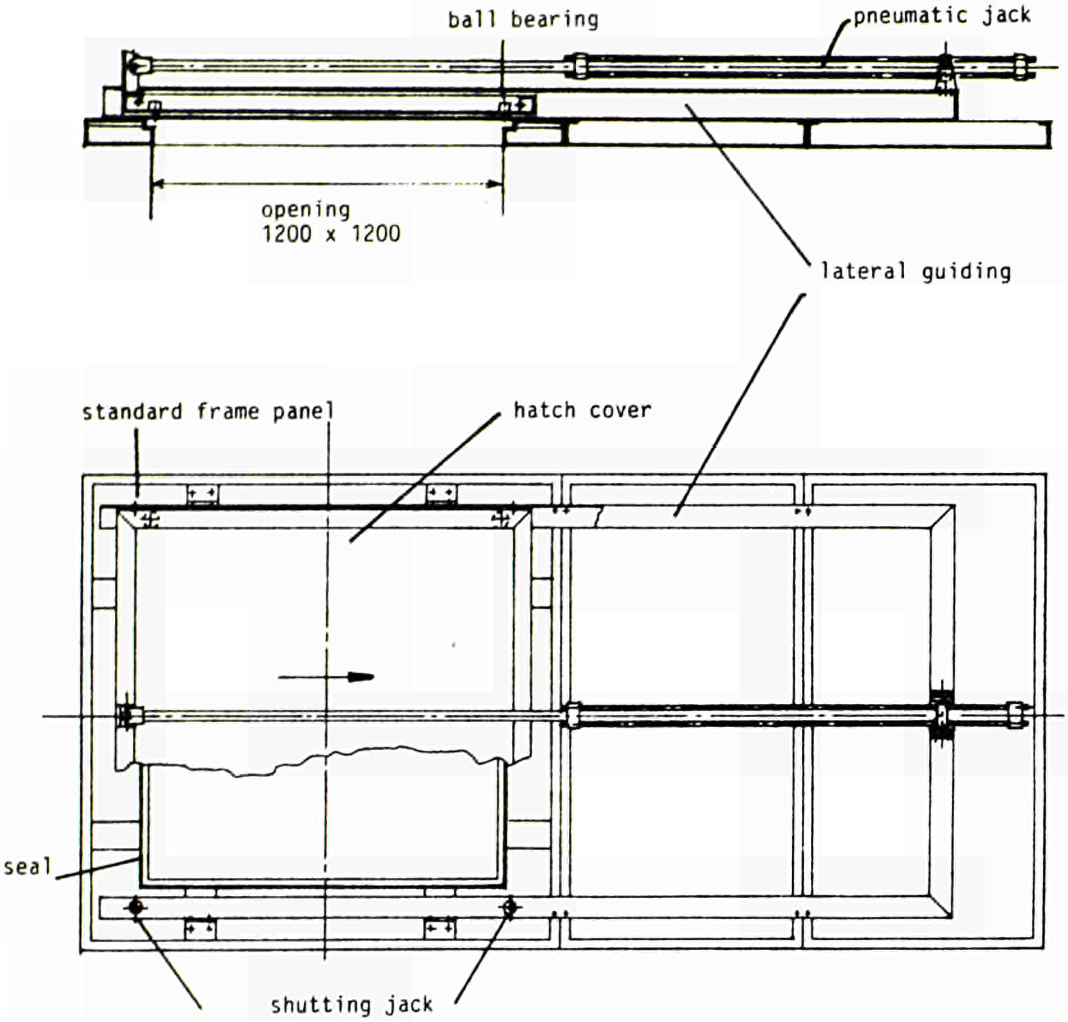
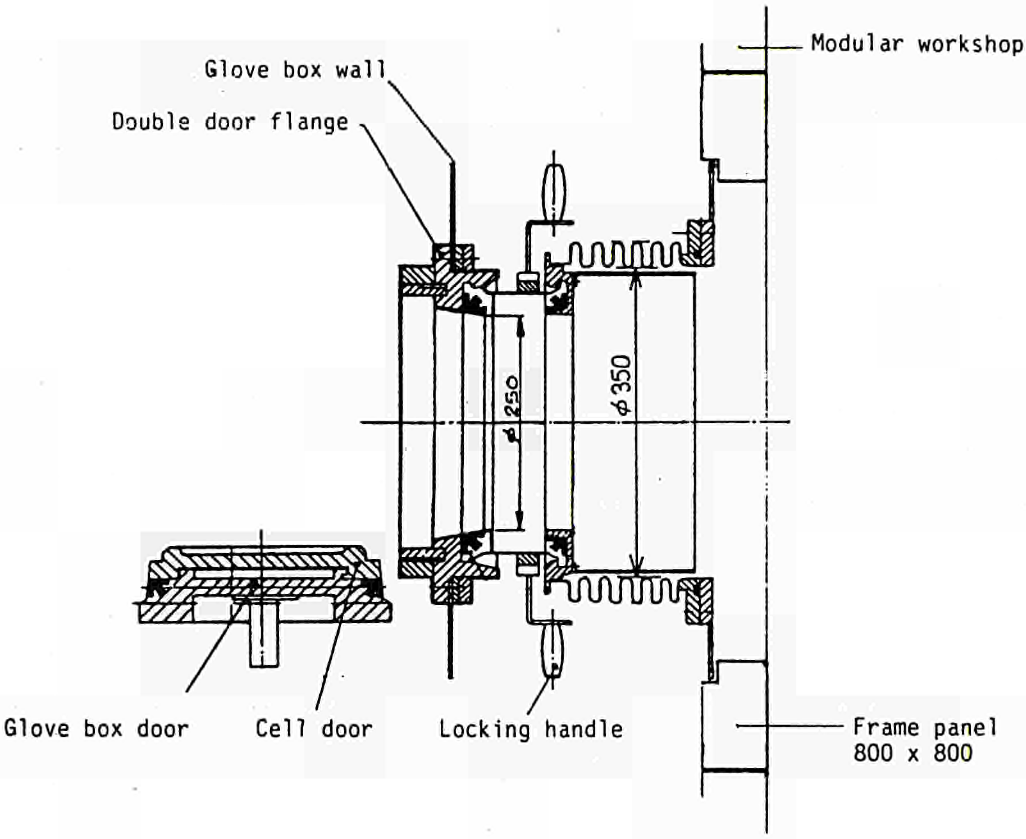
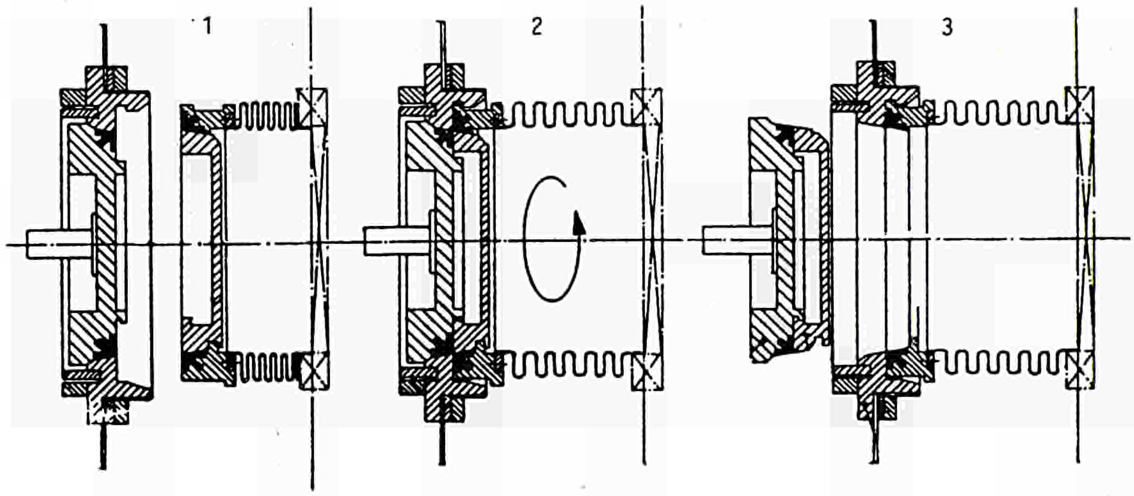


Figure 2. Sliding hatch panel





1, 2, 3 connection locking, double door deconnection

3, 2, 1 double door connection unlocking, deconnection

Figure 3. Wall equipment sealed transfer device.

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. 3.15.).

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

The work program which had been foreseen for 1987 was carried out according to the initial planning except for the B5 phase which will only start in 1988. The high pressure generator, the motorized cutting system as well as the cutting tanks are already operational. (They were even introduced at a congress which was held in 1987). The B2 phase permitted us to take the installation in charge, complete its measurement equipment and evaluate the performances and some parameters. Phases B3 and B4 have already been carried out and the material needed for the adaptation, teleoperation and telemaintenance is available at the Fontenay plant. We still have to complete the performance measurements, including the waste and aerosol measurements (phase B6) and carry out the cutting and teleoperation maintenance tests (phase B5).

Progress and Results

1. Procedure definition, study and realization of test installation (B.1.)

Figure 1 shows the diagram of the installation used, without the measurement cell, but protected by a vinyl tent.

Figure 2 shows the cutting system as well as the cutting head elements.

2. Equipment tests and training of operators (B.2.)

The personnel training was made possible due to the data collected when the program was put into operation, to the study of the documents supplied by the constructor as well as the experience we acquired.

The series of tests carried out enabled us to draw our first few conclusions on the use of the cutting system and the performances :

- sensitive spot : installation of the saphire-holder (very small parts) into the cutting head and adjustment of the acceleration tube position ;
- problem of solid waste retrieval : this operation, which in principle is an ordinary one, becomes quite difficult due to the high density of the grains which have a tendancy to rapidly sink to the bottom of the tanks and plug the filters ;
- the speed performances were inferior to the ones given by the constructor in this first series of tests (see table I) ;
- as far as the machine behavior and the safety are concerned, there is no peculiar difficulty apart from the periodical tightening of joints on the high-pressure pipes.

3. Design and manufacturing of components used in the teleoperation adaptation (B.3.)

In order to ensure the flexibility needed by high-pressure pipes, which must be connected to the telemanipulated cutting head, the original pipes, whose stiffness is important when being bent, are replaced by more flexible pipes and spirally pre-shaped. Moreover, the joint is modified so that it can be tightened or untightened with a manipulator. The tests to be performed on these pipes will be carried out in 1988.

4. <u>Study and realization of components used in teleoperation maintenance</u> (B.4.)

In order to ensure the cutting head maintenance during telemanipulation, the following solutions were chosen :

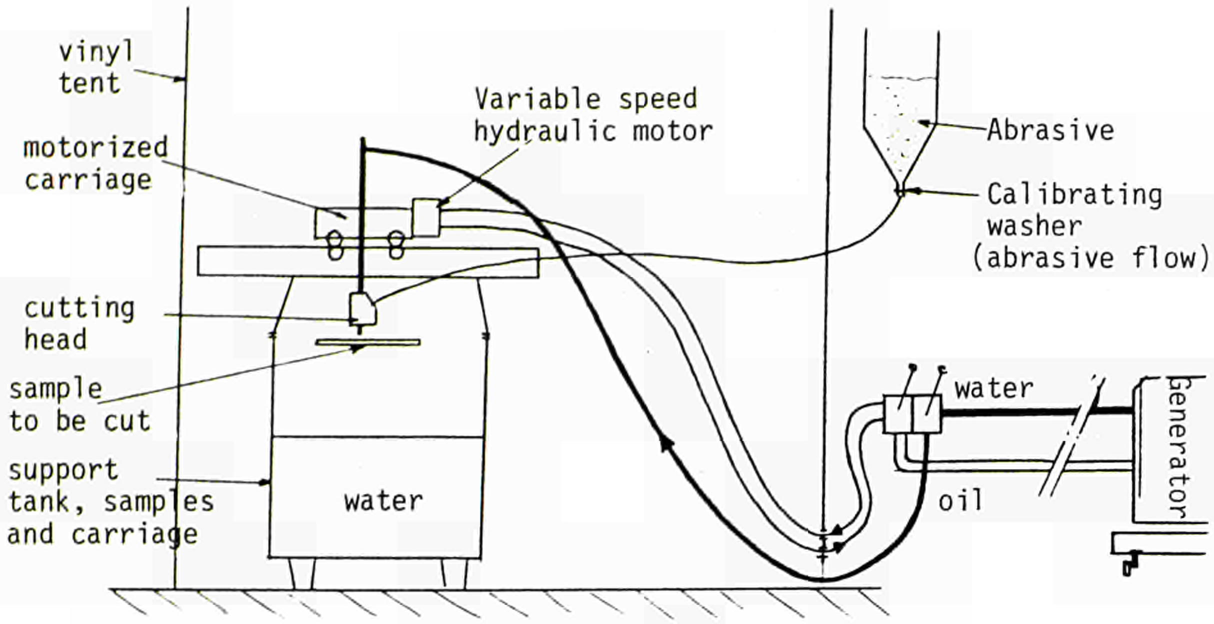
- installation of a saphire nozzle into the cutting head ; outside the cell, after decontamination in a ventilated glove box if necessary ;
- connection, disconnection of the high-pressure pipes and the abrasives pipes on the cutting head : with a manipulator and within the cell, due to a specially-fit stationnary bench.

This bench is made up of a cutting head fixation craddle and a guiding device for the high-pressure pipes and their joints so as to tighten or untighten them with a telemanipulated dynamometrical wrench. This set-up is operational after telemanipulation by inactive tests and will be put into operation in 1988.

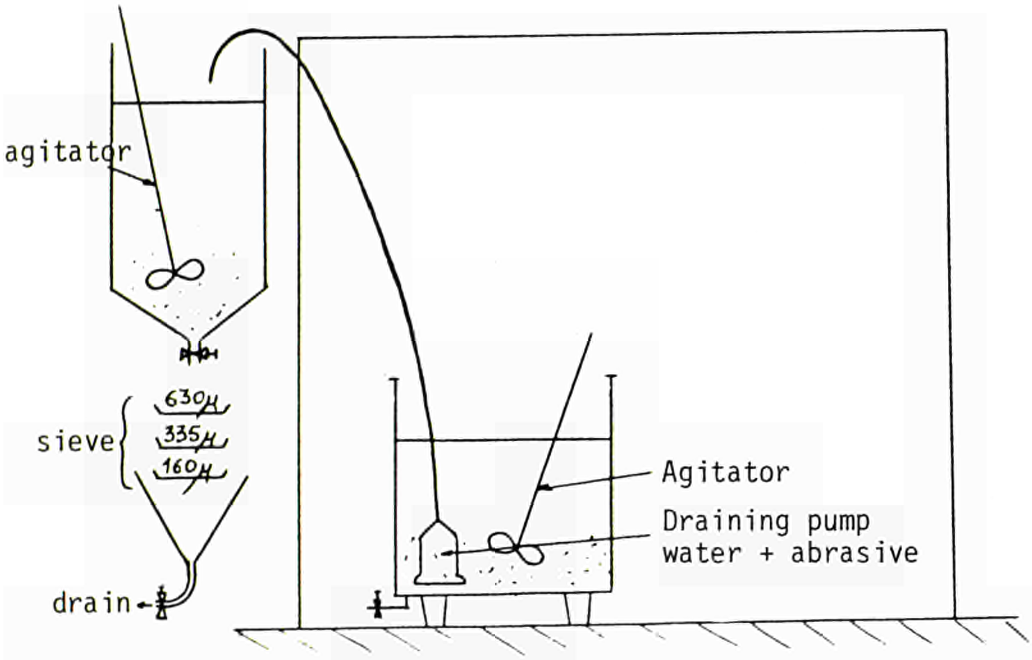
TABLE I : PHASE B2 PARAMETERS AND TESTS RESULTS

Test Number	Water saphir (mm)	Water flow (l/mm)	Abrasives Dénomination	Abrasives washer diaphragm	Abrasives flow (kg/mn)	Diam. acceleration tube (mm)	Distance tube/piece (mm)	Cutting speed (mm/mn)
	Steel A 42 Samples (10 mm)							
12	0,36	3,4	Grenal 60	16	0,51	1,5	01	94
15	"	"	"	12	0,26	"	01	76
13	"	"	"	14	0,35	"	10	86
14	"	"	"	14	0,35	"	20	69
17	"	"	Garnet 36	16	0,41	"	01	63
16	"	"	"	14	0,32	"	01	33
02	0,46	4,5	Grenal 60	24	1,56	2,3	01	112
01	"	"	"	18	0,68	"	01	125
08	"	"	"	"	"	"	01	114
06	"	"	"	"	"	"	10	120
07	"	"	"	"	"	"	20	97
05	"	"	"	16	0,51	"	01	120
03	"	"	Garnet 36	28	2,04	"	01	130
	Stainless steel samples Z2CN 18-10							
18	0,36	3,4	Grenal 60	14	0,35	1,5	10	65
19	"	"	"	"	"	"	20	60
09	0,46	4,5	"	18	0,68	2,3	10	124
10	"	"	"	"	"	"	20	102

Nota : the water pressure was constant and neared 2300 bars for all these tests.

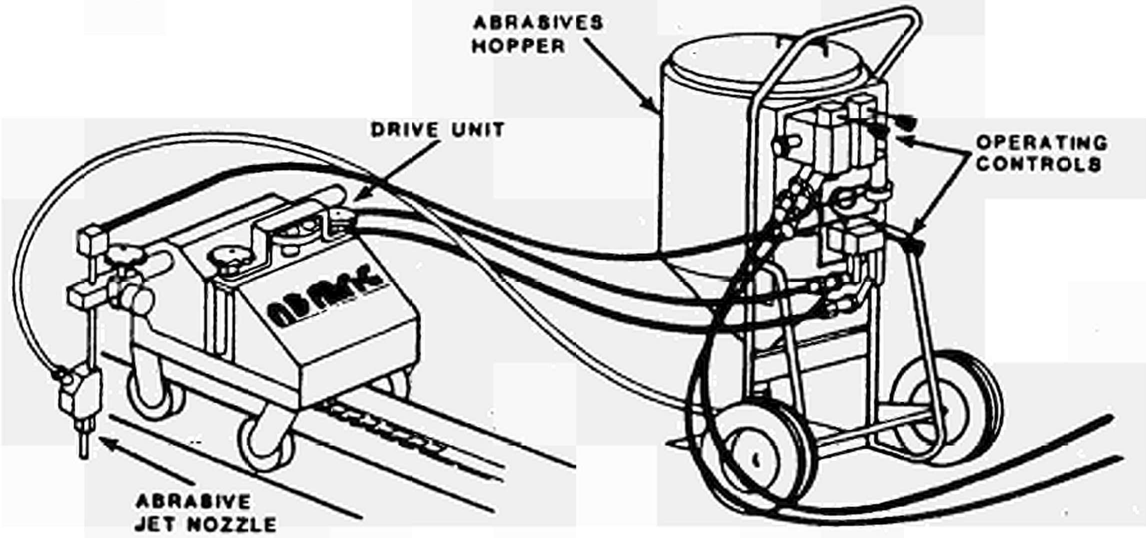


Phase 1 : CUTTING

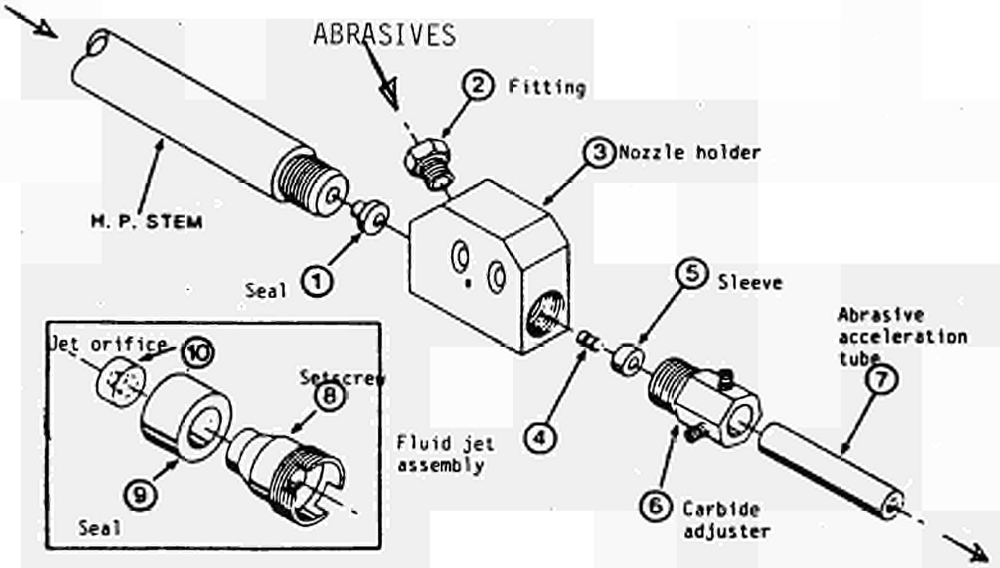


Phase 2 : DRAINING AND RETRIEVAL OF LIQUID AND SOLID WASTES

Figure 1 : Installation Diagram



SYSTEM SET UP FOR STRAIGHT-LINE CUTTING



ABRASIVE JET NOZZLE ASSEMBLY

Figure 2 : Cutting system and cutting head

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. 3.14.).

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 application of abrasive water jets gives an alternative for submerged cutting of nuclear components to other thermal or mechanical methods. The abrasive water jet is formed by a high speed water jet added with solid particles, which are accelerated by the water and focussed in a specific nozzle.

Investigations on the cutting effiency were carried out under water. A special feed system was developed to realize small abrasives flow rates with a sufficient proportioning accuracy. The test has shown the influence of the abrasive flow rate on depth of cut. In addition an abrasive cutting head was designed for underwater application. Tests were carried out with different acceleration nozzles to find an optimum design for efficient submerged cutting.

Progress and Results

1. Preparation of the test facility for underwater cutting (B.1.)

The cutting tests will be done in a water basin with a volume of $1 m^3$ and a working area of 1m x 1m. The cutting equipment consists of a 45 kW pressure pump (pressure up to 4000 bar, water flow rate up to 2.5 l/min) with an adapted abrasive cutting head. A general view of the test facility is given in figure 1.

To control the test conditions some different measuring instruments are installed. The pressure of the pump is registered by a pressure gauge, the water flow rate is given by different characteristic lines depending on the nozzle diameter and the water pressure. The feeding mechanism was changed to realize a lower feed rate. This velocity is measured by a pulse counter. Traverse rates from 2 up to 8000 mm/min are within reach.

An important part of the investigations was the development of the abrasive feed system. The abrasives are stored dry in a storage tank. The abrasive cutting head works according to an injection pump and produces a suction pressure in the mixing chamber and the abrasive supply system. For fine setting of the abrasive flow rate this suction pressure is adjustable by pressurized air and a restrictor. The coarse setting of the flow rate is realized by exchangeable orifice plates.

The setup is given in figure 2. Characteristic lines give the abrasive flow rate depending on suction pressure and the diameter of the orifice plate.

To examine the influence of the water depth on jet formation (B.3.) a pressure chamber was constructed. The installation of the equipment to control the test conditions in the chamber and the safety of operation is still going on.

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

The application of abrasive water jets in cutting radioactive material causes two main preconditions:

- It must be possible to handle the cutting system under water because of the protection of the water layer.
- Because of subsequent waste treatment it is necessary to minimize the abrasive and water flow rate.

To realize the last demand an abrasive cutting head was designed and constructed (figure 3). The addition of abrasives is done by two radially drilled holes near the water jet nozzle. Due to a ball joint it is possible to position the focussing nozzle on the water jet. Adjusting screws fix the ball joint after adjusting.

It is possible to use different focussing nozzles. They are adapted to the cutting head by a small cone. Some different materials such as boron carbide and tungsten carbide were tested.

Cutting tests in air have shown an optimum of abrasive flow rate to reach a high cutting depth. Increasing this flow rate has caused a decrease in depth of cut. First experiences under water have shown, that this optimum in case of submerged cutting is at lower abrasive flow rates than in air. Further investigations have to confirm this results.

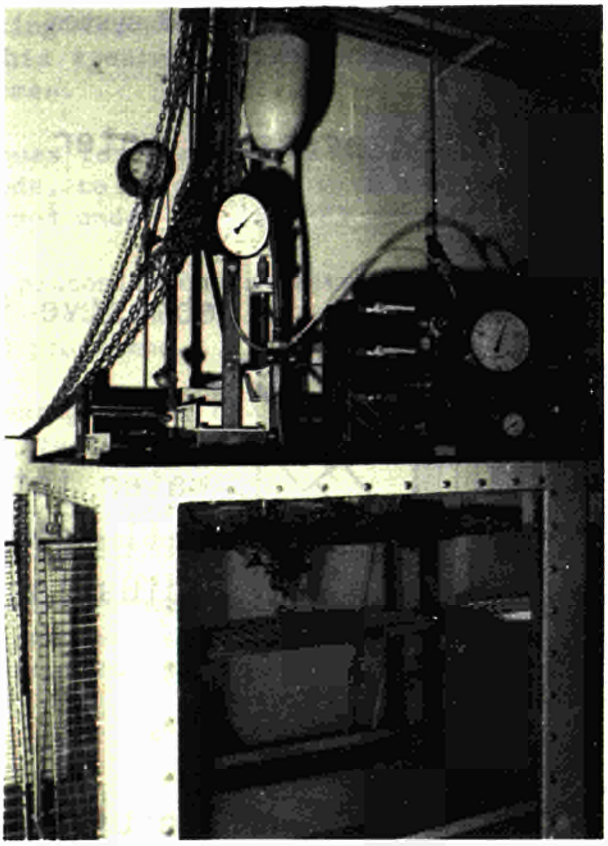


Figure 1: General view of the water basin

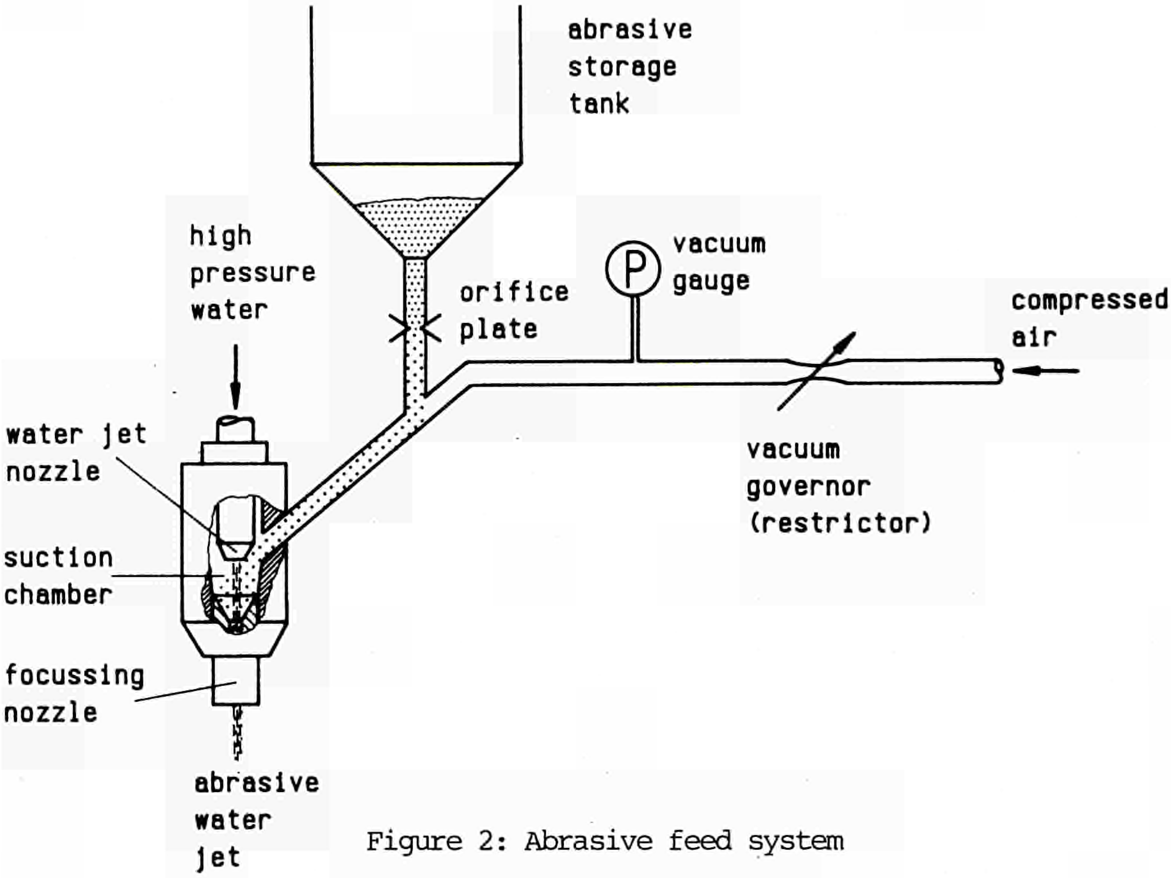


Figure 2: Abrasive feed system

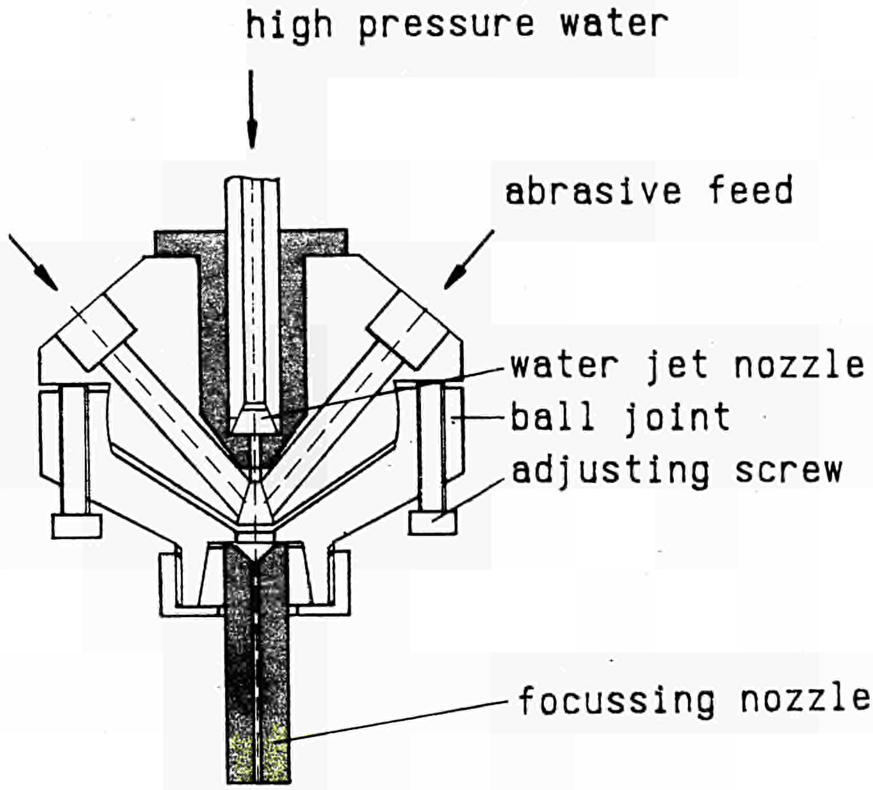


Figure 3: Abrasive cutting head

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

Nine research contracts relating to Project N°4 were being executed in 1987, including two new contracts concluded in 1987, as well as one contract whose execution has been completed in 1987.

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 1988
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.

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.

C. Progress of Work and Obtained Results

Summary

The behaviour of Europium during arc furnace steelmaking has been examined. It was clearly shown that Europium is strongly absorbed by the slag, and the amounts in the steel and the off gases were too low to measure.
Some further work has been started on the absorption of caesium by steel-making slags in the induction furnace. First results from the work suggest that absorption of caesium is strongly dependent on temperature.
A theoretical study has been made of the likely behaviour in steelmaking of less common elements which may be encountered in contaminated steel. It is expected that antimony, nickel and niobium will be retained by the steel. Strontium and americium will be absorbed by the slag.
Studies of the retention of cobalt in slag show that when it is present it is in metallic globules which are physically trapped by the slag. Most of these globules can be removed by a magnet.
Some consideration has been given to the costs and benefits of the recycling of slightly radioactive steel. It appears to be acceptable in terms of radiation exposure. The cost of recycling exceeds the value of the steel, but it will be lower than that of the alternative routes of disposal.

Progress and Results

1. (Melting in a 5 tonne Arc Furnace (B.1)

A melt of steel (3.5 tonne) was prepared in the experimental arc furnace and steel plates (8.7 kg) contaminated with Europium-154 were added to it. The steel was then blown with oxygen, and tapped and cast in the normal way. It was shown the Europium-154 was rapidly absorbed by the slag, and that the levels present in the steel and the fume were too low to detect. The results confirm those obtained previously from work in the induction furnace. It is concluded that under all normal steelmaking conditions Europium-154 will be absorbed by the slag.

2. Induction Furnace (B.2)

Earlier work had suggested that caesium could be retained by acidic steelmaking slags, but not by basic slags. Some work has begun to define more fully the behaviour of caesium.
First results from this work suggest that the initial retention of caesium by slag is strongly temperature-dependent in the range 1400-1600°C. The caesium which is initially retained by the slag is then released over a period of about 10 mins. The results suggest that the rate of release does not appear to be dependent on slag composition. This is contrary to expectations and will be investigated further.

3. Slag/Metal Chemistry of Specific Radionuclides (B.5)

A theoretical study has been made of the thermodynamics and steelmaking behaviour of some less common radionuclides which may be encountered in steel from decommissioning. It is expected that antimony, nickel and niobium will be retained by the steel. Strontium and americium will be absorbed by the slag.
The retention of cobalt in steelmaking slags has been studied further. The results show that cobalt may sometimes be present in metallic particles which are physically trapped in the slag. These particles can mostly be removed by magnetic separation.

4. Work Place Acceptance and Benefit and Detriment (B.7)

The possibility of melting radioactive steel at a local steelworks is being discussed but no conclusion has yet been reached. Radioactive steel would be diluted with non-radioactive steel during the melting process. The radioactivity of the products would be so low that they would be free for release without restriction.

Improved assessment of the benefit and detriment of recycling radioactive steel has been made difficult by the absence of data from a large scale trial. In addition, the assessment can be made properly only if the benefits and detriments of alternative routes are known. These alternative routes are not yet decided for the UK. Present estimates are that recycling 4,000 tonnes of slightly radioactive steel would be acceptable by current IAEA criteria of radiation protection (IAEA Tec. Doc. 401, published 1987). There would be benefits in the values of the steel recycled but these could not equal the cost of the recycling operation which is labour intensive. The recycling of steel would be financially attractive, however, if the cost of the alternative route of disposal is high, as it is unlikely to be in the UK.

4.2. Melting of Radioactive Metal Scrap from Nuclear Installations

Contractor: Siempelkamp Giesserei, Krefeld, Germany
Contract N°: FI1D-0016
Working Period: November 1984 - December 1988
Project Leader: L. Küppers

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 surface contamination in the environment of the melting furnace was ascertained at the control board near the furnace and at the casting place in front of the furnace. As further examinations the hall air was investigated with respect to radioactive aerosols and the dose rates at the filter plant were measured.
The diffusion path of radionuclides in the brick lining of the furnace was investigated (brick lining probes) and a melting campaign with inset of zinc galvanized steel scrap carried out..

Progress and Results

1. Radiological long-term behaviour of the melting plant(8.10.)

During the preceding research program work carried out until 31.12.1986 had the main aim in the development of a suitable charging installation with filter systems and furthermore the distribution of radionuclides in melt and residual materials had been of interest.
Since 01.01.1987 the radiological long-term behaviour of the melting plant has been investigated.
Radiological measurements on components of the melting facility and in the environment of the furnace will be carried out over 2 years. Since April 1987, 4 melting campaigns with in total 243,8 Mg melted steel scrap coming from nuclear power plants had been carried out. The melted masses of these melting campaigns are given in table I. Measurements of the surface contamination in the environment of the melting plant have been ascertained on the control board near the furnace and on the casting place in front of the furnace. The diffusion path of radionuclides into the brick lining of the furnace was examined by taking samples from the furnace lining. Furthermore aerosol-samples were collected during the melting process in various distances to the furnace.

1.1 Surface contamination on the control board near the furnace

As an estimation of the environmental exposure caused by melting of radioactive scrap the deposit of dusts was gathered on paper filters lying on the roof of the control board near the furnace. The activity of these dusts was ascertained by Gamma-measurements after the melting campaigns. The results of these measurements are listed in table II and the values with their maximum of 0,6 Bq/cm2 show only an unimportant activity level.
The limit of the admissible surface contamination within an operation surveillance area (Überwachungsbereich) in Germany is 3,7 Bq/cm2. The measured contamination values are acceptable because the area near the furnace is cleaned systematically.

1.2 Floor contamination in the casting area of the furnace

In different places at the casting areas in front of the furnace, samples of slags and dusts were swept up from regions of 1 m2 and the activity was determined. Table III shows the ascertained values of total activity, specific activity and surface contamination. In these samples a remarkable activity of Co-60 is recognizable. The contamination is explicable because the exhaust-system is not in operation at the moment of casting and during slag work. However, this part of the hall floor is decontaminated before removing the surveillance area, so that no danger is to be expected.

1.3 Measurements of radioactive aerosols in the hall air

During the melting campaigns aerosol samples were collected on paper filters by means of an aerosol-sampler and subsequently measured with a Gamma-detector. The results of these measurements are presented in table IV. In the clean gas tube behind the filter plant no activities could be found in the aerosol samples. In the environmental air of the hall an aerosol activity up to 3,0 Bq/m3 Co-60 was measured and also in the crane driver's cabin above the furnace 4,22 Bq/m3 was detected.

In Germany the limit for the derived air concentrations amounts to 14 Bq/m3 in an operation surveillance area. For the calculation of this limit an annual inhalation volume of 2500 m3 is assumed.

1.4 Control of dose rates at the filter plant

Beginning with the melting campaign 25.08.1987 the dose rate at the surface of the filter components was controlled after every melting period to estimate a possible sum up of activity. This is continuing and the results of the measurements will be presented in 1988.

1.5 Activity in the brick lining of the furnace

After the melting campaign 13.04./14.04.1987 samples were taken from the brick lining of the furnace to investigate the diffusion path of radionuclides. The radio-activity was determined at 11 points of the lining in depth of 5 mm, 35 mm and 70 mm. The results of these measurements showed a diffusion of radionuclides only up to 5 mm depth. The average specific activity of the samples amounts to 3,4 Bq/g.

2. Experiences with melting of zinc-galvanized scrap (B.7.)

During a melting campaign in total 9,11 Mg of zinc-galvanized steel sheets pressed in pellets were melted. At an average sheet thickness of 0,9 mm a total coating of the sheets of 1400 m^2 results for this melted mass. The thickness of the zinc layer is 0,018 mm for commercial sheets, so that a total zinc mass of 360 kg results from the inset material. After the melting campaign, 391 kg oxides of zinc were found in the dust drums of the filter system, this corresponds stoichiometrically to a zinc mass of

314 kg, so that an unknown mass of dust remains in the filter components. The distribution of radionuclides in melt, slag and filter dusts is given in table V.

Table I: Masses of melted scrap

melting campaign	masses of radioactive contaminated steel scrap
13./14.04.87	73,2 Mg
25./26.05.87	86,2 Mg
24.08.87	55,1 Mg
23.11.87	29,3 Mg
SUM	243,8 Mg

Table II: Surface contamination on the control board near the melting furnace

melting campaign	Surface contamination in Bq/cm2		
	Co-60	Cs-137	Sum
13./14.04.87	0,2	-	0,2
25./26.05.87	0,6	-	0,6
24.08.87	0,1	0,18	0,28

Table III: Floor contamination in the casting area of the furnace

Sample no.	distance to the furnace		activity of masses from 1 square meter in KBq Co-60	Cs-137	Cs-134	Sum
1	2 m		11,780	0,696	-	12,48
2	3 m		9,72	0,632	-	10,35
3	4 m		10,72	0,885	0,080	11,69
Sample no.	distance to the furnace	masses (g)	specific activity (Bq/g) Co-60	Cs-137	Cs-134	Sum
1	2 m	173	68,1	4,0	-	72,1
2	3 m	162	60,0	3,9	-	63,9
3	4 m	140	76,6	6,3	0,6	83,5
Sample no.	distance to the furnace		surface contamination (Bq/cm2) Co-60	Cs-137	Cs-134	Sum
1	2 m		1,16	0,07	-	1,25
2	3 m		0,97	0,06	-	1,03
3	4 m		1,07	0,09	0,01	1,17

Table IV: Measurements of aerosol samples in the area of the melting furnace

filter no.	activity Bq/m3	place, date and time of sampling
01	less than detection limit	clean gas pipe behind the filters 25.05.87 50 min
02	less than detection limit	clean gas pipe behind the filters 26.05.87 11 min
03	Co-60 = 0,8	air in the melting shop at 12.oo during melting 25.05.87 50 min
04	Co-60 = 3,0	air in the melting shop at 18.oo during melting 25.05.87 60 min
05	Cs-137 = 2,88 Cs-134 = 1,34 ---- 4,22	air in the crane driver's cabin 17.08.87
06	Cs-137 = 0,32	air in the control board of the furnace
07	Cs-137 = 0,75	air in the melting shop at 14.oo during melting 23.11.87 105 min

Table V: Distribution of radionuclides at the melting of zinc-galvanized steel scrap

	nuclide	spec. activity in Bq/g	activity in K Bq	activity distribution in %
melt	Co-60	0,96	14200	83,0
cyclone filter dust	Cs-137	2,29	322,9	1,9
tube filter dust	Cs-137 Co-60	6,84 0,51	1710 127,5	10,0 0,8
slag	Cs-137 Co-60	2,50 0,59	600 141,6	3,5 0,8

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 under publication.

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

The work performed in 1987 consisted mainly in:
- market study and buying of spraying apparatus for painting with polymers in a nuclear environment,
- two test applications under real conditions, i.e. contaminated and radioactive components;
- follow-up of characterisation tests for polyurethane.

Significant progress has been made and enabled the previous programme to be respected, especially with regard to application in a strongly alpha-contaminated area at the La Hague Center, and a strongly gamma-radioactive piece at the Grenoble Center. The characterisation of eleven formulations of polyurethane enabled their classification according to their main properties (rheology, diffusion by water, resistance to temperature) and thereby the selection of the three most appropriate formulations proposed for industrial applications.

Progress and Results

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

Polymeric formulation

Eleven formulations have been tested, composed by chemical components as follows:
- polyol associated to a polyether skeleton or an insaturated oil,
- an isocyanate (MDI) or an isocyanate associated to a prepolymer (polyether),
- mineral additives if necessary.

Mechanical and physical characteristics

Systematic tests were carried out on the eleven formulations, giving the following results:
- rheology,
  glass transition temperature varies from -50°C to + 40°C;
  shore D hardness varies from 25 to 65.
- water diffusion,
  diffusion coefficient varies from 10^{-8} to 10^{-10} $cm^2.s^{-1}$.

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

Physico-chemical tests and projection tests with the spraying apparatus enabled the selection of the most appropriate qualities produced by the following companies: BAULE (ref. nr. PU1), SOUPLETHAE (ref. nr. PU3), DEBRATHANE (ref. nr.PU8).

3. Study of a mobile projection apparatus (B.3.)

The study of a projection cabin has been continued. In 1987, a filtration module has been purchased and tested. Filters of this module retain paint particles to avoid entering the high efficiency filter step.

4. Feasibility study (B.4.)

Two pilot scale applications in a nuclear environment have proved the feasibility of painting contaminated and irradiated pieces, i.e.:
- at the La Hague Center, large highly alpha-contaminated lead shieldings (total surface $\pm$ 40 m^2, total weight $\pm$ 27 t.) have been treated with this painting process; complete decontamination has been obtained without creation of contaminated aerosols.
- at the CEN Grenoble, teleoperated painting of a highly gamma-radioactive piece originating from dismantling the G3 reactor at Marcoule (0.2 GY m^{-1}, i.e. 20 rad. m^{-1}) has been carried out successfully in a special projection cabin. In about half an hour, $\pm$ 4.5 mm thick coating was achieved in several layers.

4.5. Treatment of Active Concrete Dust by Slurry Setting Method

Contractor: Taylor Woodrow Construction Ltd.
Contract N°: FI1D-0042
Working Period: April 1987 - September 1988
Project Leader: I. Ll. Davies

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

Following completion of UK funding arrangements (CEGB, UKAEA...), work on the project commenced actively in September 1987.

After detailed discussion with associated UK organisations (Central Electricity Generating Board, Dept. of the Environment, United Kingdom Atomic Energy Authority, Health and Safety Executive and the Nuclear Installations Inspectorate), a survey of nuclear decommissioning techniques and also of sodium silicate technology have been carried out. Six principle tasks were identified (a to f) within the project. A detailed definition of each task has been developed, together with a programme without changing the overall scope of the initial work programme.

Progress and Results

1. Identification of Tasks

The following tasks have been identified:

a) Updating the literature survey to cover recent advancements in sodium silicate/concrete technology;

b) Characterisation of concrete demolition debris - based on demolition process, chemical composition of parent concrete and variation of chemical composition with particle size;

c) Laboratory-scale mix development - to establish optimum silicate composition, and debris/silicate ratio for setting of slurries;

d) Commercial feasibility of the slurry setting method and development of a Method Statement for its use in practical decommissioning;

e) Measurement of the engineering properties of set silicate/concrete slurries, including bleed (sedimentation), setting time, strength, heat of setting and dimensional stability;

f) Other aspects, including scale trials of the slurry setting concept and extension of silicate technology to non-nuclear applications.

Progress has been made in the first two tasks, i.e. a literature survey of sodium silicate/concrete interaction and characterisation of concrete demolition debris.

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

The entire melting facility (figure 1) is completely assembled with the exception of the items of the air supply system and items of the water cooling system. The permits required are expected to arrive from KfK in April 1988. The remaining work, including acceptance inspection will need about 4 more months. It is assumed that the melting facility will start operation during summer 1988.

The auxiliary equipment required for the melting operation is completely assembled.

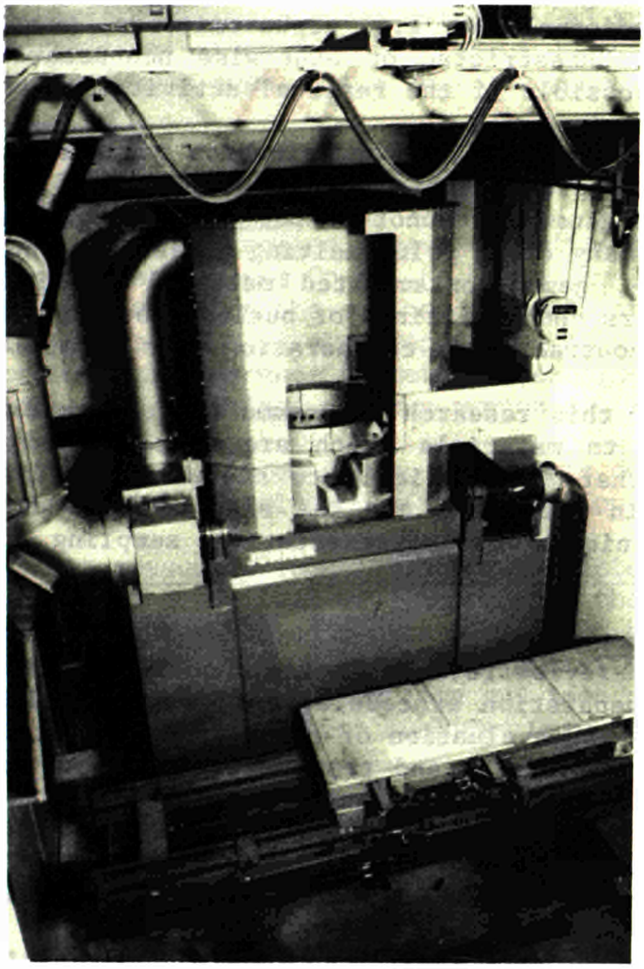


Figure 1: Melting facility

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 - December 1988
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 Obtained Results

Summary

Scrap from Nuclear Power Plants is contaminated with β, γ-emitters, mainly with corrosion product radionuclides. So far only radionuclides which could be detected by γ-spectrometry have been measured in the melt and slag. The investigation is now extended to radionuclides such as Fe 55, Ni 63 and Sr 90 which are hard to measure. Fe 55 was selected for the first laboratory-scale melt experiment. This radionuclide is homogeneously distributed in the metal melt just like Co 60. Thus Co 60 can be used as an isotopic indicator for this ε-emitter.

Scrap originating from Fuel Enrichment and Fuel Fabrication Plants is only contaminated with uranium elements which are α-emitters. The behavior of such elements was investigated using Am 241 traced scrap for melt experiments. A decontamination factor of about 100 was measured for this element in the metal melt even at the very low added quantity of 4 x 10^{-7}g Am 241.

Progress and Results

1. Preparation of traced scrap (B 2)

In order to avoid any uncontrolled losses of radio-tracers during the melt process, the three isotopes Co 60, Fe 55 and Am 241 used in the melt experiments were deposited on the surface of small steel discs by an electro-deposition method. The electro-deposition of cobalt, iron and americium was performed with reference to methods described in the literature (1-5) which were optimized for our purposes.

To accommodate the tracer coated discs, a hole was drilled into a section of the scrap material and plugged after insertion of the discs.

2. Development of direct measuring techniques for α-emitters (B3)

A low level α-measuring device was used to measure the α-activity in metal and slag specimens. The selected sample size of 26.4 cm^2 counting area yielded a detection limit of 0.05 Bq/g. The calibration of the measuring device was performed with traced iron and iron oxide samples simulating ingot and slag samples. Tracer nuclides were Am 241 and uranium isotopes.

5 mm thick discs were cut from the test ingot. These discs were mounted on a support area and ground and polished with a surface grinding maschine.

α-emitters can be measured only in a very thin surface area (approximately 8 to 14 µm, according to α-energy). Therefore multiple measurements are necessary to ensure that the activity distribution in the bulk is homogenous. This was obtained by repeated grinding and subsequent measurement of the α-activity.

3. Development of sampling and separating techniques for ε-emitter decaying by electron capture (B5)

To determine Fe 55-activity, samples were taken from the ingot, dissolved, and then the iron was electro-deposited on small steel discs. Fe 55 was measured by X-ray spectrometry with a Si(Li)-detector. The weak X-ray radiation of 5.89 keV shows a rather high self-absorption in the deposited iron layer. Therefore a calibration curve for the absorption of the X-ray radiation of Fe 55 was measured. The best results were obtained with a deposited quantity of about 10 mg Fe which had a self-absorption of about 35% (Fig. 1). The detection limit for Fe 55 is rather poor at 48 Bq/g. Thus it was very necessary to demonstrate that Co 60 can be used as an isotopic indicator for these hard-to-measure isotopes which are homogeneously distributed in the metal melt.

4. Results of the laboratory-scale melt experiment with Fe 55, Co 60 and Am 241 traced scrap (B2)

The added activity quantities and some other parameters are summarized in Table I. After cooling down, crucible, ingot and slag were separated. The ingot was divided into measuring samples and the Am 241 activity measured in the low level measuring device (see heading 2).

The slag was ground in a ball mill to a grain size of about 1µm and than transfered to a petri dish of appropriate diameter and the α-activity directly measured.

The activity collected on the off-gas filter was leached evaporated to dryness and the β, γ, ε and α-activities measured by appropriate methods.

The results of all measurements are summarized in Table II. It can be deduced from the measurement results that Am 241 is nearly totally released to the slag. The decontamination factor for the ingot is about 100. This was reached even with the very low added americium quantity of 4×10^{-7}g. In contrast to americium, Co 60 and Fe 55 are held back in the ingot as was expected. Both are homogeneously distributed, thus Co 60 can be used as an isotopic indicator for this ε-emitter which is hard to measure.

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Table I

Material, Additives and Boundary Conditions of Laboratory Melt Experiment

Material	Additives	Added activities	Boundary conditions of melt
steel scrap 3120g	coal: 90g (3%) SiC: 75g (2.5%) slag former: 30g (1%)	Co 60: 2.5×10^4 Bq Fe 55: 3.3×10^5 Bq Am 241: 5.0×10^4 Bq	temperature: 1480 - 1530°C dwell time of melt: 15 min. cooling down time: 6 h cover gas: argon with 10% hydrogen

Table II

Results of a test melt with Am 241, Co 60 and Fe 55

	Co 60		Fe 55		Am 241	
	activity concentration Bq/g	total activity Bq	activity concentration Bq/g	total activity Bq	activity concentration Bq/g	total activity Bq
Added Activity	8.0	2.5×10^4	105.7	3.3×10^5	16.0	5.0×10^4
Measured activity in the ingot	8.7	2.7×10^4	73 ± 25	2.3×10^5	0.16	490
Measured activity in the slag	1.1	42	-	n.d.	1.4×10^3	5.5×10^4
Measured activity in the off-gas filter		58		n.d.		27

n.d. = not detected

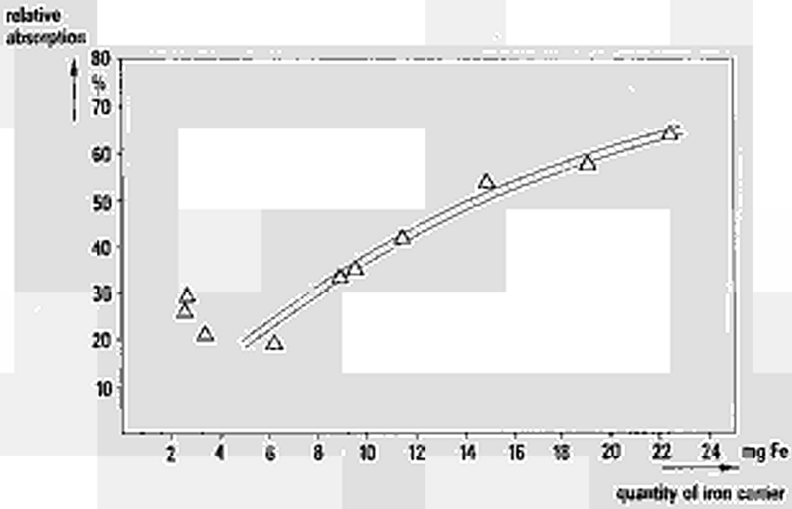


Fig. 1 Fe 55 Acti[redacted] Measu[redacted]nt by X-Ray Spectrometry Using Si(Li)-Det[redacted]r. Cal[redacted]ion Curve for Absorption of 5.89 keV [redacted] Radia[redacted]

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

In parallel to the questions for surface storing conditions which were discussed with competent organizations, studies were carried out on highly-sensitive analysis methods for future lixiviation research, on the core-sampling equipment to extract large samples from G2 reactor core, and on impregnation product formulas which were tested on inactive graphite.

Progress and Results

1. Storing conditions

Neutron flux calculations, complementary to the ones which were already known, were conducted as well as the activations which resulted from them.

In parallel to this, measurements which were usually associated to combustions, were made on small samples taken from inside the G2 reactor channels.

According to the usual rules peculiarly to the burial site, tritium contamination necessitates coating (value exceeding 7.4 MB/kg) whereas the margin is quite large for gamma emitters.

For beta emitters, generally those doses which are assumed likely to be received by the public after 300 years, period after which the shallow land burial site can no longer be submitted to restrictive utilization conditions were calculated. The maximum individual dose resulting from the use of water (including for drinking purposes) and the use of water from a well which would be bored nearby, does not exceed 1.6 10^{-3} Sv/year after 1800 years and would be due to the C^{14} taking into account the following lixiviation rates :

^{3}H	$5.10^{-3}\ y^{-1}$	^{63}Ni	$2.10^{-5}\ y^{-1}$
^{14}C	$10^{-4}\ y^{-1}$	^{36}Cl	$10^{-2}\ y^{-1}$

2. Sample constitution

To meet the standards established for waste pre-characterization, lixiviation tests must be carried out on ortho-cylindrical standard samples measuring 80 x80 mm. Such a core-sampling operation was undertaken in the G2 core.

The core-sampler is horizontally centered between the fuel channels and goes through the entire reflector and moderator until it reaches the core centre. The prism-shaped graphite bricks will then be drilled transversally.

The core-sampler will be introduced through a "glove-finger" measuring 5.3 m in length and 161 mm in diameter. This necessitated the adaptation of a standard core-sampler notably to guide the tool at such a distance and to cut into the graphite. Two sheet-iron plates, which thickness are 12 and 20 mm must be successively drilled.

Particular care has been taken for :

1) the ventilation : the air must flow from the reactor hall to the lock and from the lock to the reactor core. 2) also the non-return of the strong irradiating chips coming from the pierced sheet-iron plates.

3. Graphite characterization

It first necessitated the design of a first-rate destructive analysis method in order to chemically separate the isotopes. This method will favorably be used to follow the radionuclide loosening during the lixiviation tests. The water absorption has been tested : on radioactive big samples, a weight increase of 15 per cent in 100 hours was commonly observed.

4. Study of an impregnation procedure

The first laboratory tests were carried out with the objective of impregnating the graphite and filling up most of its porosity. Bitumen and tar-epoxy formulas have given good results on inactive samples.

4.9. Separation of Contaminated Cement Stone and Non-contaminated Concrete Aggregates

Contractor: TNO, Delft
Contract N°: FI1D-0068
Working Period: January 1987 - December 1988
Project Leader: R. Wiegers

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 impermeable 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

The first part of the working program (B.1) has been finished. In this program two different types of contaminated concrete were separated into cementstone (hardened cement) and aggregates, by means of milling and washing with acids. By measuring the specific activity of the contaminated concrete (before separation), the cementstone and the aggregates (after separation) it could be seen that the specific activity of the aggregates was very low (± 1 Bq/g) and mainly due to contamination in the cementstone.

Work continued with the development of a separation technique on laboratory scale. The methods of separation are milling and thermal shock. In the milling process the following parameters are important: filling, use of liquids and the diameter of the milling vessel. For three types of concrete (with different strength properties and different mixture compositions) the influence of filling, dimension of the vessel and watercontents were partly measured.

Progress and Results

1. Checking the hypothesis that only the cementstone is contaminated (B.1)

In this part of the program the hypothesis is tested that in contaminated concrete only the cementstone is contaminated and not the aggregates. By separating two samples of contaminated concrete (into cementstone and aggregates) and measuring the specific activity of the samples, this hypothesis was confirmed. These measurements were carried out with a Germanium detector in a Heath Shield configuration.

In tabel I the origin of both samples and the specific activity of both the concrete and aggregates is given. The Dodewaard sample was not entirely separated in the contrary to the Borsele sample. Differences in concrete properties and mixture compositions lead to different results.

2. Development of a separation process (B.2)

After testing the hypothesis the developing of a separation process for concrete in order to separate the concrete into cementstone and aggregates was started. For the test program 9 types of concrete were used: three types of concrete with different compressive strength, each of them with three different granule size distributions. The choice of the different types of concrete is based on the fact that a nuclear power plant is not only built with high strength concrete but also with low strength concrete for the non-constructive parts. Because of the higher porosity of the low strength concrete a higher level of contamination can be expected. In table II all the types of concrete are given. To separate the concrete, three techniques will be checked on their performance:

- milling (with or without balls),
- thermal shock,
- surface active additives.

Until now research was done on the effectiveness of various milling techniques depending on different parameters of which the most important are:

- filling of the vessel,
- use of liquids,
- size of the vessel,
- additives,
- use of balls.

For different types of concrete, having the same nominal size of aggregates (31.5 mm), the size of the vessel (within laboratory limits), and the use of liquids and filling grade of the vessel were optimized. This part of the research program has been carried out by first crushing the concrete and then fill the vessel. In table III the measurements carried out, are given.

The results obtained by using different milling techniques were measured by dry sieving after every separation step. To obtain the granule size distribution the fineness modulus according to Rengers-Antonisse was calculated. This method is often used in concrete technology. The fineness modulus can be calculated by adding the total percentages of material in the sample that is coarser than each of the following sieves. These results can be compared easily. In table IV the fineness modulus of the crushed concrete (starting point) and after the different separation steps, is given. The first fineness modulus has a theoretical value which can only be realised when 100% separation of the cementstone and the aggregates is achieved (provided that the aggregates are not crushed).

In this first stage of development of a separation process it can be concluded that (on laboratory scale) an optimisation of the above mentioned parameters was found. In the following steps thermal shock and the use of additives together with the milling technique for the concrete with the highest strength (B 45) will be integrated. Based on these results the process will be optimized before continuing with the other types of concrete.

Table I: Activity of the samples in Bq/g

	Borsele sample	Dodewaard Sample		
		Co 60	Cs 137	Mn 54
concrete	230	0.34	1.76	2.1
aggregate	1	0.06	0.6	0.22

Table II: Composition of concrete mixtures. Quantities per m^3 of concrete mortar.

	Cement	Water	Fine aggregates	Coarse aggregates	Compressive strength (N/mm^2)
High strength B 45					
Nom 31.5	461	175	464	1256	
Nom 16	500	190	494	1152	52,5
Nom 4	658	250	1331	13	
Medium strength B 22.5					
Nom 31.5	301	175	708	1155	30.0
Nom 16	328	190	738	1062	
Nom 4	431	250	1533	15	
Low strength B 17.5					
Nom 31.5	280	175	734	1148	25.0
Nom 16	288	190	789	1045	
Nom 4	379	250	1579	16	

Table III: Schedule of experiments

1. Crushing by Blake jaw crusher (just for preparation of the sample)
2. Rolling mill (dry or with liquids, with or without balls)
3. Thermal shock
4. Additives

Table IV: Results of the experiments

Sample	Water	Filling	Fm1	Fm2	Fm3	Fm4
B 17.5	10%	35%	6.02	5.63 (6.5)	5.63 (0.0)	4.55
Nom. 31.5		50%	6.60	5.59 (15.3)	5.48 (2.0)	4.55
B 22.5	20%	35%	6.71	5.61 (16.4)	5.54 (1.3)	4.65
Nom. 31.5		50%	6.89	5.39 (21.8)	5.27 (2.2)	4.65
B 45	0%	35%	7.26	6.75 (7.0)	6.52 (3.4)	4.73
Nom31.5		50%	7.11	6.67 (6.2)	6.41 (3.9)	4.73

Fm1 = Fineness after breaking with the Blake jaw crusher: starting point of the test.
Fm2 = Fineness after 20 hours of breaking on the rolling mill.
Fm3 = Fineness after another 20 hours of breaking on the rolling mill.
Fm4 = Original fineness of the aggregates.

Between the brackets the percentage of decrease of the fineness modulus is given.

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 being executed in 1987.

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 - June 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 and was such that the year 1987 was devoted almost exclusively to data collection on three main tasks:

Task 2 The constraints and effects of manufacture on design

Task 3 Transport hazards

Task 4 The influence of disposal parameters on design

Work on Task 2 has concentrated on large reinforced concrete and ferrous containers. Task 3 has been concerned with the study of transport in the Federal Republic of Germany, France and the United Kingdom. Reference routes have been selected for detailed survey which are considered to be representative of the average type of rail journey. This has led to a deduction of accident probabilities for ten transport accident scenarios.

In carrying out Task 4 the following aspects have been examined:

a) disposal site restrictions

b) the identification of key radionuclides

c) the effect of cement formulation on the migration of the key radionuclides

d) safety assessment methodology

e) the performance of ferrous metal containers

f) gas production as a consequence of corrosion, radiolysis and biological action

The results of the data collection phase are to be synthesised in the final two tasks of the study - Task 5 Package Design Criteria and Task 6 Assessment of Package Designs. Work on Task 5 began towards the end of the period under review. Its aim is to collate and assess the package design criteria produced during Tasks 2, 3 and 4 and to set down clear requirements for Task 6.

Papers, memoranda and reports produced on concrete and ferrous packages, transport and disposal have been studied and points arising have been discussed with the authors. To data a few areas of incomplete assessment have been identified and these gaps are being filled as part of Task 5.

Progress and Results (B.1, B.2)

1. Task 2

Work on Task 2 has concentrated on large reinforced concrete containers for material in the Low Specific Activity category and on ferrous metal containers for Type B transport of more radioactive materials. It has been shown that there is a strong case for affirming that reinforced concrete is an ideal cheap material for Low Specific Activity Industrial Packages. In general two types of formulation are required:

a) for the outer casing

b) for the cement grout which infills the voids between the pieces of solid waste

The attenuation of 1 MeV gamma rays by Ordinary Portland Cement (OPC) based concrete is about 15/cm thickness. Greater shielding can be provided by using high density aggregate such as a product of the steel industry termed 'Supershot'. At a density of 4.2 g/cm^3, compared with 2.4 g/cm^3 for OPC concrete, 'Supershot' concrete has a significant cost advantage because it leads to thinner walls and greater payload. Various types of reinforcement such as carbon steel, stainless steel, nylon, Kevlar, other plastics and glass fibre have been considered but it is concluded that carbon steel is the most suitable material for reinforcement at the present time.

2. Task 3 (B.3)

The Transport Hazard survey has investigated those features of container design which are affected by transport. As well as considering the more obvious physical limitations placed by transport on package dimensions and weight, the restrictions of present transport regulations must be taken into account. In addition, the study has addressed the waste inventory and the accidents which could occur to a container on its journey from a power station site to the repository. In consultation with the national railway authorities representative reference routes have been chosen. For France the reference route was between Marcoule and Soulaines; in the Federal Republic of Germany it was between Biblis and Konrad; in the United Kingdom it was between Sellafield and Elstow (but this should not be taken as a commitment by UK NIREX Ltd). Each of these routes has a typical cross-section of hazards associated, inter alia, with passing through industrial areas and hilly country. The routes have been subjected to desk and field studies with classification under headings such as underbridges, embankments/high walls, overbridges, tunnels and fire hazards. An example of the results of the transport hazard surveys leads to the prediction that, for decommissioning waste being moved from Biblis A PWR to Konrad, the most likely accident scenario is for a container train to have a sidelong impact with a second train and that this has a predicted rate of occurrence of 2.6 x 10^{-4}/y.

3. Task 4 (B.4)

The objective of Task 4 has been to determine the influence of various disposal parameters listed in the Summary, on the design of waste packages.

Since most radionuclides which are relevant to reactor decommissioning are short-lived, the number of radionuclides which is important in disposal is very small. For example, the major contributor to radiation dose on shutdown, Co-60 has a relatively short half-life of 4.3 years and therefore will be 99% decayed within seven half-lives, ie 37 years. Of the anticipated arisings from decommissioning wastes, only Ni-63, Ag-108, Ar-39 and additionally C-14 (from gas cooled reactors) are significant after three hundred years decay and these isotopes have a low toxicity compared with alpha emitting wastes from fuel reprocessing.

Due to the low toxicity expected from decommissioning wastes, safety assessment methodology is relatively unimportant. The concept of assessing the release into the 'near field' (ie the man-made parts of the repository) before considering the 'far field' imigration though the geology of the repository appears to be widely accepted. Provision for massive dilution/sorption and decay in the far field results in the insignificant impact of decommissioning waste disposal on the environment. Because the man-made parts of a repository are likely to contain large quantities of cementitious material thereby controlling the pH at about 12 for long periods of time, it is likely that this chemical regime will dominate. Thus in cemented waste forms it is unlikely that changes in formulation, as between Ordinary Portland Cement, Blast Furnace Slag and Pulverised Fuel Ash, will have any significant effect on the ultimate release of activity to the environment. Corrosion of mild steel from either the waste or container cladding may release considerable quanitities of hydrogen gas over the extensive storage period. Radiolysis may also contribute hydrogen, from some of the more active wastes. Diffusion through unclad concrete containers would prevent pressure rise and retention of hydrogen within the waste matrix, but the gas would be released into the repository.

The constraints of disposal on design derive principally from dimensional and weight restrictions rather than on the release of radionuclides.

Thus in the Federal Republic of Germany the major disposal site restrictions at Asse and Konrad are on the container size and weight. For the United Kingdom it is anticipated that there will be a maximum weight restriction of about 60 Te.

4. Task 5 Package Design Criteria

Broadly speaking, the package design criteria are seen to fall into three categories: economic, technical and political.

Up to a point, larger packages are more efficient but above a certain size very large packages impose an unreasonable constraint on handling and transport operations.

The question of disposable or reusable shielding is also largely an economic concern. The total costs of manufacturing, filling, storage, transport and disposal must be considered before decisions regarding detailed repository design can be taken.

There is also a need to consider long term above ground storage. Long term storage and repository charges may well govern the decisions between concrete and steel packages and between disposable and reusable shielding. Approximate whole life package costings have been prepared in an attempt to compare the greater packing efficiency obtained using ferrous containers against the smaller cost of a less efficient concrete package.

Analysis of decommissioning arisings shows that very few components will need to be transported in Type B packages and that Industrial Packages predominate. There is a well developed fleet of Type B packages in Europe so that it would appear that the effort in this project ought to be concentrated on Industrial Packages.

The technical constraints of the entire package life from manufacture through to final disposal have been considered. There are a few absolute constraints such as international transport regulations, package size and weight but the majority are imposed by a desire to achieve a balance between safety and cost. As expected, there do not appear to be any insuperable technical problems.

The question of how the regulatory authorities and the general public view large decommissioning containers cannot be answered at present. Task 5 is addressing the safety issues and will set down suggested package integrity constraints for discussion.

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

The main part of the second work period was devoted to the study and selection of materials to be tested as potential fibre-cement composites for the large container application.

Following the remarks of CCE experts about the EVERITE-type substitution fibres to asbestos which were forecasted to be selected, a wider investigation has been undertaken, in order to enlarge the experimentation field to proper candidate materials.

This has lead to change the basic assumption of the scope of work, which was limited to the manufacture method used for asbestos-cement pipes (Hatschek process). To take into account other types of fibre-cement composites implies to consider also simple molding manufacture technique. Three types of materials are proposed for experimental evaluation :

- With EVERITE (pipe manufacture method) :
3 % cellulose fibre, 1 % polyethylene, 3 % glass fibre, cement and silica fumes
- With LAFARGE and TREFILERIES DE CONFLANDEY (molding process) :
Special concrete matrix with 10 - 15 % wt of steel fibres
- By SGN own means (molding process) :
cement matrix with glass fibres

After the experimental evaluation of these three composites, has been initiated.

Progress and Results

1. Fibre-cement material with the EVERITE process (B.2)

The specificity of the Hatschek process implies to select fibres enabling thorough mixing with an aqueous suspension of water and cement, easy pumping of the mix, and filtration to eliminate the excess water without any decantation of the materials. The technical and economic limitations lead to select three composites, already evaluated for other applications : (see ref. /1/).

- Cellulose fibre, Ordinary Portland Cement and silica fumes, with a special thermal treatment
- Cellulose, polyethylene, polyvinylalcohol fibres, OPC and silica fumes
- Cellulose, polyethylene, alkali-resistant glass fibres, OPC and silica fumes.

The problems of pure cellulosic compounds on one hand, and the lack of long term knowledge on organic materials on the other hand, lead us to evaluate the third material only.

2. Special concrete reinforced with steel fibres (B.2)

LAFARGE NOUVEAUX MATERIAUX, associated with TREFILERIES DE CONFLANDEY, developed new steel fibres - concrete composite, called "MEF 32", based on the use of a high compacity mortar, reinforced with 12 % steel fibres, allowing high performances to be obtained (see Ref./1/).

The main mechanical properties of the mortar and composite are summarized in Table I.

3. **Concrete reinforced with glass fibres** (B.2)

The main interest of this type of composite material, manufactured by a simple molding process (dispersion of fibres in water, cement and additives feeding, mixing and molding) is to compare the products with those obtained by the EVERITE process.

Two types of fibres were studied : alumina and alkali-resistant glass fibres. The high mechanical properties of both types are attractive, but the cost of alumina fibres is prohititive for this application. It is proposed to evaluate a composite material using glass fibres, blast furnace slag cement and additives, based on SGN's own experience in cementation.

4. **Experimental programme** (B.3)

The experimental evaluation of the three materials was started at the end of 1987, and will mainly be performed in 1988. The programme is based on the fundamental safety rules in application in France for the subsurface disposal. It will include (see ref /1/) :

- Mechanical tests (compressive and tensile strength)
- Physical tests (porosity, permeability)
- Thermal tests (freeze/thaw cycles, resistance to temperature)
- Chemical tests (corrosion by nitric acid)
- Irradiation tests (gamma ray resistance)
- Immersion tests.

Table I : Properties of the "MEF 32" reinforced mortar

	Ordinary concrete	MEF 32	
		Without fibres	With 12 % Steel fibres
Compressive Strength (MPa)	20	150	200
Tensile strength (MPa)	3,0	15	42
Young modulus (MPa)	30 000	58 000	65 000
Deformation at failure (%)	< 0.5 %	< 0.5 %	5 %

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 - June 1988
Project Leader: L. Küppers

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).

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.

C. Progress of Work and Obtained Results

Summary

An estimation of the demand for cast iron containers usable for wastes coming from the dismantling of nuclear facilities was made by means of a computer program. The reference type A-container had already been chosen in 1986. Based on an estimation of masses of operational wastes from NPP. For this container the detailed design and the calculation of stability was carried out in 1987.
Furthermore, after manufacturing the first prototype container, a test program was prepared in agreement with the authority requirements.

Progress and Results

1. Estimation of demand for cast iron containers as packagings for wastes, issued from the dismantling of nuclear facilities (B.1)

For the choice of container types suitable for the storage of wastes issued from dismantling of nuclear facilities the waste-spectrum of the BWR Brunsbüttel and of the PWR Biblis A was taken as a basis. For both kinds of NPP, the BWR and the PWR, the demand for type-A-containers made out of cast iron has been ascertained for immediate dismantling and also for dismantling after a thirty years' inclusion. Containers suitable for disposal in the KONRAD-mine have been considered as well as containers for the GORLEBEN disposal site.

1.1 Container for disposal in KONRAD

The dismantling wastes suitable for storage in cast iron container are medium level radioactive structural wastes in majority with a relatively high density. Caused by the limitation of the total weight of a container disposed in KONRAD (20 Mg) therefore only the type I and type II container with a wall thickness of 150 mm seem to be suitable. These types have a load capacity of 6,18 Mg and 4,59 Mg respectively at a filling density of 2 Kg/l to 3,2 Kg/l (by an assumed shielding-thickness of 150 mm). These values are to be seen in table I.
The estimation of container quantities was carried out by means of the NIS-computer program CONTY /1/ and the results are given in table II.
For the container type II with 150 mm wall thickness a quantity of 363 containers are to be expected for both reactor types in case of immediate dismantling.

1.2 Containers for disposal in GORLEBEN

For disposal in the GORLEBEN mine, however, the total weight of packages is limited to 40 Mg, so that here the bigger type VI as cast iron container gets interesting for dismantling wastes, too. The estimated quantity for this type A-container is shown in table II : 11 containers for the PWR and 10 containers for the BWR.
The container type VI had already been chosen as an optimized container for operational wastes and therefore it was taken as the reference container.

2. Design and manufacture of type-A-container made out of cast iron (B.2)

The basic design of the waste container type VI chosen in the estimation of masses of wastes is to be seen in figure 1.
This container is usable as a type-A-package as well as an industrial package type II.
The design of the corner fitting fixation was optimized for the prototype container, that means two solutions were developed :
- casting the ISO-corner fittings together with the container and
- fixing the ISO-corner fittings with screws.

For the first method the corner fittings are moulded as sand mould and are casted together with the container itself in one production step out of cast iron with nodular graphit. The qualification of this material for the ISO-corner fittings will be evaluated later during the handling tests.
The ISO-corner fittings of the alternative fastening method are made of material GS 45.3 and the dimensions and test loads correspond to DIN ISO 1161 /2/. These corner fittings are welded (10 mm fillet) to a distance plate. The distance plate itself is fixed to the container by means of screws.

3. Defining specifications for manufacturing and testing (B.2)

The necessary specifications of type A-containers were defined, including :
- ultrasonic tests,
- surface tests,
- flow process charts (FPP = Fertigungsprüffolgepläne) and check list for fabricating the container and the container-lid,
- flow process charts (FPP) and check list for assembling the container and for the test program,
- material data sheets (WB = Werkstoffdatenblätter) for all essential container elements for the stability and the handling,
- desired/actual dimension protocols of container and container-lid.

4. Calculation of stability (B.2)

The calculation of stability should prove that the container is able to withstand the loads resulting from normal transport conditions. For this, the stresses in the container walls and the stability of the different fastening methods of the corner fittings were examined.
- Stresses in the container walls :

The maximum stresses (in a considered flat load-bearing structure) were found in the cross-sections below the corner fittings. The maximum stress value of 47,3 N/mm2 was calculated for the case of lifting the container by the bottom corner fittings according to ISO 1496./3/.
However, this value is less than the authorised stress of 225 N/mm2 for the nodular cast iron GGG 60.
- Stability of the fastening of corner fittings :

The stability of the fastening of corner fittings was assessed arithmetically for the two described methods. In case of handling the container with a side wall spreader the maximum stresses were found in the weldings and screws, because of the container lifting only by two top corner fittings, but a sufficient safety margin to the authorised yield strength in screws and cast iron was demonstrated by calculation.

5. Manufacture of the prototype container (B.3)

Having achieved specification and construction documents the necessary casting patterns were produced and the casting was carried out . The cast iron for the container was melted with an addition of 4,22 Mg of scrap coming from the dismantling of nuclear facilities. This corresponds to a percentage of 20,4 % of the melted iron. By means of sprue-specimen (Angussproben) the container and at the container-lid the achieved mechanical values were determined. The investigated values of tensile strength, tensile yield stress and elongation fulfil the required values for nodular cast iron GGG 60 (table III).

For testing both methods of fastening the corner fittings the prototype was produced with 4 corner fittings out of GS 45.3 and with 4 corner fittings out of nodular cast iron.

After mechanical treatment the assembling of the container with lid and sealing was carried out. During all fabrication steps the quality assurance proved the respect of the requirements according to the specifications of flow process charts and material data sheets.

6. Preparing of the test program (B.3)

The container will be tested with respect to the requirements and conditions of the final depository KONRAD /4/, of regulations for safe transport of dangerous materials /5/ and according to ISO 1496. Regarding these requirements a test program was defined and the construction of test devices carried out. The test program that will be carried out in 1988 includes in detail :

- Testing of the longitudinal restraint (Längswiderstandsfähigkeit) of the container bottom,
- Lifting by the bottom corner fittings,
- Handling tests with a side wall spreader,
- Stacking test,
- Free drop test from 0,8 m distance to a target steel plate.

References

/1/ DIN ISO 1161, ISO-Container der Reihe 1 - Eckbeschläge Anforderungen, Juli 1981

/2/ ISO 1496/1, Series 1, Freight Containers - Specification and Testing

/3/ PTB-Braunschweig, Anforderungen an endzulagernde radioaktive Abfälle (Vorläufige Endlagerungsbedingungen - Schachtanlage KONRAD - Stand November 1986)

/4/ Gefahrgutverordnung Straße - GGVS - mit Anlagen A und B, BGBl. I, Nr. 40 vom 30.07.1985

Table I : Dimensions of cast iron containers for dismantling wastes for the Konrad mine

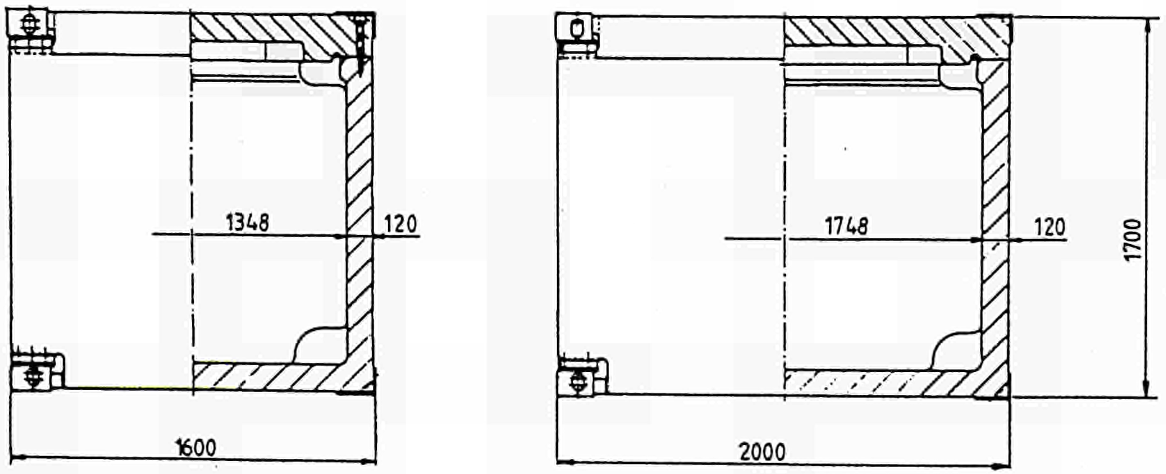
Type	Dimensions L x B x H (m)	Disposal volume (m^3)	wall thickness (mm)	empty weight (Mg)	load capa-city (Mg)	inner volume m^3	filling density (load/inner vol)
I	1,6 x 1,7 x 1,45	3,9	150	13,82	6,18	1,91	3,2
II	1,6 x 1,7 x 1,7	4,6	150	15,41	4,59	2,36	2,0
III	1,7 x 3,0 x 1,7	8,67	100	17,30	2,7	6,30	0,4
IV	1,7 x 3,0 x 1,45	7,39	120	18,3	1,7	4,88	0,35
V	2,0 x 3,2 x 1,7	10,88	80	16,60	3,4	8,89	0,38
VI	2,0 x 1,6 x 1,7	5,44	150	17,40	2,6	2,88	0,9

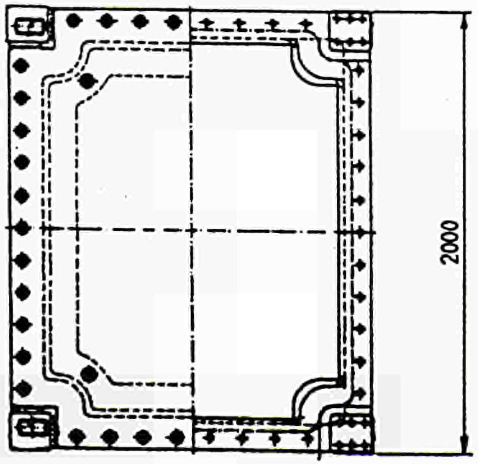
Table II: Estimated quantities of cast iron container suitable for dismantling wastes from BWR Brunsbüttel and PWR Biblis A

		Container type II wall thickness 150 mm KONRAD-mine	Container type VI wall thickness 80 mm GORLEBEN-mine
immediate dismantling	BWR	261	10
	PWR	102	11
dismantling after thirty years' inclusion	BWR	70	6
	PWR	69	5

Table III: Mechanical properties of the container's material

Test no.	Rm N/mm2	$R_{p0,2}$ N/mm2	A_5 %	Z %
required GGG-60 DIN 1993 T.2	550	340	1	-
obtained values				
010-1	558	370	3,0	3,0
010-2	536	379	2,9	3,1
010-3	554	361	3,0	4,8
011-1	539	370	2,6	4,6





Gebindemasse	: max.	20 000 kg	total weight
Eigengewicht	: ges.	14520 kg	empty weight
Nutzlast	:	5480 kg	load weight
Nutzvolumen	:	3280 l	load volume

Figure 1: Design of a cast iron container

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

Eight research contracts relating to Project N°6 were being executed in 1987, including one new contract concluded in 1987, as well as two contracts whose execution has been completed in 1987.

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.

3. <u>Comparison with gross-counting</u> (B.5)

A comparison of gross-counting and Bayesian peak-fitting indicates that for a fixed counting time the detection limit for peak-fitting is less than that for gross-counting by up to 30%, and that for a fixed detection limit the counting time for peak-fitting is less than that for gross-counting by up to 50%.

Hence, the time-related costs associated with activity measurements can be reduced by up to 50% by using Bayesian peak-fitting instead of gross-counting.

4. <u>Extension to overlapping peaks</u> (B.5)

For a single sample, gross-counting can only be used to calculate the detection limit of an isolated spectral peak lying on a linear background. However, the hypothesis test based on the general linear model can be applied to a number of overlapping spectral peaks as well as to an isolated spectral peak. Hence, Bayesian peak-fitting can be used to calculate the detection limit of a spectral peak which lies among a group of overlapping spectral peaks.

5. <u>Testing of multiple hypotheses</u> (B.8)

In most applications it is of interest to determine whether there is any evidence that one or more of a number of radionuclides are present in a given sample. The hypothesis test has been generalized so that a number of hypotheses, one for each radionuclide, can be tested simultaneously.

However, if the detection limit for each radionuclide is to remain unchanged, then the sample counting time must be increased: for example, to keep the same detection limit per radionuclide when testing for the presence of two radionuclides, the sample counting time must be increased by about 40%; if the number of radionuclides is increased to five, then the sample counting time must be increased by about 100%. Clearly, this variation of the detection limit with the number of radionuclides under investigation has important implications for experimental design and interpretation.

6. <u>Final Conclusion</u>

The study carried out within the foreseen timescale showed the expected results. One among the most interesting ones may be the possibility of reducing time-related costs up to 50% when making activity measurements by the peak-fitting method (for a fixed detection limit) instead of gross-counting.

Testing of the method under real conditions is now the logical follow-up to the work achieved.

C. Progress of Work and Obtained Results

Summary

The assessment of spectrometry system stability has been completed. It has shown that even in the case of a spectrometry system that is subject to sudden, rapid spectral drift it is still possible to restrict the standard deviations in the peak centroid and half-width to 2% and 6% respectively over a period of two months by using a digital spectrum stabilizer (B.2).

The type II error probability function derived from the general linear model has been shown to provide a very close, and quite adequate, approximation to the true type II error probability function determined by simulation (B.5).

It has been shown that for a fixed counting time the detection limit obtained with peak-fitting is less than that obtained with gross-counting by up to 30%, and that for a fixed detection limit the counting time required for peak-fitting is less than that required for gross-counting by up to 50%. Hence, the time-related costs of making activity measurements can be reduced by up to 50% (B.5).

The calculation of detection limits in the case of a single sample has, in the past, been restricted to applications where an isolated photopeak lies on a linear background, since gross-counting can only be applied in these circumstances. However, for peak-fitting in the case of a well stabilized spectrometry system, detection limits can be calculated for a peak that lies among a number of overlapping peaks (B.5).

The hypothesis test has been generalized so that a number of hypotheses, one for each radionuclide, can be tested simultaneously. It has been shown that if the detection limit for each radionuclide is to remain unchanged, then the sample counting time must be increased by 40% (100%) when testing for the presence of two (five) radionuclides rather than one in the same sample (B.8).

Progress and Results

1. Assessment of spectrometry system stability (B.2)

Measurements of spectrometry system stability have been made over a period of two months using a digital spectrum stabilizer. During this period, the standard deviations in the peak centroid and half-width were 2% and 6% respectively, even though the spectrometry system was known to be subject to sudden, strong spectral shifts.

These variations in the peak centroid and half-width are so small as to be undetectable when fitting a small-area spectral peak. Hence, it is unnecessary to randomize critical values and detection limits with respect to the Bayesian probability distribution of these parameters.

2. Error probability function (B.5)

The detection limit associated with peak-fitting is derived from the type II error probability function of the test statistic. This test statistic is a modified form of the likelihood ratio statistic associated with the general linear model - the covariance matrix is not assumed to be proportional to the identity matrix as is the case in the general linear model but is replaced by a suitable estimate.

The type II error probability function derived on the assumption that the covariance matrix is known is related to the non-central F-distribution. This approximation has been compared with simulated values of the true type II error probability function. The approximation has been shown to be quite adequate given the precision required by spectrometry applications.

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 - September 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

Analyses of steel, graphite and concrete samples were completed during 1987. The range of cobalt-59 impurity in WAGR steels was larger than expected, up to 0.3%. Small changes in assessed activity inventories will result. The analyses of graphite and concrete suggested that tonnages of active waste may be reduced, with cost savings in consequence. An evaluation of a LASER-based microanalyser concluded that the sensitivity required for cobalt-59 determination was unlikely to be realised. Steel sampling provision at WAGR was likely therefore to concentrate on drilling, and laboratory analysis of swarf samples. Plans were made for sampling the WAGR loop tubes, the highest activity items.

WESTD code development was completed; the code in several versions will be tested during actual dismantling, rather than sooner.

Progress and results

1. Analytical development (B1, B2, B3)

Further analysis of steel, graphite and concrete samples from WAGR were completed during 1987. The most important findings were:

(1) for steel components, that the variation in cobalt impurity content probably has a larger range than expected, extending from less than 200 ppm to 3200 ppm in one stainless steel shield plug sampled.

(2) for graphite, that the principal gamma emitting isotopes now present are cobalt-60 and europium 154.

(3) for activated concrete, that europium 152 activity dominates the gamma emission.

Neutron shield plugs

Among the items of intermediate level waste are plugs above the fuel, in 200 of the 253 channels in the reactor. An inactive plug manufactured in 1961 as one of the original supply has been analysed for 12 important elements, including cobalt. The most significant finding is that no less than five different stainless steels were used in the manufacture, with a larger than expected range, from 0.02 to 0.32%. The most significant results are collected in Table 1, which refers to analysis of upper and lower helix sections of a WAGR shield plug, the outer shroud tube, and an end-connector component. The range of compositions is clearly seen. The accuracy of the analyses was checked against reference steels provided by the National Bureau of Standards (NBS C1153 and 1155). When the plugs are packed for disposal, careful account of this wide range will be taken, when estimating amounts of other radioactivities from the measured cobalt-60 emission.

Graphite: gamma activity and disposal options

During the year, analyses of principal gamma emission from WAGR graphite moderator samples was completed. The maximum activities for samples near the core centre were from Co-60, to to 23 KBq/gm, and Eu 154 up to 0.96 KBq/gm. The detailed results suggest that some core graphite may be below previously assumed waste activity limits; however, analysis for H3 and C14 are required, to confirm this, before cheaper disposal options can be envisaged.

Concrete: gamma activity and disposal options
Samples of WAGR concrete, from inner to outer face of the 2.8m thick side shield, have been examined. Preliminary measurements of the dominating Eu-152 activity of the innermost samples have been made in several UK laboratories. The maximum Eu152 activity was estimated at 1.8KBq/gm, resulting from Europium impurity at approximately 1 part per million. A revised estimate of the tonnage of concrete above the 0.4 Bq/gm 'de minimis' limit has been made. The original estimate was 750 te. Analysis shows that this could be revised downwards to 580 te if H3 is assumed present, and further reduced to 400 te if not.

2. Sampling/Analysis scheme (B4, B5)

The Laser microanalyser, developed by the UK Central Electricity Generating Board for steel analysis in UK reactors, has been evaluated during the year for possible use in WAGR decommissioning. The requirement for cobalt-59 determination to ±20% at 100 ppm was set. This device, developed for silicon and other element analyses, is unlikely to achieve the WAGR requirements. Provision is therefore being made for steel sampling using a simple drill arrangement, together with surface sampling using an abrasive (diamond-impregnated) paper. The abrasive paper method was demonstrated during 1985 (report EUR 10740) and has since been further developed.
Among the most active steel items in the reactor are the stainless steel loop tubes. Plans were made during the year for sampling two of these tubes by direct drilling. This sampling and analysis (due by mid 1988) will confirm the preferred sampling/analysis scheme to be generally recommended.

3. Tests of codes for curie assay (B6, B7, B8)

Code development was completed during the year. Tests of the WESTD code, which calculates source strength from measurement of gamma fields at the surface of a component or waste box, showed satisfactory agreement with the comprehensive shield code, RANKERN. Tests under item B8 are likely to await opportunity during actual dismantling, rather than earlier. This postponement will avoid extra handling of sources, and exposure of personnel to radiation hazards, which would otherwise be necessary.

TABLE I

Analysis of WAGR shield plug steel for principal elements additions and impurity cobalt

Ref No.	Identity and position	% Cr	% Ni	% Mo	% Nb	% Fe	% Co	% Mn	% Ti	*% S	% P
X7978	1 Top, upper helix	18.3	8.99	0.67	<0.01	67.8	0.10	0.87	0.43	0.02	0.03
X7979	2 Bottom, upper helix	17.9	9.16	0.33	<0.01	68.8	0.10	0.89	0.69	0.01	0.02
X7980	3 Top, lower helix	18.1	9.71	0.33	0.97	67.6	0.32	0.71	<0.01	0.03	0.03
X7981	4 Bottom, lower helix	18.1	9.11	0.36	1.00	68.0	0.10	0.80	<0.01	0.03	0.03
X7982	5B Bellows above top helix	18.5	9.53	1.10	0.70	67.3	0.09	0.52	<0.01	0.03	0.04
X7983	5A Top, top outer tube	18.3	11.4	0.13	0.02	67.3	0.06	1.00	0.46	0.02	0.03
X7984	6 Bottom, top outer tube	18.2	11.3	0.14	0.02	67.2	0.06	1.00	0.46	0.02	0.03
X7985	7 Top, bottom outer tube	18.0	11.6	0.23	<0.01	67.6	0.07	0.96	0.53	0.02	0.58
X7986	8B End connector metal	19.8	8.26	0.03	<0.01	66.2	0.02	1.75	0.81	0.01	0.01
X7987	8A Bottom, bottom outer zone	18.0	11.6	0.23	0.02	67.8	0.07	0.96	0.53	0.02	0.03
NBS C1153	FOUND	16.7	8.79	0.23	0.50	70.7	0.12	0.51	0.01	0.02	0.03
NBS C1153	GIVEN	16.7	8.77	0.24	0.50	71.2	0.13	0.50	0.01	0.02	0.03
NBS 1155	FOUND	18.4	12.1	2.38	<0.01	65.3	0.11	1.61	<0.01	0.01	0.01
Remarks:	GIVEN	18.5	12.2	2.38	-	64.5	0.10	1.63	-	0.02	0.02

*S by X-ray fluorescence

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

Summary

During 1987, essential improvements on the basic rig were made with respect to simplifying and improving contamination measurements.
Progress was also made on the series of round and flat counters by only small modifications to adapt them for a new basic rig. The adapters for connection on stationery measurement equipment received practically oriented modifications, e.g. acustic-control. Beyond that, extensive tests under real conditions (e.g. in nuclear power plants) were carried out and all relevant test data collected.

Progress and Results (B.1., B.2.)

Technical arrangements were made to obtain full compatibility between the basic rig and all R+A counter types (flat, round). Apart from this, the completely modified electronic of the rig allowed a substantial improvement of the gas supply. The former, large, heavy and finally expensive gas supply of the previous basic rig was fully automatised by means of a special electronic timing circuit allowing to adjust the gas flow for the cleaning phase (fast), and the permanent phase low. Also the high voltage is automatically turned on. This means for the user a better counterprotection and the complete omission of mechanical gas-supply elements. Besides, a nearly 2.5 times higher gas volume was obtained with a plexiglas gas tank (having the same weight as the former one).

The basic rig was developed to the newest standard of this technique and based on safety requirements according to the german "Unfallverhütungsvorschriften". A major advantage of the new rig is the integrated rechargeable accumulator. This means more autonomy, a longer working time and the possibility of external or internal power supply. Electronic and counting units were completely digitalised.

Proved complementary Metalloxyde (C-MOS) technology was chosen. The initially foreseen LED-display was replaced by an illuminable 4-digit LCD in order to reduce the energy consumption. A safer control of the gas tank pressure was obtained by LED signalling too low (adjustable) rest pressure. The charging level of the accumulator is also controlled by LED. Only for the high voltage control, a manual switch-over is necessary. The service items are reduced to 5 elements:
1. Power on/off; 2. HV-display; 3. alpha/alpha + beta; 4. short/long gate time; 5. Alarm/loudspeaker.
The display device works with 4 LED (partially DUO-LED's) and 4 digit LCD with which a 4 decade-counting is possible without switching over.

The realisation of a common calibration factor was obtained by a counter-internal timeing resistor, switched parallel to an internal RC-circuit during the connection of the counter to the basic rig, controlling in this way the "gate time" of the counter.

On the whole, considerable improvements of the contamination measurement technique were obtained.

Final Conclusion

The research carried out in the frame of this contract showed essential improvement of the contamination measurement technique related to curved surfaces (e.g. inside tubes > Ø 22 mm at distances of up to 6 m) obtained by developing an integrated portable measuring device composed by a series of round and flat gas flow detectors, connected to a sophisticated basic rig by means of special cables and adapters.

Testing under real conditions (e.g. KRB-Gundremmingen, NPP Biblis, RBU, ALKEM) of a prototype (main characteristics: dimensions 22x20x11 cm high, weight 3.2 kg and autonomy service time $\pm$ 10 h) showed satisfactory behaviour of the device, so that it can be concluded that the programme aim has been reached.

However, it seems to be interesting to extend the application (present reference surface: 200 cm^2 - effective 130 cm^2) of this measurement technique to larger areas by a range of appropriate detectors connectable to the same basic rig.

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 - August 1988
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.

Progress of Work and Obtained Results

Summary

The limits of activity concentration for unrestricted use or handling of waste material as defined in the StrlSchV do not necessarily reflect the radiotoxicity of particular nuclides. Therefore, the radiotoxicities of waste nuclides, based on the definition of these limits and the annual limits of intake as set forth in the StrlSchV, respectively the dosefactors for intake as set forth in ICRP 30, were determined. For easy comparison, they were further normalized to the toxicity of Co60 as the most prominent waste nuclide.

It was found that most of the beta/gamma-emitters are overestimated while the alpha-emitters are underestimated by factors of 5-100.

Combining these radiotoxicities with the nuclide abundances in typical waste categories, it was shown that the measurement of nuclides decaying by low energy beta-particles or electron capture would be radiologically unnecessary, since the usually applied methods of pure beta- or gamma-measurements cover about 90 % of the radiological relevance of the waste nuclides. Only if alpha-emitters in the order of 10^{-3} of gross activity are present, a separate alpha-measurement is undispensable.

For determination of detector efficiencies, suitable area standards with Co60, Cs137, Sr90/Y90, Am241 were fabricated. Detectors used were 2 GM counters, 5 different proportional counters and 3 NaI-detectors. Tube geometry was employed, and the contamination layer was changed from inner to outer tube wall.

For further measurements of alpha emitters, some special detectors were purchased and suitably adapted.

Progress and Results

1. Compilation of nuclide inventories (B.1.)

For different kinds of waste material, a compilation of nuclide inventories was undertaken. It was found that in most waste categories, Co60 and Cs137 can be considered the leading nuclides, the main exceptions being activated steel waste and waste from fuel manufacturing plants.

2. Determination of Relative Radiotoxicities (B.2.)

According to StrlSchV, two different limits apply for use and handling of radioactive materials without authority approval:
- area related activity concentration (Bq/cm^2)
- mass related activity concentration (Bq/g)

These limits do not necessarily reflect the radiotoxicity of a particular nuclide. Therefore, the annual limits of inhalation respective ingestion (ALI) as defined in StrlSchV were taken to compare the toxicity of the waste nuclides. Relative Radiotoxicities (RR) were obtained by normalizing these ALI to those of Co60 as the most prominent nuclide and the two aforementioned activity concentration limits (table I). It was found that - compared to the RR of the ICRP 30-ALI - the RR of most of the nuclides are overestimated, particularly H3, Ca45, Ni59, Ni63, Tc99, Sb125, I129, Cs135/137. The actinides, on the other hand, are underestimated by factors of 5 to 100 depending on the path of intake. For direct irradiation, the dosefactors of all waste nuclides (exception: Ag110m) are smaller than those of Co60.

3. Selection criteria of measurement methods (B.2., B.3.)

If radiation protection aspects are taken into consideration besides the necessary administrative definition of upper limits of activity concentration, then a combination of nuclide abundance in waste and relative radiotoxicities will yield a measure of importance of covering certain nuclides in a measurement. Table II shows, as an example, three different NPP waste categories. In column 3, the RR based on surface contamination limits and weighted by the nuclide abundances and normalized to its sum = 1 are shown. It follows that for compressible raw waste, a gamma- as well as a beta-measurement will cover almost all of the radiologically significant impact of the nuclides present in that waste. The weighted RR of the activated steel waste reveal that either gamma- or beta-measurement will cover about 90 % of the radiological impact though more than 80 % of that activity is not directly measurable (Fe55, Ni63, Ni59). Only when alpha-activity is not negligible (in the order of 10^{-3} of gross activity) and intake by inhalation cannot be excluded, pure gamma- or beta-measurements will miss about 50 % of radiological relevance. Therefore, the experimental work to obtain detector efficienies will concentrate on beta-, gamma- and alpha-emitters; nuclides decaying by low energy beta-emission or electron capture will only be of secondary interest.

4. Fabrication of suitable activity standards (B.4.)

Area standards with high flexibility to be adapted to odd geometric forms were fabricated using chromatographic paper. The homogeneity of activity distribution was verified with autoradiographic film. Nuclides used were Co60, Cs137, Sr90/Y90, Am 241.

5. Determination of detector efficiencies (B.3., B.5.)

Of particular interest are non-standard geometries like curved or cornered pieces. Detection efficiencies of different detectors (2 GM counters, 4 butane prop. counters, 1 Xe-prop. counter, 3 NaI(Tl)-detectors of different dimensions) were determined for tube geometry (9 different diameters) and contamination on the inner and outer tube walls with beta- and gamma-emitters. It was found that the highest efficiency for measurement of inside contamination was obtained by use of a 3"x3" NaI(Tl) detector. For outside contamination, the efficiencies of NaI and area proportional counters are comparable. Direct measurement of inside contamination with GM counters is highly effective but prone to detector contamination or mechanical damage. The use of area proportional counters is limited by their size; small diameter tubes with inside contamination have to be axially cut in halves if measurement is to be effective. Depending on tube diameter, the efficiency will be lowered by a factor of up to 3.

6. Procurement and adaptation of special detectors for alpha- (and beta-) measurements (B.4.)

The following special detectors were purchased: a plastic scintillator, a Si junction detector and photodiodes of 4 different dimensions. The latter are relatively inexpensive and produced with small entrance areas that permit measurements in sharp corners or edges. Since their use as nuclear detectors has not become standard yet, the necessary modifications to assure lighttightness and the fabrication of suitable preamplification stages had to be undertaken.

Table I: Relative Radiotoxicity (table IX StrlSchV) (RR)

nuclide	RR (inh.)	RR (tab. IX) / RR (ICRP 30)	RR (ing.)	RR (tabl. IX) / RR (ICRP 30)
H 3	0,002	7,4	0,01	4,5
C 14	0,03		0,04	
Ca 45	0,27	8,7	3,75	31
Mn 54	0,26	8,4	0,27	2,7
Fe 55	0,01	0,8	0,04	1,8
Ni 59	0,02	3,1	0,17	22
Ni 63	0,14	10	1,2	55
Co 60	**1,0**		**1,0**	
Zn 65	0,14	1,5	0,35	0,66
Sr 90	7,4	1,3	83	17
Y 90	0,08	0,2	1,7	4,3
Zr 93	0,07	0,05	0,04	0,69
Tc 99	0,14	3,7	0,21	4,0
Ru 106	1,6	0,73	2,7	2,7
Ag 110m	0,84	2,3	1,1	2,8
Sb 125	0,35	6,3	0,35	3,5
I 129	11,1	16	182	20
Cs 134	0,68	3,1	3,75	1,4
Cs 135	0,10	4,8	0,30	1,2
Cs 137	0,63	4,2	2,3	1,2
Ce 144	1,4	0,82	2,7	3,5
Pm 147	0,14	0,74	0,15	3,9
Sm 151	0,14	1,0	0,09	6,4
Eu 152	0,74	0,74	0,43	1,7
Eu 154	2,3	1,8	1,5	4,2
Pu 238	445	0,20	0,67	0,045
Pu 239	510	0,21	0,77	0,048
Pu 240	510	0,21	0,77	0,048
Pu 241	9,6	0,20	0,02	0,06
Am 241	144	0,06	0,91	0,01
Cm 242	7,4	0,09	0,14	0,05
Cm 244	96	0,07	0,46	0,01

Table II: Abundance of prominent nuclides in different NPP wastes

1. Compressible raw waste

	abundance	weighted normalized RR	measurement method Beta	measurement method Gamma
Cs 137	0,43	0,32	x	x
Co60	**0,40**	**0,47**	**x**	**x**
Cs 134	0,15	0,12	x	x
Mn 54	0,01	0,004	x	
Sr 90/Y 90	0,01	0,08		x
I 129	5 E-7	6,6 E-6		
Σalpha	1 E-6	6,0 E-4		
	1,0	1,0	0,91	0,99

2. Steel waste of FR 2

	abundance	weighted normalized RR	Beta	Gamma
Fe 55	0,77	0,05		
Co60	**0,15**	**0,89**	**x**	**x**
Ni 63	0,07	0,06		
Mn 54	0,001	0,002	x	
Ni 59	0,0007	8,3 E-5		
Zn 65	0,0001	8,3 E-5	x	
Eu 152	0,0001	4,4 E-4	x	x
	1,0	1,0	0,89	0,89

3. Waste of KRB-A Gundremmingen

	abundance	weighted normalized RR	Beta	Gamma
Co 60	**0,67**	**0,42**	**x**	**x**
Fe 55	0,06	0,0004		
Sb 125	0,006	0,013	x	x
Pm 147	0,06	0,005		
Cs 137	0,06	0,025	x	x
Cs 134	0,03	0,012	x	x
Sr 90/Y 90	0,03	0,144		x
Ni 63	0,01	0,0009		
Pu 241	0,01	0,06		
Σalpha	0,001	0,32		
	1,0	1,0	0,47	0,61

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: P.W. Gray

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

Gamma ray attenuation coefficients of common soil types have been measured as a function of moisture content and gamma ray energy.

Statistical detection limit formulae have been deduced for two cases: firstly where photopeak areas are recorded for each spectrum measured; and secondly where only the total counts in each spectrum are recorded.

Progress and Results

1. Gamma Ray Attenuation Coefficients (B.3.)

Gamma ray attenuation coefficients of three different soil types have been measured in the laboratory using a germanium detector. The three soil types were: silty clay, silty loam and sandy loam. Attenuation coefficients in each soil type were measured as a function of moisture content and of gamma ray energy using sources of Am-241, Ba-133, Cs-137 and Co-60, covering the energy range 60-1330 keV. Moisture content was varied between zero and 36%.

The resulting attenuation coefficients are shown in Table I, and are consistent with literature values.

2. Detection Limits (B.5.)

Statistical detection limit formulae have been deduced for two cases: firstly where photopeak areas are recorded for each spectrum measured; and secondly where only the total counts in each spectrum are recorded.

In the case of photopeak area, the detection limit can be expressed as:-

$$D_p = \Psi^{-2}\{1-\alpha\} + 2 \,.\, \Psi^{-1}\{1-\alpha\} \,.\, \left(\frac{b.t.\,W_i}{1-W_i/W_o}\right)^{\frac{1}{2}}$$

Where Ψ is the standardised normal distribution function;

α is the probability of missing a photopeak which is present (usually taken to be 0.05);

b is the background count-rate per keV in the peak centroid;

t is the counting time.

W_i and W_o are the photopeak windows, in keV, used to determine the net photopeak area; W_i is the inner window defining the limits of the photopeak, while W_o, which includes W_i, covers a region sufficiently wide to determine the magnitude of the continuum under the photopeak.

In the case where total spectrum counts are recorded, the detection limit can be expressed as (symbols remain as defined above):

$$D_T = \Psi^{-2}\{1-\alpha\} + 2 \,.\, \Psi^{-1}\{1-\alpha\} \,.\, (2\ b\ t)$$

Table I : Gamma Ray Mass Attenuation Coefficients in Soils

(10^{-3} m^2 kg^{-1})

Soil Type	Moisture Content	Radioisotope and Gamma Ray Energy (keV)								
		241Am	133 Ba					137 Cs	60Co	
		60	81	276	303	356	383	662	1173	1332
Silty Loam	0%	28.4	21.2	12.3	12.0	11.1	11.6	9.0	-	6.6
	20%	30.9	24.5	14.7	14.4	13.1	12.6	10.6	8.0	7.7
	36%	33.7	27.6	17.3	16.7	15.4	15.0	12.5	9.4	9.0
Silty Clay	0%	26.1	19.5	10.8	10.6	9.9	10.3	8.1	6.1	5.7
	20%	30.8	24.0	14.0	13.5	12.7	12.3	10.3	7.9	7.4
	30%	31.8	25.3	15.6	14.8	13.8	13.6	11.3	8.5	8.0
Sandy Loam	0%	28.1	20.8	11.5	11.2	10.5	10.0	8.6	6.8	6.5
	20%	33.5	25.5	15.2	14.4	13.6	13.6	11.0	8.3	8.0
	36%	35.5	28.3	17.2	17.0	15.7	15.6	12.6	10.0	9.3

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 which have been undertaken in 1987 pertain to the B.1 and B.2 working steps as described in the work programm.
In the B.1 working step an inventory of the components and structures to be removed in the dismantling process of a fast breeder reactor has been worked out. The main features of these parts : location, material, weight, size ... have been collected.
In the frame of the B.2 working step, applied regulations in the member states of the European Community, with respect to radiation protection, radioactive material transportation and radioactive waste storage have been gathered and synthesized.

Progress and Results

1 - Inventory of the structures to be removed (B.1)

An as far as possible exhaustive review of the components and structures to be removed when dismantling a large fast reactor, namely SPX1 has been achieved. Obviously this inventory is limited to either activated or contaminated parts. This inventory encompasses parts located inside the reactor concrete pit, pipes and tanks of the reactor cover gas loop and devices of the fuel handling system. 28 main items have been identified (table I), 3 of which belongs to the reactor cover gas loop and the fuel handling devices, they likely will not be activated but only contaminated and need no more than a coarse description.
A standard form collecting the required data for dismantling planning for every previously quoted item, has been set up. These forms have been filled up (table II). Of paramount importance for radioactive source assessment is the knowledge of the composition of the materials of the components. For this purpose rather lengthy breakdowns are required when a large number of materials is involved, for instance control rod mechanisms (table III).
Furthermore, as far as usual isotopes are of concern, the chemical compositions of the materials have been derived from the analysises performed during the construction phase. With regard to peculiar elements such as silver, niobium, europium whose radioactive daughters are likely to be taken into account for long decay time, their concentrations in construction materials arise from a short bibliography survey (Ref. 1).

2 - Investigation for criteria related to waste treatment shipment and long term storage

A search through the documents quoted hereafter (Ref. [2] [3] [4]) has been achieved and limits regarding radioactive materials subject to authorization and package classification have been collected. With respect to waste classification with view to long term storage the only rule available found are those of La Hague, french disposal facility.

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European Conference, Luxembourg 22-24 may 1984.
Proceedings pp. 61-74.

[2] Journal officiel des communautés européennes
n° L/246/1 du 17/09/1980.

[3] OECD AEN 1980
Réglementation relative au transport des matières radioactives.

[4] P. JOURDE-CEA
2ème Colloque de Vincennes "Techniques mises en oeuvre pour le conditionnement, le stockage et la gestion des déchets radioactifs", 12 décembre 1984.
Recueil des communications pp. 7-24.

TABLE I : LIST OF COMPONENTS

PRIMARY CIRCUIT

Dismountable Components	Index Card n°	Non-dismountable components	Index Card n°
Primary pump	2	Hot ring	11
Pump-diagrid connexion	3	Safety vessel	12
Intermediate heat exchanger	4	Safety cooling system	13
Transfer mecanism	5	Main vessel	14
Road mecanism	6	Inner vessel	15
Integrated purification unit	7	Safety heat exchanger	16
Rotating transfer lock	8	Baffles	17
Fuel failure location	9	Core support structure + core catcher	18
Fuel storage transfer machine	10	Diagrid + support	19
		Diagrid support of protection	20
		"Dead body" *	21
		Core cover plate	22
		Small rotating plug	23
		Large rotating plug	24
		Roof slab	25
		Fuel storage	26
		Na auxiliary capacities	27
		Fuel transfer system	28

* : "Dead body" : Intermediaire structure beetwen diagrid and inner vessel.

TABLE II : COMPONENTS

COMPONENT CONTROL ROD DRIVE MECHANISM

Moto-reductor set used to introduce partially or totaly the absorbing set packed into the supported assembly to control or stop the reactor. Erected on the core cover plug. 21 SCP * (10 SCP1 + 11 SCP2) + 2 spare parts 3 SAC * + 1 spare part

DIMENSIONS :	overall diameter	350 mm
	Height	15 000 mm
	Mass	2.2 t

COMPOSITION : see after

ACTIVATION	: yes (high)
CONTAMINATION	: yes (low)
PRESENCE Na	: yes

* SCP = main control system

* SAC = complementary stop system.

TABLE III : MATERIAL OF COMPONENTS

CONTROL ROD MECANISM
BASE MATERIALS

	MECANISM SCP TYPE 1	MECANISM SCP TYPE 2	MECANISM SAC
Tight case	Z2 CND 17.12	Z6 CN 18.10	Z6 CN 18.10
Rack and pinion block	Z2 CND 17.12	Z6 CN 18.10	Z6 CN 18.10
Travelling tube	Z2 CND 17.12	Z2 CND 17.12	Z2 CND 17.12
Rack	GKH (AUBERT DUVAL)	32 CDV 12	32 CDV 13
Rack pinon drive	Z4 CND 16.05	40 CDV 12	40 CDV 12
Clip screw drive	32 CDV 13	Z2 CND 17.12	–
Bolt drive of the previous screw	Ni Cu 29AI 3 Fe Mn	UA 9 N5 Fe	–
Clip fingers	Inconel 718	Z6 NCTDV 25.15	–
Bush fingers	Stellite grade 12	Stellite grade 6	–
Clip axel	Inconel 718	Z6 NCTVD 25.15	–
Elements of kinematic chains	25 NC 6 30 NC 11 32 CDV 13 35 NCD 6	Z 12 C 13 35 CD 4	Z 12 C 13 35 CD 4

6.7. Methodology for Assessing Suitable Systems for Management of Reactor Decommissioning Wastes

Contractor: NRPB
Contract N°: FI1D-0051
Working Period: January 1987 - December 1988
Project Leader: G.M. Smith

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

The first task of this study was to identify the appropriate radiological protection criteria for decommissioning. These criteria are annual individual dose and risk limits and upper bounds, and the requirement that the protection of public and workers should be optimised. Exemption criteria, or the definition of material which does not require to be disposed of as radioactive waste, is also important because of the large quantities of low activity material involved. A range of exemption levels is to be included in the assessment of waste inventories to determine the implications of different exemption levels on management strategy. Decommissioning wastes from typical PWR, AGR and Magnox reactors have been reviewed. Many differences are apparent: some are due to fundamental differences among reactor types, others are due to variations in design of reactors of the same generic type. Still others result from differing assumptions made regarding what constitutes radioactive waste and what constitutes non-radioactive material.

Progress and Results

B.1 Factors relevant to waste management decisions

The forms of radiological protection criteria which should be used throughout all stages of decommissioning are dose limits, supplemented by dose upper bounds, risk limits and risk upper bounds, together with the requirement that protection of public and workers should be optimised, ie that all exposures should be as low as reasonably achievable (ALARA), economic and social factors being taken into account. The current principal ICRP recommendation is that the individual effective dose equivalent should not exceed 50 mSv per year for a worker and 1 mSv per year for a member of the public averaged over a lifetime. Regarding optimisation, at highest levels of decision it will be necessary to use a quantitative decision-aiding technique which can take into account all the relevant factors. The procedure is illustrated in Figure 1, and the relevant measures of radiological impact in Figure 2. At lower levels of decision a simple cost-benefit analysis may be sufficient as a basis for a choice between options and, day-to-day, ALARA may in practice be equivalent to exercising health physics judgement.

Exemption of material from consideration as radioactive waste is a further criterion which can have an influence on management decisions. No general numerical recommendations have yet been made by the major advisory bodies. ICRP and IAEA have issued tentative guidelines which broadly agree on the level of exposure below which a source may be exempted, but a major outstanding issue is the definition of a source. ICRP note that sources which fail to meet exemption limits may still be treated without special measures to obviate radiologial impact provided that analysis shows that the detriment is less than the cost of avoiding it.

B.2 Definition of wastes

Published information is being condensed into a form showing decommissioning wastes from three reactor types: PWR, AGR and Magnox reactors. Major differences in quantities and characteristics are apparent, reflecting the differences in size and type of materials used in major components, eg the graphite moderator in the Magnox design as opposed to the more compact steel pressure vessel of the water-moderated PWR. The degree of activation and contamination of components within or external to the reactor building varies considerably from such design differences. However, whereas for any one type of reactor the inventory data for the more active materials is fairly consistent between reports, quantities given for the larger amounts of less active materials are much more variable, probably because the exemption criterion that has been assumed varies significantly between different reports from 0.4 Bq/g to 74 Bq/g. Some authors do not give the exemption criteria they have assumed.

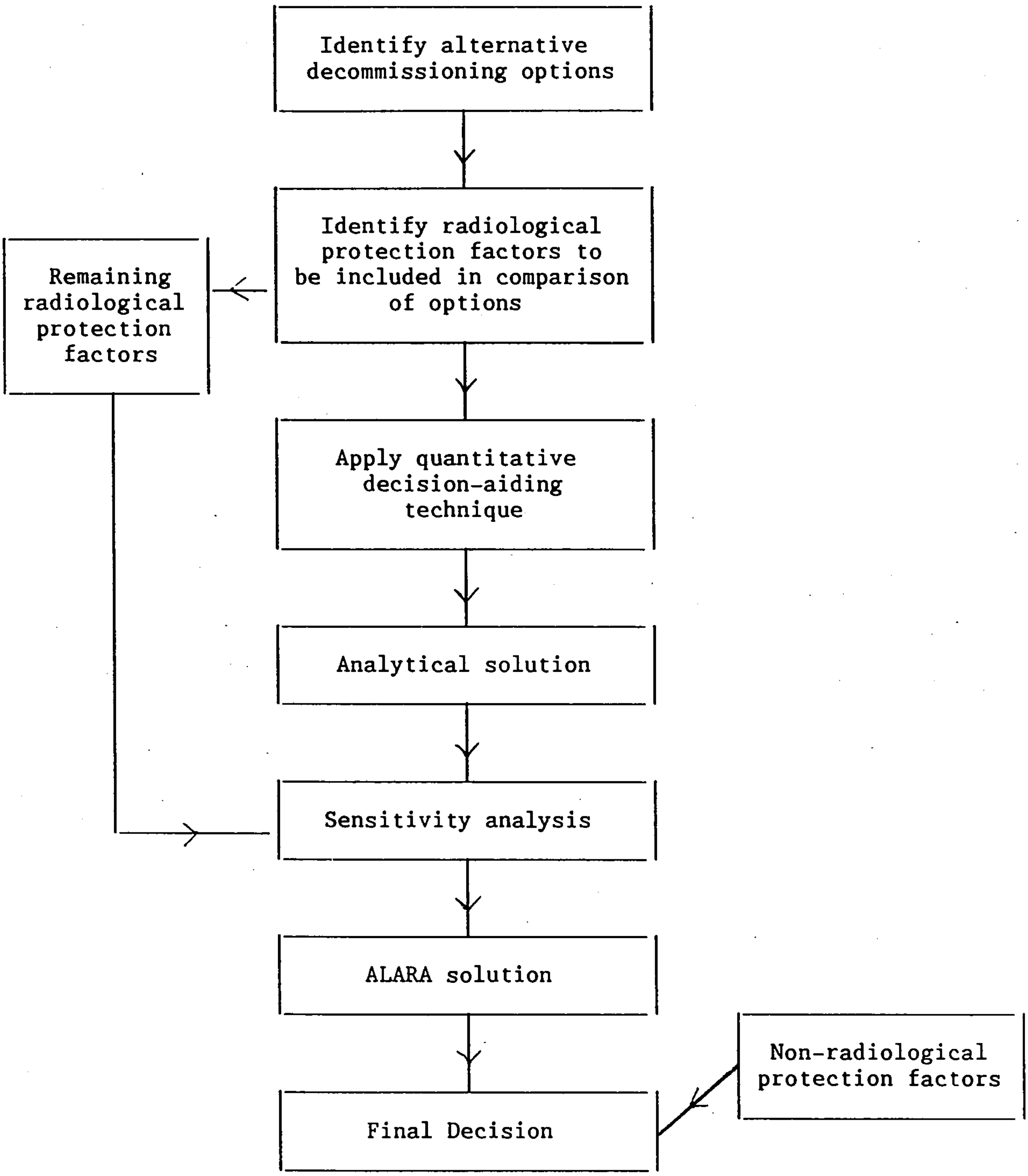


Figure 1

Schematic Representation of ALARA Procedure for Providing an Input to Decommissioning Decisions

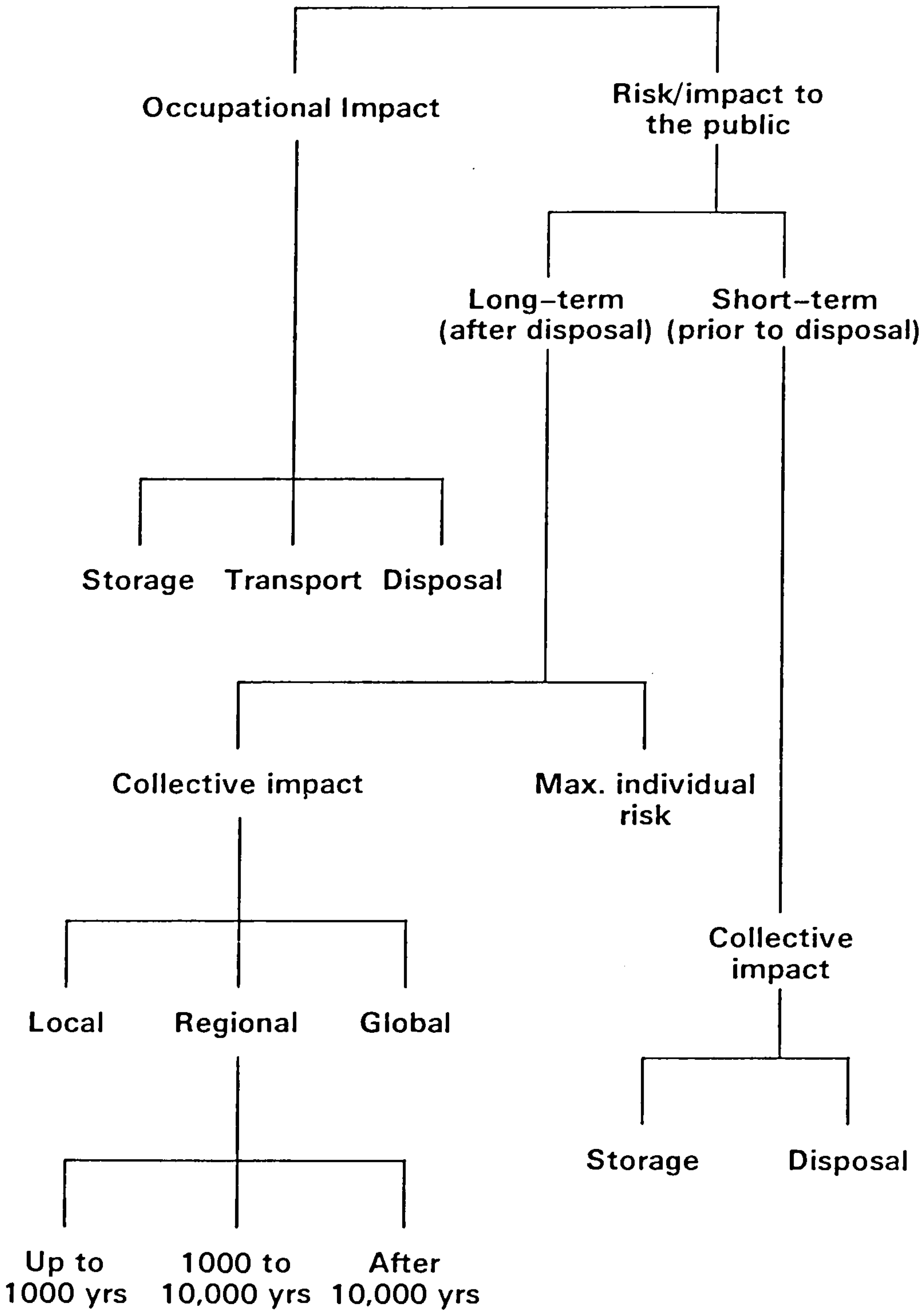


Figure 2. A structure for the radiological impacts pertinent to comparative assessments of waste management options

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 - May 1988
Project Leader: A.M. Chapuis

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 1987 has been concerned with data collection in reactor plants and in uranium enrichment plants, with calculations of neutron fluence rates in the reactor building and with the study of laboratory and semi-industrial fusion tests of aluminium, copper and alloys.

Progress and Results

Data collection in reactor plants first consisted in an evaluation of the quantities of copper, aluminium and alloys employed for the construction of the reactor building. The main origine of these materials is electrical cabling.

In the reactor building these electrical cables can be submitted to neutron fluxes. The fluxes can be calculated from dose rates which have been measured at different points on the different levels of the building.

Neutron fluence rates are comprised between 4 E+2 and 2 E+6 $n.cm^{-2}.s^{-1}$. Resulting activity due to impurities activation will probably be a low part of the total activity, the main part coming from residual contamination probably due to different incidents. As far as contamination is mostly cleaned after each incident, the residual contamination level and the different possible compositions are unknown. In the further calculations, compositions can be estimated from radionuclides' spectra known from different incidents in PWR power plants.

Most of the cables are protected with a PVC isolation sheath except earthing cables.

When isolated cables are recycled, different technics can be involved ; the most usual are the technics of separation of isolation sheath from metal. As most of the contamination will stay on the sheath, the possible uses or eliminations of this product are of interest in this study.

The recycling of aluminium and copper scraps by fusion have been tested in an induction furnace of 700 kg capacity. These tests have shown that both aluminium and copper fusion produce small particule aerosols, the inhalable fraction being comprised between 0.9 and 1.

Mass concentrations are respectively 2.5 $mg.m^{-3}$ and 33 $mg.m^{-3}$ in these experimental conditions.

From the point of view of radiological protection, studies have shown that copper could be decontaminated from uranium contamination, but it seems that it would not be so easy for contaminated aluminium.

Other melting technics will be studied, using contaminated materials coming from an enrichment installation dismantling ; they will give informations on the partition of some contaminants during the melt.

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 being executed in 1987, including one new contract concluded in 1987.

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

Due to ENEA budget difficulties, the order for construction of the dismantling device was not placed till December 1987 and the commissioning is expected by mid-1988. Consequently, no research work was performed in 1987.

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 - December 1988
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

In 1986 nine valves (2-3 inches) had been ordered with the selected deposits: three globe-valves, three valve bodies with bonnet and three swing-check valves /1/.

During 1987, five valves were manufactured, four remaining valves are in progress (B.3.). The test procedure was defined, including endurance tests, thermal shock and erosion tests. The reference valves with Stellite were implemented in the DOUBLEAU loop and the endurance tests began in December 1987 (B.4.).

Progress and Results

1. Valve manufacturer supply (B.3.)

In relation with valve supply, the work performed included following items:
- progress of valves manufacturing;
- verification of documents in association with the supply, e.g. plans, procurement specifications of materials, manufacturing process, qualification of welding processes, laboratory reports on cobalt-free alloys and manufacturing report;
- special follow of hardfacing deposit performance;
- resistance to thermal shocks of hardfacing surfaces in valves;
- dimensional statements of hardfaced surfaces and parts in contact with hard alloys;
- hydrotests and water tightness control of the valves at last;
- procurement of an electric actuator for automatic opening and closing of the globe-valve during endurance loop tests and microcontacts for control.

Five valves, including electrical and automated equipment, were manufactured without any problem:
- two globe-valves (one with Stellite, the other one with Colmonoy),
- one valve body with bonnet (Stellite),
- two swing-check valves (one with Stellite, the other one with Cenium alloy).

For the remaining valves, the manufacturer met some problems with the hardfacing deposits: cracking, unsticking, etc. These problems were solved by modifying the welding procedures for the swing-check valve with Colmonoy. This valve will be delivered at the beginning of next year.

On the other hand, the welding procedures are still under investigations for the two other valves with Cenium: one globe-valve and one valve body with bonnet.

As regards the last valve body with bonnet, the planned deposit material (Everit 50) is not available on the market. This valve, with an alternative alloy (Colmonoy 4), will be available at the beginning of next year.

2. Implementation of valves in the DOUBLEAU loop (B.4.)

Framatome assessed the principles of the test programme to be performed on valves mounted on DOUBLEAU loop with respect to PWR working conditions. On the basis of these principles, CEA established the detailed test procedure /2/ including endurance tests, thermal shock and erosion tests. The performance of cobalt-free alloys will be evaluated comparatively to cobalt-base alloys by successively testing valves with Stellite, Colmonoy and Cenium.

The first three valves with Stellite including the swing-check valves (3 inch), the globe-valve (2 inch) with its actuator, and the valve body (2 inch) with bonnet have been assembled as shown on Figure 1 on the branch test during the last quarter of 1987. Letters C, R and R' indicate the respective location of these valves on the branch test.

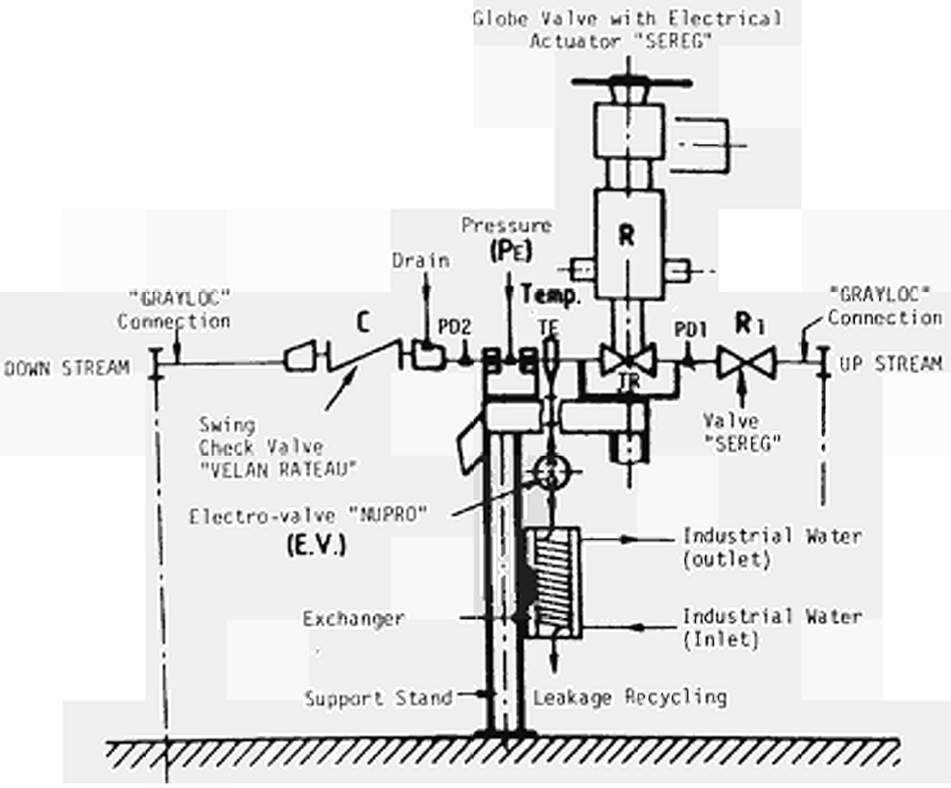
After preliminary control tests, endurance tests on these valves began in December 1987; Figure 1 gives the sequence of valves opening and closing during one cycle; 1500 cycles are planned in the programme.

The principles of thermal shock tests are described for one cycle (hot shock + cold shock) in Figure 2; 20 cycles are planned in the programme with $\Delta\Theta$ = 100°C and $\Delta\Theta$ = 250°C.

As regards the erosion tests, the programme carried out by the valve supplier on its facilities is now completed. In connection with erosion loop tests at 70 and 320°C, this work was aimed at establishing the size of the orifice to be machined on a special globe-valve front disk. Such special disks are now available for Stellite and Colmonoy alloys.

References

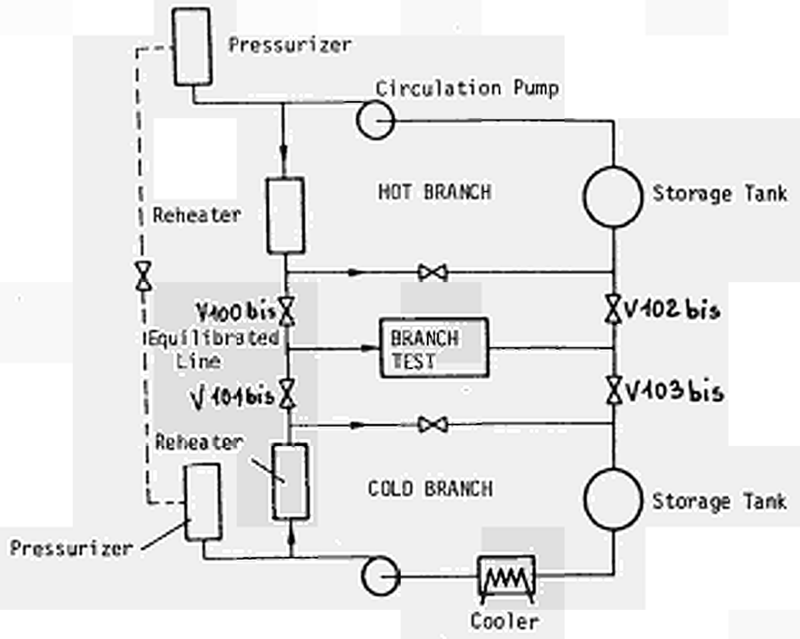
/1/ The Community's research and development programme on decommissioning of nuclear installations. Second annual progress report (year 1986). EUR 11112.



Globe R	Swing C	Electro-valve E_v	PHASE
O	O	C	Flowrate in R,C (320°C-150 bars)
C	C	C	R closing without ΔP-No Flowrate
C	C	O	Depressurization between R and C, P ≤ 5 bars
C	C	C	Electro-valve closing
O	O	C	R opening with ΔP ~ 150 bars

O : Opened - C : Closed

Figure 1 : Principles of endurance tests



HOT BRANCH		COLD BRANCH		PHASE (O : Opened - C : Closed)
V100 bis	V102 bis	V101 bis	V103 bis	
O	O	C	C	Hot Shock
C	C	O	O	Cold Shock
1 Thermal Cycle = 1 Hot Shock + 1 Cold Shock				

Nota : R Valve will be keep in opened position during these tests

Figure 2 : Principles of thermal shock tests

7.3. Pre-stressed Concrete Reactor Vessel with Built-in Planes of Weakness

Contractor: Taylor Woodrow Construction Ltd, Southall, UK
Contract N°: FI1D-0054
Working Period: October 1986 - December 1988
Project Leader: I.Ll. Davies

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 the six month period since the commencement of this project in July 1987, the work has concentrated mainly on the review of the previous analytical work described in Report Ref. EUR 9399 EN /1/ and on design and construction aspects of the structural models.

The initial strategy of the study was to seek verification of design and analytical parameters relating to the planes of weakness concept by constructing and testing two small scale models. However, the review of the previous analyses suggested that a more general approach would be more appropriate. The strategy therefore adopted has been to review the previous analyses to identify key aspects governing the design, construction and testing of the models and complement this with a comparative analytical study.

The initial intention was to prestress the models circumferentially using an arrangement of wire wound prestress on the outer vertical surface. However, subsequent investigations revealed that no suitable wire winding facilities were commerically available. It was therefore decided to provide the circumferential prestress for the models by the use of curvelinear tendons of a type similar to the straight vertical tendons.

The review of the previous analytical work is now complete and this enabled decisions to be made in respect of choice of the planes of weakness pattern, the material to be used to form the weakness planes and the load cases to be considered in the analytical phase of the present study.

Design of the models is now also complete and the construction drawings have been prepared.

Progress and Results

1. Conclusions of the Review of Previous Analyses (B.1.)

The previous analyses and associated work, described in Report Ref. EUR 9399 EN, were reviewed and the following decisions made in respect of the present study:

- The pattern for the planes of weakness has been selected as that shown in Figures 1(b) and 2.
- The material chosen to separate adjacent blocks, and thus form the planes of weakness, is polythene. The blocks will be cast insitu and the polythene located sequentially as casting proceeds.
- The load cases which will be considered in the present analytical study are initial prestress, prestress and design pressure, and prestress and overload pressure. These load cases are compatible with the model test loadings at which instrumentation data will be recorded.

This part of the study is complete.

2. Analyses (B.2., B.3.)

All relevant design details of the models have now been established and analytical work commenced in the last weeks of 1987. As yet, there is no significant progress to report.

3. Structural Models (B.4., B.5.)

The basic dimensional parameters and other details of the models have been determined and are shown in Figures 1 and 2. The internal cavity of each model will be lined using a 0.7 mm thick mild steel liner whose external surface will embody tangs which will be anchored into an externally applied grout screed. The planes of weakness blocks, with polythene at their interfaces, will be cast sequentially around the liner, followed by the structural concrete with 8 mm single size aggregate.

With a design pressure of 1.9 MN/m^2 and a factor of safety of 1.7, each model requires 18 vertical prestressing tendons and 19 hoop prestressing tendons, the arrangements of which are shown in Figure 2.

Each vertical tendon consists of one plastic coated strand, 8 mm in diameter, comprising seven wires, six of which are spirally wound around a central king wire. The vertical tendons will be stressed at one end to a jacking load of 56 KN, and after lock-off, will be suitably shimmed at that end to recover wedge pull-in at lock-off. The vertical tendons will be housed in plastic ducts passing vertically through the models.

Each hoop tendon consists of two strands, each of which encircles the vessel circumference once, as shown in Figure 2. The hoop strands are the same type, diameter and composition as the vertical strands and will be stressed at both ends to a jacking load of 41 KN. They will, after lock-off, be suitably shimmed at both ends to recover wedge pull-in at lock-off. The hoop strands will not be located in ducts but will be cast into the concrete just inside the reinforcement at the outer vertical surfaces of the models.

The vertical and hoop strands will be anchored in conventional barrel and wedge anchors located on steel plates which distribute the bearing load to the concrete. The vertical strands will be anchored against the top and bottom surfaces of the models while the hoop strands will be anchored against two concrete buttresses running the full height of the outer vertical surface.

In order to distribute any high local forces, and consequent cracking that may otherwise develop, the models will incorporate an arrangement of bonded reinforcement as indicated in Figure 3.

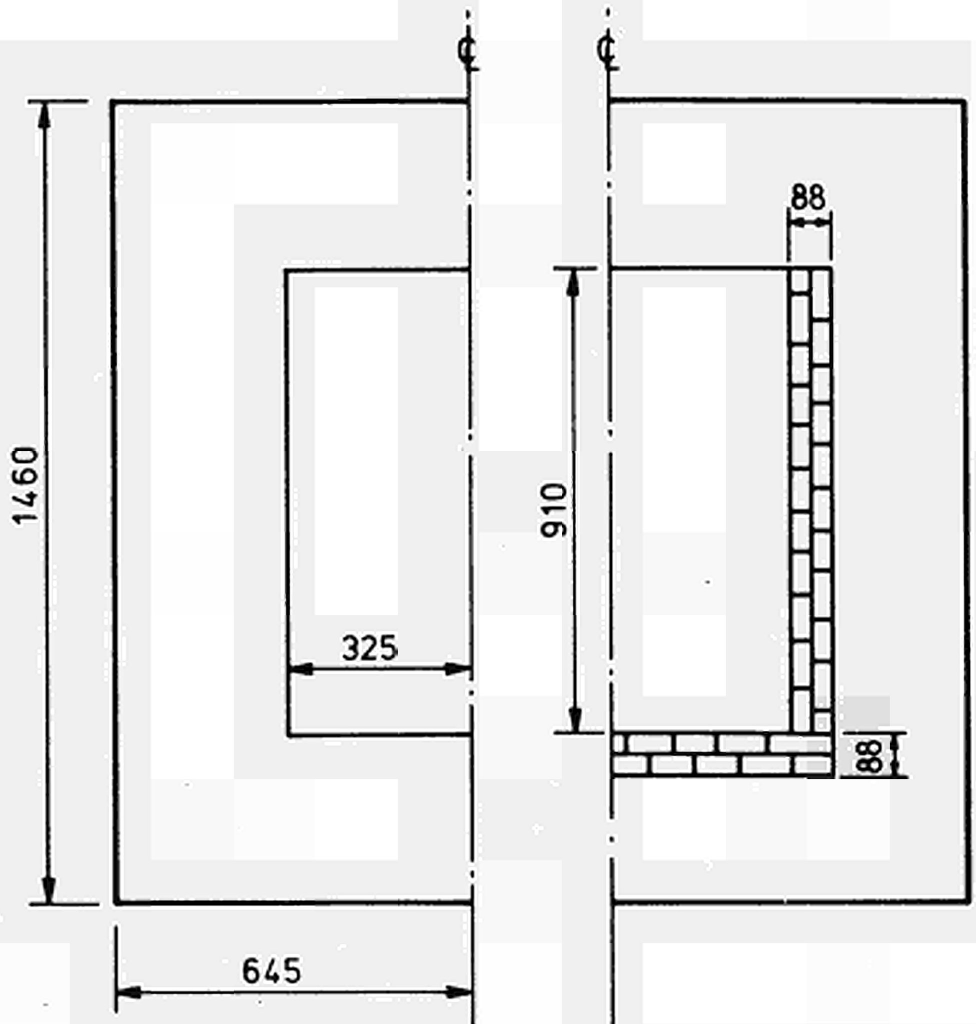
Procurement of materials for construction of the models has commenced and proposals for testing them have been prepared.

4. Conclusions (B.6.)

From the work so far completed it may be concluded that it should be possible to construct, test and analyse models which should provide meaningful and comparable results.

References

/1/ PATON, A.A., BENWELL, P., IRWIN, T.F. and HUNTER, I. Commission of the European Communities Report EUR 9399 EN (1984).



(a)
Half section through model without planes of weakness

(b)
Half section through model with planes of weakness

Design pressure: 1·9 MN/m^2 (275 psi)

Factor of safety : 1·7

FIGURE 1 : BASIC DESIGN AND DIMENSIONAL PARAMETERS

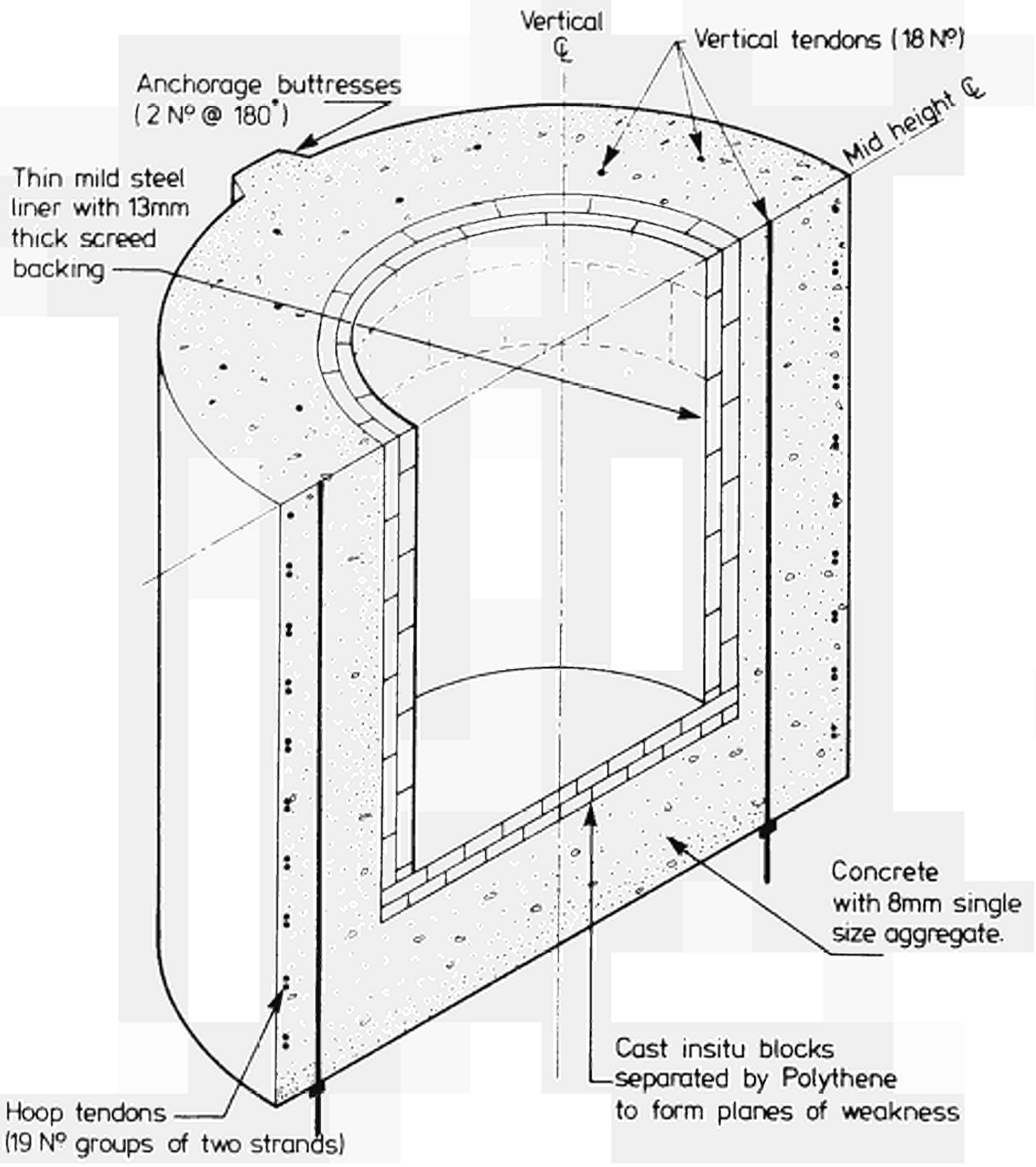


FIGURE 2 : QUARTER CUT AWAY VIEW OF MODEL
WITH PLANES OF WEAKNESS

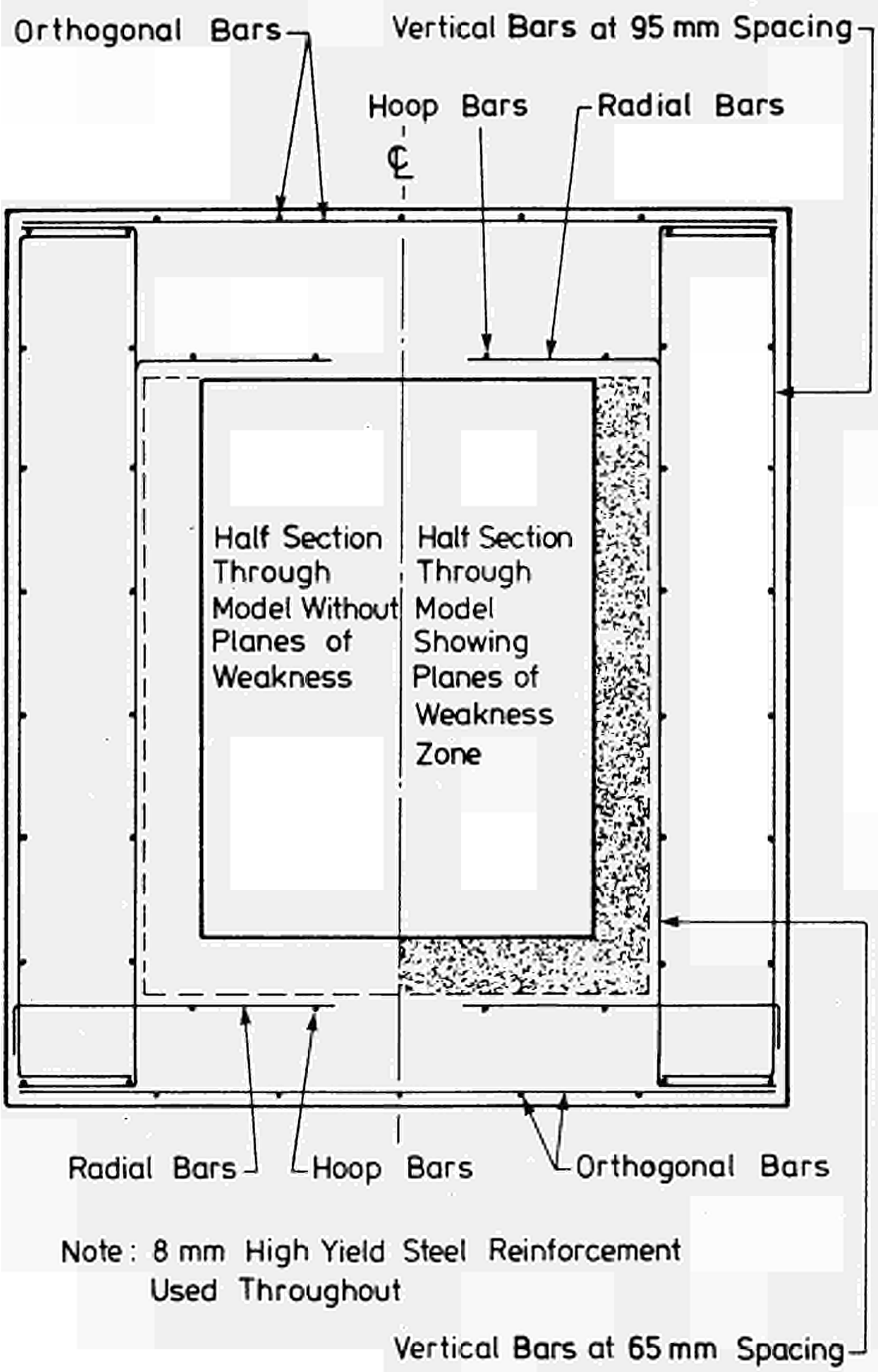


FIGURE 3 : REINFORCEMENT LAYOUT

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

With reference to points B.1. and B.2. of work programme, the activities completed in 1987 are:

- characterisation of process parameters through laboratory tests;
- detailed design of the impregnation prototype unit;
- construction of the prototype;
- preliminary tests of the prototype.

Operating tests were sucessfully completed and the unit is ready to start research activities according to points B.3. and B.4., which are scheduled to take place in 1988.

Progress and Results

1. Optimisation of the impregnation parameters (B.1.)

The input data for developing the design of the prototype were obtained through preliminary laboratory tests aiming at optimising the process parameters in view of obtaining the desired depth and degree of impregnation.

Several series of tests on 4x4x16 cm concrete samples were performed in a special working chamber at various temperatures and pressures.

The optimal conditions for obtaining an efficient application to the concrete are defined as follows:

- concrete dehydration at 140-160°C for 6-8 hours;
- concrete degassing for 4-6 hours at a residual pressure of 50 kPa;
- monomer injection at an overpressure of about 20 kPa;
- thermopolymerisation of monomer at 80°C, using water or nitrogen in overpressure of about 20 kPa.

2. Manufacture and implementation of the prototype device (B.1.)

The impregnation prototype device was manufactured and then mounted on a lift truck built expressly (Figures 1 and 2). The main operating features of this unit are:

- impregnation surface for each cycle (1.8m x 0.75m)	1.35 m^2
- temperature setting range during dehydration	5 - 300°C
- temperature gradient range	5 - 60°C/h
- max. vacuum attainable in the impregnation chamber during degassing	10 kPa
- max. overpressure during monomer injection	25 kPa
- max. power demand during dehydration	about 7 kW
- power demand during thermopolymerisation	about 3 kW
- attraction strength developed by the sucking device with vacuum corresponding to 50 kPa	3000 daN
- thrust developed on the impregnation chamber by a monomer injecting pressure of 20 kPa	2700 daN

3. Testing of the prototype (B.2.)

Preliminary tests were carried out according to work schedule and the unit satifactorily performed all operations, attaining, or even exceeding, the targets established in the project.

The unit was then shipped to the ITALCEMENTI laboratories to start the research work, which is scheduled to take the whole year 1988.

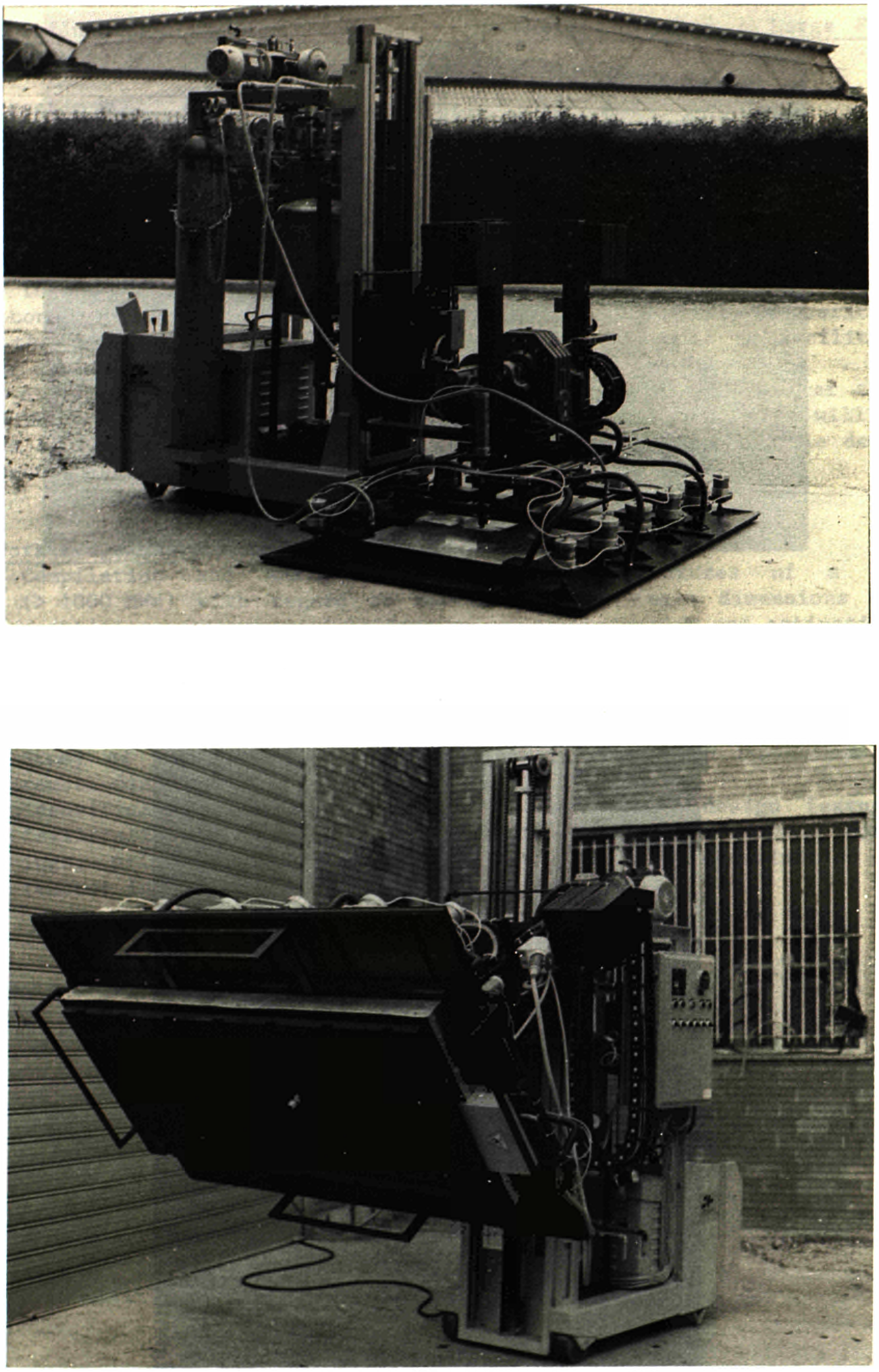


Figure 1 : The impregnation prototype device

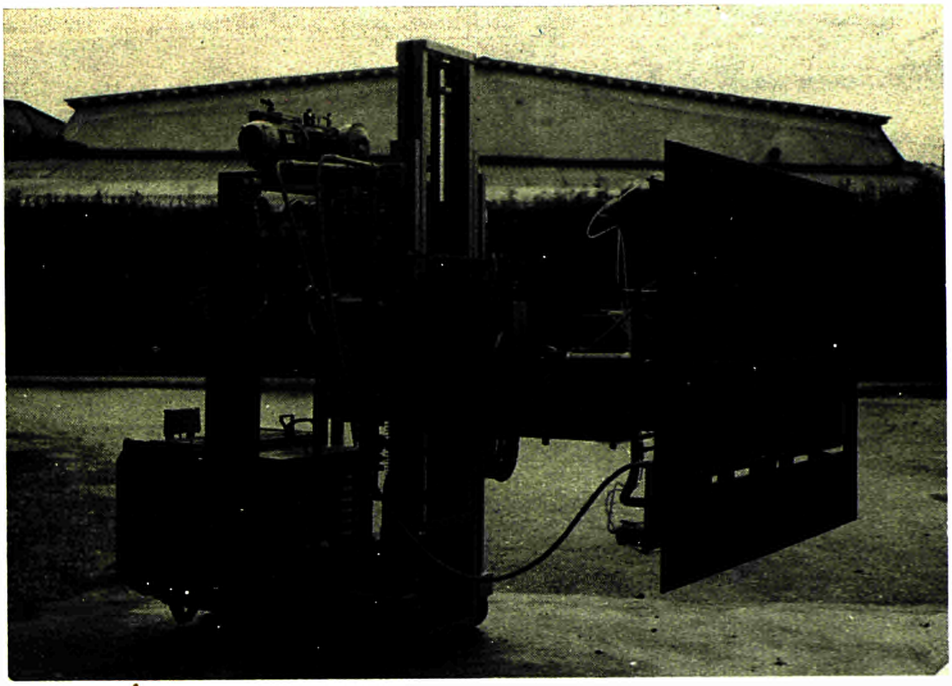


Figure 2 : The impregnation prototype device

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 - June 1988
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

Main features of major components were collected for the two Fast Breeder Reactors: Superphenix 1 (SPX1) built in Creys-Malville and Superphenix 2 (SPX2), a 1500 MWe French project. This second project, called European Fast Reactor (EFR) is the most representative of future European projects, and is based on a pool breeder Nuclear Steam Supply System (NSSS) (Figure 1). Peculiarity of this design is the containment of all active products in a vessel, characterised by structural simplicity. The calculation of activation and contamination of these components was performed, for a low decay after 30 years in operation and final shutdown. It shows that activation decreases with a factor > 10^8 between diagrid (the most active) and the primary vessel. It was then possible to give a first approach on decommissioning SPX2. Three solutions were given with: no Co-60 decrease, 35 years of decrease (factor 100) and 100 to 150 years of decrease (factor > 10^5). The first solution is constraining and the operations of dismantling are long and expensive.

Progress and Results

1. Design features of a large Fast Breeder Reactor (FBR) (B.1.)

The general structure of a fast breeder reactor always includes 3 main systems: a primary sodium system removing the heat produced in the core (this sodium is activated by neutrons), a secondary sodium is non-radioactive, and a third steam-water system. Two design solutions are then possible: either the primary system is entirely contained in a large pool-type vessel, or the vessel only contains the core (loop type). Most large FBR are designed on pool type, and all European studies used this concept after the SPX1 construction (SPX2 in France, CDFR in the United Kingdom, SNR2 in Germany). So, the next EFR will be a pool-type NSSS, and advanced SPX2 will give good approach works.

All the active sodium is enclosed in double containment, which considerably limits activated or contaminated components. Figure 2 shows the SPX1 reactor block. Table I gives, for each component, weight and dimensional features and localisation of activation and contamination.

2. Determination of important features for decommissioning (B.2.)

Only components which are in the main vessel will be activated or contaminated. Also, a calculation of the activation and contamination by corrosion and fission products was achieved. The main parameters are:

- reactor 30 years in operation,
- low decrease after shutdown (calculation if T1/2 > 1 month),
- cobalt activation and some major nuclides as Ag (not for impurities),
- surface contamination on the slab and the internals,
- 0.2% Co in stainless steel and 1% Co in Inconel 718 (specified).

The first results (Table II) show for each component that:

- activation is more important in the sodium (Na-22) than fission products (Cs-137),
- activation is not very important for removal components, except for those on the core cover plug, which are activated in the lower part,
- contamination by corrosion products is important for the intermediate heat exchanger (IHX) and pumps,
- activation is very important for structures near the core: diagrid, strongback, inner vessel,
- important activation decrease from diagrid (near the core) to the main vessel (behind pumps and IHX).

3. Various stages of decommissioning (B.3.)

Various decommissioning scenarios (Table III) were suggested according to levels proposed by IAEA.

Storage periods are determined as follows:

- total decrease of Co-60 nuclide: 100 to 150 years: factor 10^5 to 10^6;
- partial decrease of Co-60 nuclide: 35 years: factor 100;
- reactor building integrity: normally, structure performances will be maintained 30 years after final shutdown;
- long-term integrity is about 100 years.

Stage 1: after final shutdown, only roof components (pump, IHX, mechanism...) will be removed. Internal, secondary loops and all components which have been in sodium will be dismantled after 30 to 40 years of storage (remote dismantling).

Stage 2: dismantling of nuclear structures except internal structures in the reactor block. Buildings (except reactor building) are dismantled. Reactor block and structures (vessels, diagrid...) will be dismounted after 150 years (total Co-60 decrease): deferred dismantling.

Calculation of long-period nuclides will be made to study this stage.

Stage 3: after shutdown, all structures will be dismantled. Long and expensive operations are the cutting of activated internal structures under biological shield (water).

Table I: Design features SPX2 components

Component	Ø (m)	H (m)	Weight (t)	Activation	Contamination	Nb
Pump	2.2	15	100	low	yes	4
Intermediate HX	3	17	70	low	yes	8
Rod mechanism	0.3	10	1.5	yes	yes	33
Load machine	2.2	16	20	yes	yes	1
Burst pin location	0.5	8	1	low	yes	6
Purification unit	1.5	10	20	low	yes	1
Emergency HX	1.5	11	10	low	yes	4
Roof slab	20	2.5	1100	low	low	
Plugs	10	3.3	800	low	low	
Core cover plug	4.5	9	150	yes	yes	
Main vessel	20	16	360	low	yes	
Inner vessel	19	10	200	yes	yes	
Strongback	13	3	200	yes	yes	
Diagrid	7	1	150	yes	yes	
Core catcher	12.5		17	yes	yes	
Others			30	/	/	

Table II: Calculation of activation and dose rate

COMPONENTS	TYPE	NUCLIDE	ACTIVITY (Bq)	DOSE RATE (Sv/h)
PRIMARY SODIUM	ACT	Na22	1.6 E+6/g	
	PF	Cs137	2.5 E+4/g	
PRIMARY PUMP	PC	Mn54	1.8E+12	0.04 at 0.5m from hydraulic channel(impeller, discharge bowl)
IHX	PC	Mn54	3.7E+12	0.06 at 0.3m from tube bundle
EMERGENCY HX	PC	Mn54	4 E+11	2 E-3 at 0.1m from tube bundle
PURIFICATION UNIT	PC	Mn54	4.4E+11	0.2 at 0.1m
LOAD MACHINES	ACT	Co60	1E+6 /cm3	0.04 in contact with the lower part
ROD MECANISMS 1	ACT	Co60	3E+8 /cm3	30 in contact with the holder
" " 2	ACT	Ag 110m	1.5E+9 /cm3	90 in contact with electromagnetic
VISUS	ACT	Co60	3.5E+8 /cm3	20 in contact with the lower part
BURST PIN LOCATION	ACT	Co60	3E+5 /cm3	4 E-3 in contact with the lower part
MAIN VESSEL	ACT	Co60	8E+2 /cm3	2 E-5 in contact with bottom
	PF	Cs137	3E+3/cm2	
INNER VESSEL	ACT	Co60	3E+4 /cm3	0.18 in contact with inner vessel
ROOF SLAB	PF	Cs137	3E+3/cm2	
ROTATING PLUGS	PF	Cs137	3E+3/cm2	
CORE COVER PLUG	ACT	Co60	2.5E+7 /cm3	0.75 in contact with lower plate
DIAGRID	ACT	Co60	6E+10 /cm3	1000 in contact with upper plate
STRONGBACK	ACT	Co60	1.4E+8 /cm3	3 in contact with lower plate
CORE CATCHER	ACT	Co60	1.63E+4 /cm3	0.5E-3 in contact

ACT: activation
PF: fission products
PC: corrosion products

1 rem/h : 10^{-2} Sv/h
1 Ci : 3.7 10^{10} Bq

Table III: Stages of decommissioning a FBR

COMPONENT	FINAL SHUT-DOWN	LEVEL 1	AFTER 35 a	LEVEL 2	AFTER 150 a	LEVEL 3
PRIMARY NA	*					
MAIN VESSEL			*		*	*
INNER VESSEL			*		*	*
DIAGRID			*		*	*
STRONG BACK			*		*	*
CORE CATCHER			*		*	*
ROOF SLAB			*		*	*
LARGE ROTATING PLUG			*		*	*
SMALL ROTATING PLUG			*		*	*
CORE COVER PLUG			*		*	*
PRIMARY PUMPS	(*)	(*)				
IHX	(*)	(*)				
EMERGENCY HX	(*)	(*)				
LOAD MACHINES	(*)	(*)				
SMALL COMPONENTS	(*)	(*)				
PURIFICATION UNIT	(*)	(*)				
SECONDARY SODIUM	*					
SECONDARY LOOP			*	*		*
SECONDARY PUMP		*		*		*
STEAM GENERATOR			*	*		*
TANKS			*	*		*
NA/AIR HX			*	*		*
NA/AIR LOOPS			*	*		*
						*
CELLS			*	*		*
LIQUID WASTE BUILDING			*	*		*
TURBINE		*		*		*
REACTOR BUILDING			*		*	*
OTHERS BUILDINGS		*		*		*

* : component dismantled in the stage

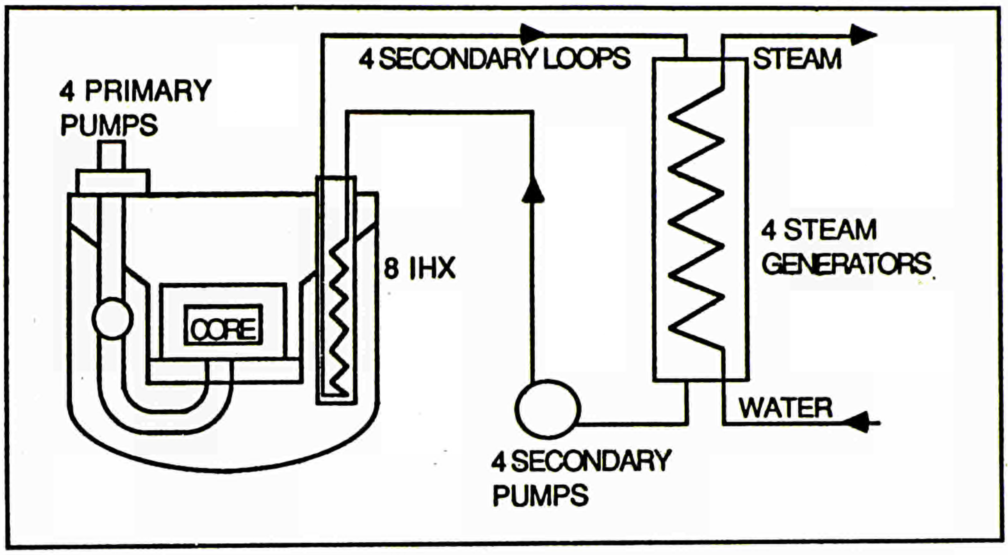


Figure 1: Pool type fast breeder reactor

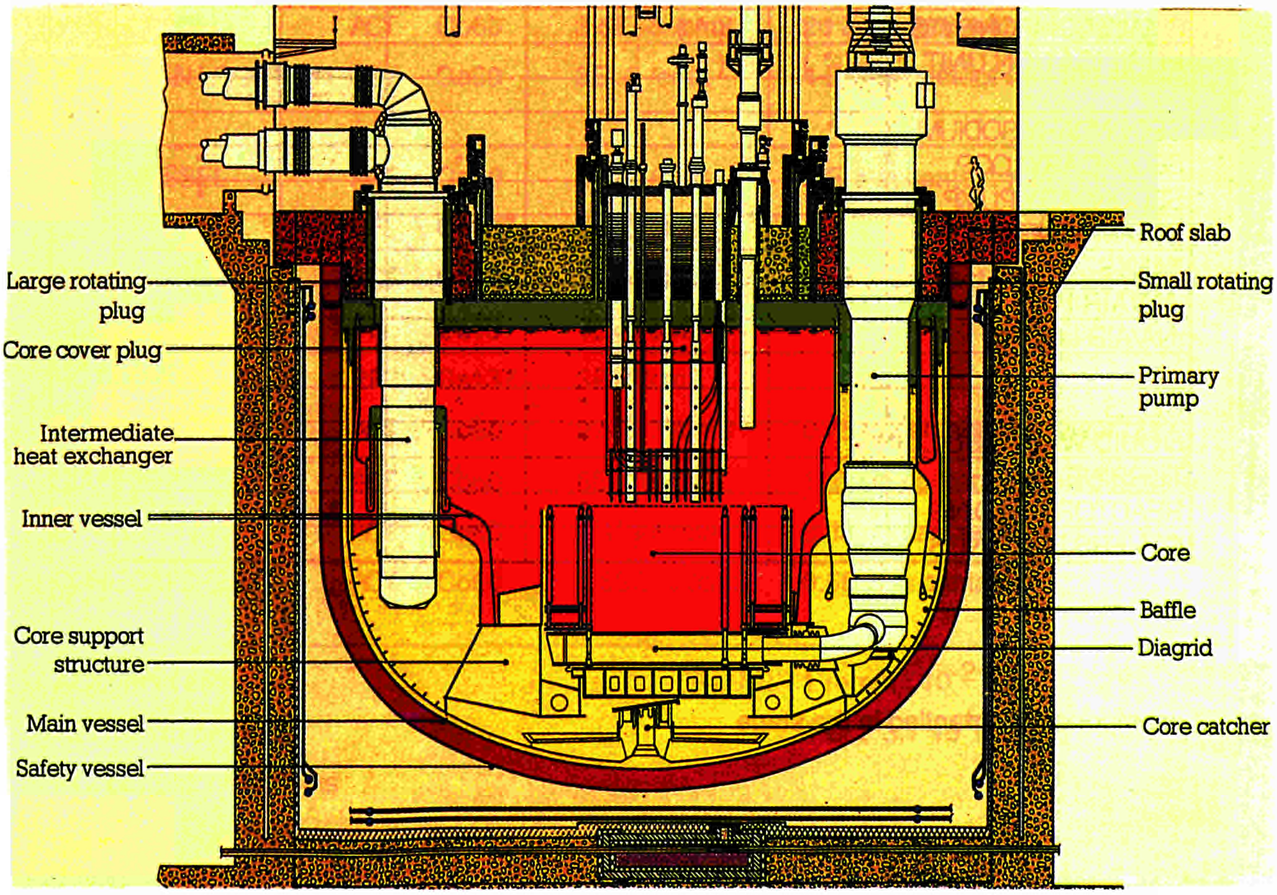


Figure 2: Fast breeder reactor block

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.

Twelve research contracts relating to Section C were being executed in 1987, including one contract whose execution has been completed in 1987.

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 - December 1988
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

Using a specially designed ultrasonic machine two in scale testings were performed in the hot chemical lab. of Garigliano BWR station.

The experiments were carried out on assemblies of about 20 pin tubes 40 cm long each. The procedure consists of a US step in water for half an hour followed by some steps with aggressive chemicals, for about 2 h.

The in scale testings showed that the objective of decontaminating the tube specimens to less than 1 Bq/g was clearly achievable.

To reduce the process time the best test temperature is 60°C for both water and aggressive chemicals. In the heating solution of the US machine the addition of a wetting detergent is advisable.

Progress and Results

1. Laboratory investigations on the ultrasonic and chemical procedure (B.3)

1.1 Ultrasonic machine for in scale testing

From the results of the preliminary in scale testing, it was seen that one of the reasons why total decontamination could not be obtained was the feature design of the ultrasonic system.

In order to design the later experiments better, a specific facility was designed and acquired from "Pavesi s.n.c." It was installed in the hot chemical laboratory of Garigliano BWR station in June, 1987.

A schematic view of the machine is presented in Fig. 1. The test tank is in stainless steel about 3 mm thick, coated with a thin teflon layer (0.3 mm). The tube specimen support is plastic (PVC); the top and bottom grids have 25 holes to allocate the 40 cm long tube specimens; the specimens, which stay on the bottom grid, are about 5 cm up from the bottom of the tank. To improve a possible recirculation of the test solution several passage holes were made in the lattice of the grids.

Views of the machine are given in Fig. 2.

1.2 2nd In Scale Testing

The main purpose of the second in scale testing was to discover whether the ultrasounds with aggressive chemical decontamination can meet the unrestricted release limit of materials of 1 Bq/g, testing 40 cm long tubes.

As a conclusion of the first in scale testing, the following procedure schema was selected for the second test :

- apply ultrasounds in demineralized water;
- apply ultrasounds in aggressive chemical solution.

The test conditions are summarized as follows :

- test materials : 21 tube specimens 40 cm long each for a total of 8.4 m, 76.4 dm of exposed surface and about 5 kg of weight;
- test temperature : 60°C;
- ultrasounds : 21 kHz/3.25 kW;

- test time : 30 min with demineralized water;
  2 h with aggressive chemicals (3% HF + 5% HNO_3);
- solution in the heating tank : with wetting detergent (3% Deoxil-88).

The test results are summarized in Table I. They show that the objective of cleaning to less than 1 Bq/g is clearly achievable.

Nevertheless different tubes have different behaviours in terms of the time required for a total decontamination.

This time does not appear correlatable with any characteristics of the test procedure and the tube sampling, it is therefore an empirical variable; the maximum test time was 2 h for 1 tube (5% total) but this value might change.

It was noticed that the tube specimens which are clean and shiny in appearance, proved to be decontaminated (clean also in terms of radioactivity). This is very important because it allows us to perform a quick screening of the tubes at the end of the test.

The results of the final 60-Co radiometric measurements on tube specimens in different ranges of radioactivity, are presented in Fig. 3

1.3 <u>3rd In Scale Testing</u>

One more in scale testing was performed both to confirm the results of the second in scale testing and to optimize the decontamination process, mainly with regard to the test temperature.

Furthermore, remembering that in the second in scale testing some tube specimens were decontaminated to much less than 1 Bq/g a target was set of decontaminating all the tube specimens to less than 0.25 Bq/g.

Before testing the following topics were defined:

- to perform the decontamination by a step in demineralized water followed by a step with aggressive chemical;
- the decontamination with US in demineralized water was scheduled in a single step half an hour long, at 60°C;
- for the decontamination in aggressive chemicals the 3% HF + 5% HNO_3 solution was selected;
- the decontamination with US in aggressive chemicals was scheduled with four steps, half an hour long each, at room temperature (without heating) followed if necessary by more steps at 60°C.

The test results are summarized in Table II and in Fig. 4. They allow the following considerations:

- all the tube specimens were found to be decontaminated to less than the prescribed limit of 0.25 Bq/g; nevertheless only 6 tube specimens (30% of total) were decontaminated to less than 0.25 Bq/g working at room temperature during the steps with aggressive chemicals; furthermore 1 of the 6 tube specimens needed 2 h test time with US in aggressive chemicals while the other 5 tube specimens required only 30 minutes;
- the final average value of 60-Co residual radioactivity was around 0.07 Bq/g and only one tube specimens was still contaminated to more than 0.1 Bq/g;
- the US in aggressive chemicals must be applied at temperatures around 60°C to be sure of meeting the decontamination limits prescribed;

- some tube specimens, always different, behave differently in terms of time required for total decontamination. This is true for about 25% of tube specimens and confirms the results obtained in previous in scale tests. This behaviour is not yet understood.

2. Radiometric measurements (B.4.)

Some measurements of low beta and gamma emitters such as 55-Fe, 59-Ni and 63-Ni were performed in order to see their relative presence in contamination layers. The measurements were performed by dissolving the oxide layer (with some µm of the base metal) of 3 cm long tube specimens from the pin tubes.

The results of the measurements on non-decontaminated specimens allow the following conclusions:

- the inside surface: there is always less 55-Fe than 60-Co (ranging from 40 to 60% of 60-Co); there is always less 63-Ni than 55-Fe (ranging from 40 to 80% of 55-Fe) nevertheless it appears that the lower the 55-Fe the higher the 63-Ni; the 59-Ni is always lower than the detectable limit of 0.1 Bq/cm^2;
- on the outside surface a similar conclusion can be drawn, although the absolute values are very low and they do not allow precise evaluations.

The results of the measurements on decontaminated specimens, from the 2nd in scale testing, allow the following remarks to be made:

- the outside surface appears to be much cleaner than the inside surface;
- the 60-Co data of these measurements appear to be in good agreement with the 60-Co data from the radiometric measurement system of Garigliano Station; furthermore the average data obtained by the radiometric measurement system appear much more conservative than the data obtained by the specific measurements;
- it appears possible to correlate the 55-Fe and 63-Ni data indicating that the sum of 55-Fe and 63-Ni is quite similar to the value of 60-Co;
- looking at the data of some tube specimens it seams that the centre zone remains more contaminated than the ends.

Table I - Summary of 60-Co radioactivity results in the 2nd in scale testing.

Decontamination step	No. of tube specimens	60-Co radioactivity (Bq/g) Average	Max	Min	Decontamination factor (average)	No. of tubes fully decontaminated
Initial	21	35.7	60.3	23.5	-	-
US/Water/60°C						
1st step: 3 min.	21	3.0	3.9	1.3	11.9	-
2nd step: 10 min.	21	1.9	2.2	1.1	18.8	-
3rd step: 17 min.	21	1.54	3.3	0.8	23.2	-
US/HF-HNO_3/60°C						
1sh step: 30 min.	21	0.6	1.9	0.02	59.5	12
2nd step: 30 min.	9	-	1.7	0.02	-	8
3rd step: 60 min.	1	0.34	1.0	0.02	105.0	1

Table II - Summary of 60-Co radioactivity results in the 3rd in scale testing.

Decontamination step	No. of tube specimens	60-Co radioactivity (Bq/g) Average	Max	Min	Decontamination factor (average)	No. of tubes fully decontaminated
Initial	20	21.2	43.7	19.6	-	-
US/water/60°C (30 minutes)	20	1.24	1.98	0.77	17	-
US/HF-HNO_3/room temp.						
1st step: 30 min.	20	0.89	1.58	0.05	24	5
2nd step: 30 min.	15	0.98	1.34	0.58	22	-
3rd step: 60 min.	15	0.74	1.20	0.20	29	-
US/HF-HNO_3/60°C						
4th step: 30 min.	15	0.34	0.99	0.04	62	9
5th step: 30 min.	6	0.16	0.51	0.04	132	4
6th step: 30 min.	2	0.07	0.12	0.04	303	2

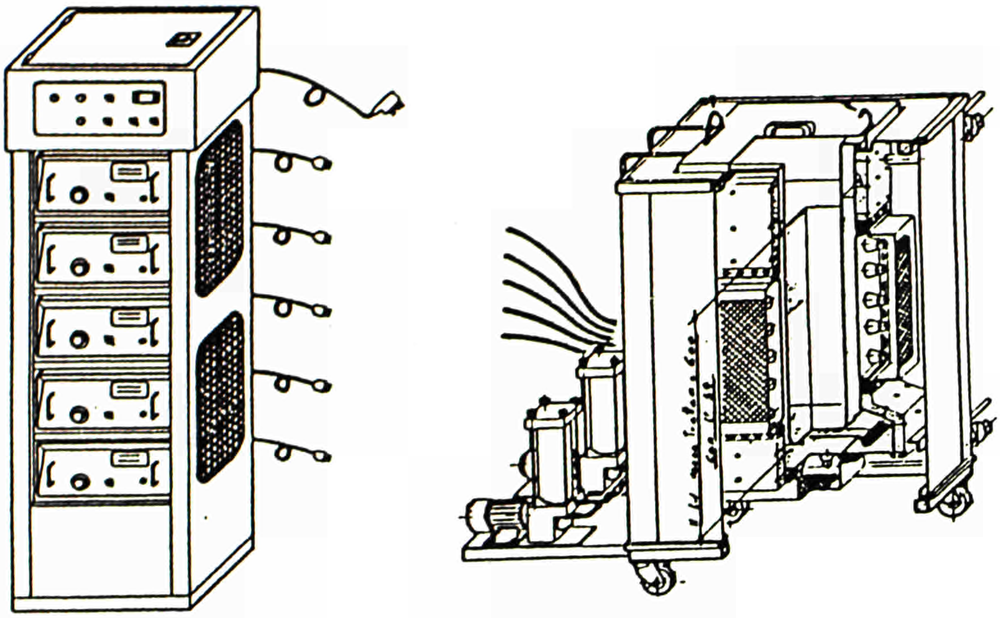


FIGURE 1 - Schema of the ultrasonic wash machine used for in scale testing.

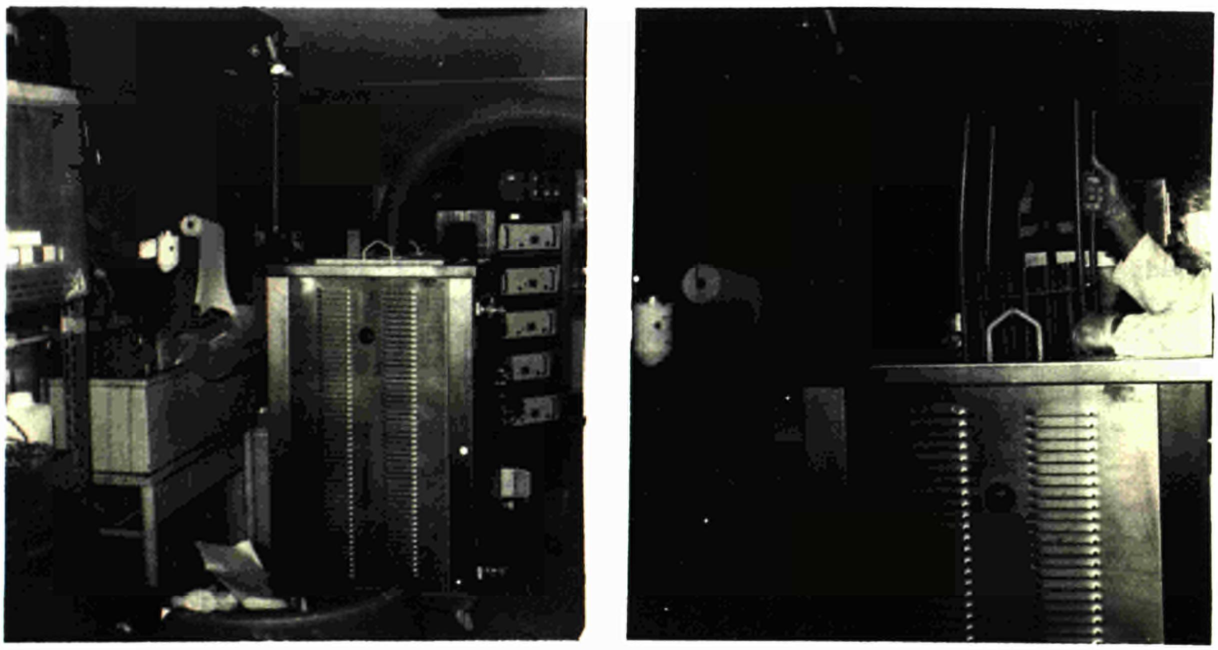


FIGURE 2 - Views of the ultrasonic wash machine used for in scale testing:
- left: general view;
- right: loading of the test specimens.

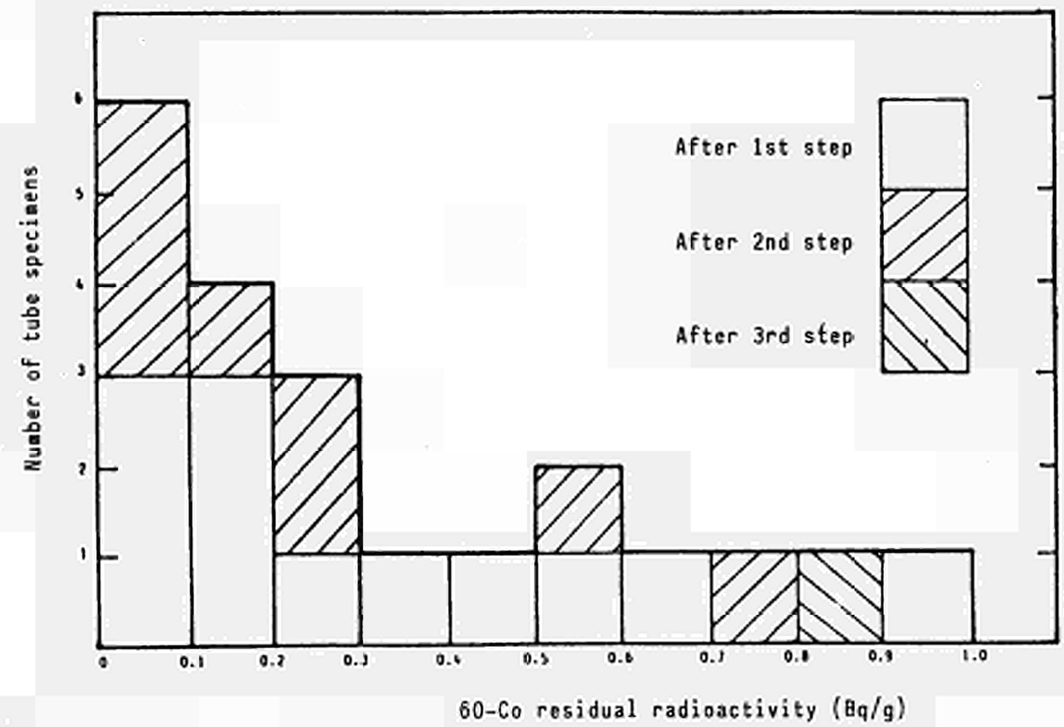


FIGURE 3 - Residual radioactivity of tube specimens at the end of 2nd in scale testing.

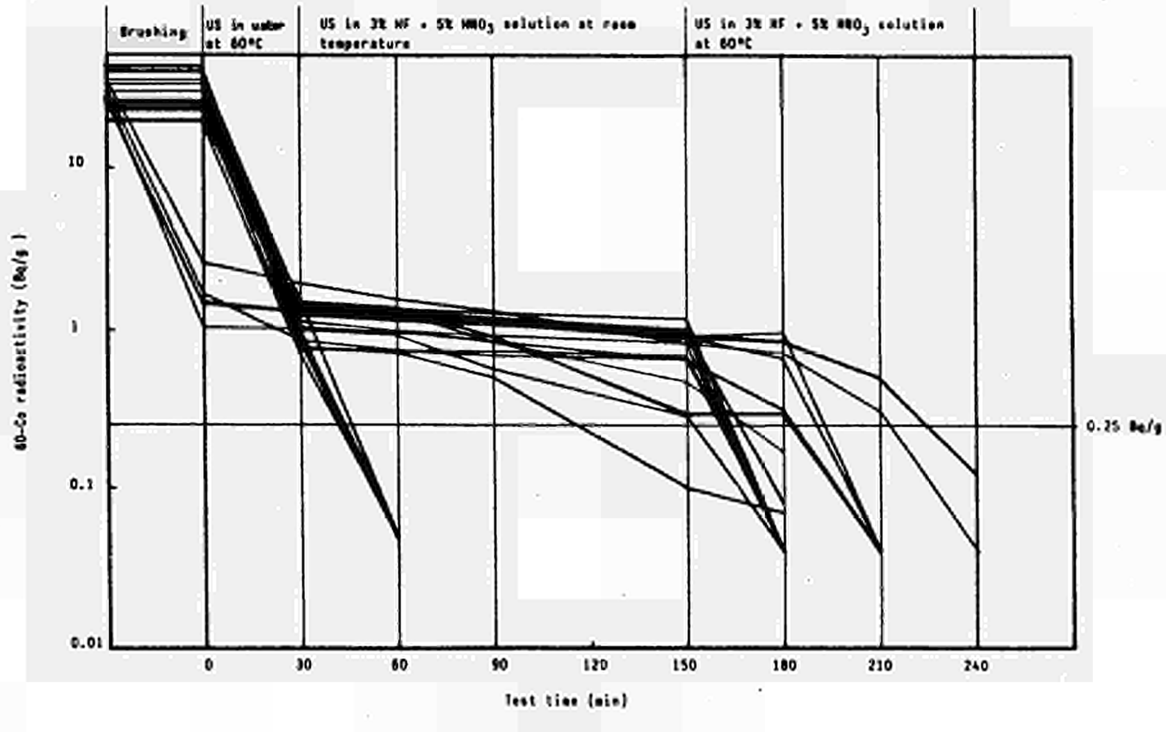


FIGURE 4 - Radioactivity versus test time in the 3rd in scale testing.

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

The last glove box has been dismantled with the same procedure as before. Similar results have been obtained as presented in the APR-86 under reference EUR 11112, so no further presentation of results will be given.

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 - December 1988
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 current year 1987 the following applications were carried out by this program:
- removal and dismantling of several pipes and armatures
- dismantling of components of the low-pressure turbine
- dismantling and decontamination of preheater bundles
- decontamination of several pipes and armatures
- decontamination of turbine components

Furthermore detailed studies on long-term surveying and optimization of electrolytes were done.

Progress and results

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

In the reporting period some already recommended cutting techniques were applicated.
Four bleeder steam pipes situated between the high-pressure turbine and the preheaters no. 1 - 4 with diameters from 350 until 1000 mm and a total mass of 18 tons, as well as 6 tons of several pipes and armatures of the drainage system from the high-pressure turbine with diameters from 65 until 150 mm were removed and cut.

Torch cutting was selected for the removal of pipes and armatures in situ because of higher cutting velocity and easier handling compared with mechanical machinery. During the thermal cutting of the bleeder steam pipes with a specific contamination between 25 and 120 Bq/cm^2 an airborne aerosol activity of about 5 to 8 Bq/m^3 was detected. The drainage pipes with an initial contamination of 15 Bq/cm^2 created an aerosol concentration of 0,5 - 2 Bq/m^3. The cutting work in situ was executed in 830 manhours and entailed to a total job dose of 2100 μSv. The following mechanical cutting of the pipes to 1 meter pieces for pool decontamination was mainly done by hack saws. These cutting works required 450 manhours and produced a total dose of 600 μSv. The effort of personnel and time for the removal in situ and the cutting works afterwards are given in the tables I and II. According to previous experiences, the removal and dismantling of pipes and armatures creates relatively high mass specific costs. It proved effective to remove pipes in large pieces and to cut them for pool decontamination by applying cutting techniques which are favourable with regard to radiological aspects.

At the end of 1987 also 4 heavy support plates of the low-pressure turbine with a total mass of 42 tons were dismantled. These support constructions consisted of thick plates. Formerly they stabilized the upper and the lower shell of the low-pressure turbine inside the main condenser. Because of the type of construction torch cutting with propane gas was recommended. The whole cutting work of about 42 t was done with 3 workers in 850 manhours. Principally all thermal cutting works were executed by operation of mobile suction filter systems and under protection masks. The total effort including costs is shown in table III.

2. Selection and large scale application of techniques for cutting and decontamination of components from the turbine house (B.1., B.2.)

Two service companies GNS and KAH were contracted for the treatment of 8 preheater bundles formerly installed in five feedwater preheaters.
The work at KRB A included the removal and the dismantling of the bundles for transport. Because the bundles consisted of austenitic steel or a copper-nickel-iron alloy the segmentation was executed by a carbon arc supported by compressed air. Ten cuts through the preheater bundles with diameters of about 2 m took 220 manhours. Five bundles (67 tons) will be decontaminated by GNS and will be molten down for a further reuse in the nuclear industry. Until now about 20 tons of the material have been successfully decontaminated with a mixture of nitric acid and hydrofluoric acid. The residual activity was about 0,1 Bq/g. The external dismantling was performed within 370 manhours. Decontamination and the conditioning of acids for the treatment of 20 tons of bundle material required 180 manhours.
Three preheater bundles (25 tons) should be treated by the working group TN/KAH, Heidelberg. It was tried to decontaminate the material by sandblasting for unrestricted release. Because this decontamination method failed, the working group decided to shear off, compact and to condition the material for final storage. A packing density of 3 t/m^3 could be realized.
Comparing both concepts, there is an economical advantage for the recycling method.

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

Based on the experiences on different decontamination procedures about 35 tons of several pipes and armatures of different systems have been selected to prove the economy and efficiency of the recommended methods.
The flow chart in Fig. 1 represents the procedure for the decontamination of primary medium-attacked components. In the first decontamination step mainly pickling and electropolishing was applied. The decontamination on only air-contaminated components was executed by the following procedures:

- brushing with pickling agents
- wetting with detergent solutions
- brushing with phosphoric acid
- high-pressure water cleaning (70 bar)

The decontamination of different contaminated pipes and armatures (about 35 t) could be done in 230 manhours. Acid regeneration, transports and surveying measurements amounted to a total treatment time of 700 manhours. Table IV informs about the total treatment costs for pipes and armatures, not including the dismantling effort.

The total personal dose for the decontamination was 930 µSv. The applied decontamination procedures were found to be economical. The reason was a particularly low production of secondary waste.

A further demonstration of large scale decontamination was carried out with components of the turbine. The channel walls and the water separators with a total mass of 120 tons have been decontaminated prior to melting the material in a foundry in order to produce cast iron casks. The purpose of the decontamination in dip tanks with phosphoric acid was to remove slightly adhering contaminated layers and to reduce the fixed contamination to approx. 1 Bq/g for melting. The decontamination of about 120 t was executed within 1370 manhours and by a total job dose of 2800 µSv. For additional works 950 man-hours were necessary.
Table V gives a review about the effort and costs for this decommissioning campaign. Wiping tests showed that these components were below 0.37 Bq/cm^2 after decontamination. This material with a maximum residual activity of 2 Bq/g will be molten down in an authorized foundry.

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

The electrolyte used for decontamination of contaminated plant parts is regenerated in a closed cycle since March 1985 in order to reduce iron content including radioactivity and to minimize the secondary waste. Consequently, extensive experiences on decontamination and regeneration are available whose evaluation led to the application of optimized operating conditions as well as to a best possible tuning between the requirements of decontamination and regeneration of the electrolyte.

The surveying of the electrolyte served to establish the interdependence between the method of decontamination and the chemical composition of the electrolyte. The investigation of the influence of continuous regeneration on the chemical composition was of special interest. It could be shown that an enrichment of cesium 137 that could have been expected with regard to the chemical properties of this nuclide could not be ascertained by measurement of samples dating from March 1985 to November 1986. Also an enrichment of chromium(III) that, like cesium, does not give a slightly soluble oxalate could not be detected. Consequently, neither an increase of the activity concentration nor of the concentration of metal ions seems to occur during the permanent use of the continuously regenerated electrolyte.

The optimization of the electrolytes had the aim of high decontamination efficiency, best possible capacity for dissolved iron as well as a good efficiency of regeneration. An important condition for achieving the aims is a high portion of iron(II) that can be attained by choosing the proper decontamination method: while by electrochemical decontamination (electropolishing) a relatively high amount of iron(III) arises, a high percentage of iron(II) gets into the solution by chemical decontamination (pickling). The influence of the iron(II)/iron(III)-ratio in the scope of low iron concentrations is small, indeed; in the case of high concentrations of total iron and high percentages of iron(III), however, a deposition of iron(III)-phosphate on the surface and, by this, a recontamination may occur, so that in the case of higher iron concentrations pickling should be preferred to electropolishing.

Because of the higher solubility of the bivalent iron in phosphoric acid in contrast to iron(III), decontamination can then be continued until reaching higher iron concentrations. This advantage of a longer operation time of the electrolyte is accompanied by a better efficiency of regeneration: as only iron(II) forms a slightly soluble oxalate that is precipitated during regeneration, the iron content can be decreased the more, the higher the portion of iron(II) referring to the total iron content is.

The concentration of phosphoric acid during the precipitation of iron oxalate plays a roll, too. In a diluted phosphoric acid the iron content can be reduced enormously; then, however, a longer evaporation time is necessary in order to restore the initial concentration of phosphoric acid. With regard to the requirements of efficiency of decontamination, operation time, and regeneration, the initial concentration should be between 20 and 40 per cent.

Table I

Effort and costs for the removal of about 24 t of several pipes and armatures

mass specific effort for removal in situ	: 34 manhours/ton
costs for consumable material	: 190 DM/ton
waste costs for conditioning and final storage	: 250 DM/ton
shared costs for equipment	: 100 DM/ton*

Table II

Effort and costs for the cutting of about 18 t of several pipes and armatures for decontamination

mass specific effort for cutting	: 25 manhours/ton
costs for consumable material	: 190 DM/ton
waste costs for conditioning and final storage	: 170 DM/ton
shared costs for equipment	: 100 DM/ton*

Table III

Effort and costs for the cutting of about 42 t of components from the low-pressure turbine

mass specific effort for cutting	: 20 manhours/ton
costs for consumable material	: 260 DM/ton
waste costs for conditioning and final storage	: 240 DM/ton
shared costs for equipment	: 100 DM/ton*

* referring to the total inventory of the turbine house (2500 t)

Table IV

Effort and costs for the decontamination of about 35 t of several pipes and armatures

mass specific decontamination effort	:	20 manhours/ton
waste costs for conditioning and final storage	:	340 DM/ton
shared costs for equipment	:	150 DM/ton*

Table V

Effort and costs for the decontamination of about 120 t of channel walls and water separators

mass specific decontamination effort	:	24 manhours/ton
waste costs for conditioning and final storage	:	340 DM/ton
shared costs for equipment	:	150 DM/ton*

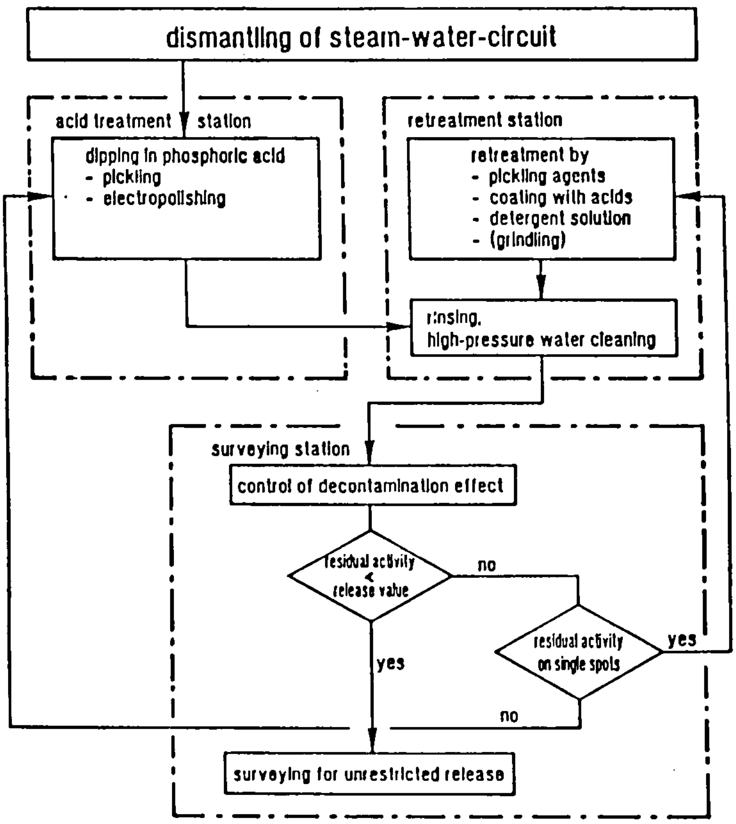


Fig. 1 Flow chart for decontamination of components of the steam-water-circuit

* referring to the total inventory of the turbine house (2500 t)

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 - December 1988
Project Leader: J.R. Wakefield

A. Objectives and Scope

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

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

B. Work Programme

B.1. 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 decontamination plant.
B.3. Decontamination of the heat exchangers and conditioning of the secondary waste.
B.4. Dismantling of the boilers and conditioning of the scrap for final disposal.

C. Progress of Work and Obtained Results

Summary

Design studies and laboratory experiments have established that the most effective methods for decommissioning WAGR Heat Exchangers (HEs) were likely to be (a) a fill and drain process using dilute hydrochloric acid (HCl) and (b) refluxing with an azeotropic mixture of HCl. The problems of disposal of HCl-based effluent via BNFL, and particularly the large quantities associated with method (a) have proved impossible to resolve within the timescale of the 5-year programme. Process control studies for the refluxing method have, however, continued at Winfrith and are reported here.

Proposals for a reduced programme for HE decontamination that can be completed within the running time of the contract were submitted for CEC consideration in December 1987. These proposals are also described.

Progress and Results

1. Assessment of the Problems - Reflux Process Control (B.2.)

An earlier progress report described the development of a steady-state model of the reflux process as applied to a WAGR HE. A dynamic simulation has now been developed to study the formation of the condensation zone and its progression down a bank of tubes. The simulation is based on the following decontamination procedure. Fluid at 108°C is circulated downwards through the economiser, evaporator and superheater in series until the exit temperature shows that the tubes have been heated to 108°C. HCl vapour at 108°C is passed upwards over the tubes. Cold fluid is added to the inlet to bring the economiser tube temperature down by a few degrees. A condensation zone will form at the top of the economiser and gradually move downwards. Its rate of descent is controlled by the temperature of the coolant. The oxide layer on the gas side of the tubes, which incorporates the radioactivity, will be dissolved in this zone.

The condensation rate of the vapour is controlled by the heat transfer processes in four regions: from the vapour to the condensate film surface; through the film to the outside of the oxide layer; through the oxide layer and tube wall; and from the inside tube wall to the coolant fluid. Figure 1 illustrates the temperature profile through the four regions. Differential equations describe the relationship between the temperature of a region and the radial heat fluxes into it and out of it. Longitudinal conduction is ignored. Partial differential equations describe the behaviour of the vapour density and the temperature of the coolant fluid in terms of time and longitudinal position. The basic equations solved in the model are listed in Appendix 1. Figure 2 shows the model's simulation of a condensation zone moving down the economiser when the coolant inlet temperature is 5°C below that required for equilibrium with the boil-off rate of HCl. Figure 3 shows the effect of various combinations of coolant inlet temperature and flow rate on condensation zone shape and size. The lower flow rate yielding a shorter region is preferable because it implies a lower input temperature which will be easier to control. When the zone reaches the lower end of the tube bank the coolant outlet temperature will show a sharp change.

It may be concluded from the studies carried out so far that an HCl refluxing process on a WAGR HE is feasible and can be controlled by balancing the heat inputs.

2. Design of the Decontamination Plant (B.2.)

In order to keep the quantity of effluent down to a level that can be transported to the BNFL plant in a road tanker, it is proposed to use a recirculating spray decontamination technique and apply it to one superheater bank only. The reagent will be nitric acid because it is compatible with BNFL waste streams, although laboratory trials have shown it to be a less effective decontaminant than HCl. Following a feasibilty study by Engineering Projects Design Office, Risley, Northern Engineering Industries International Research and Development Co. Ltd (NEI-IRD) were invited to prepare a cost estimate for a decontamination plant based on this specification.

The acid, initially 0.5 molar HNO_3 mixed with 0.0025 molar citric acid as a complexing agent, will be applied to the top of the superheater by means of spray bars. These will be inserted through drillways in the boiler shell and gas duct in order to reduce radiation dose to personnel during installation and subsequent maintenance. The reagent falling through the tube bank will be gathered in the existing conical transition duct below the superheater and piped to a filter and then to an acid sump from which it will be pumped to the spray bars. A total of about 1500 litres will be needed to charge the circuit. The process will be continued until the measured radiation level at the superheater ceases to fall. After sampling to determine its radioactivity and chemical composition the effluent will be pumped through a final filter into a road tanker and transported to BNFL for disposal. The process will be repeated using fresh reagent which may be of greater strength if necessary - the objective being to reduce the superheater gamma dose-rate from its present level of 15 $mSv.h^{-1}$ to 0.5 $mSv.h^{-1}$ at contact, to permit "hands-on" dismantling. Activity take-up at the filters will be continuously monitored. The filters will be conditioned for disposal to the Drigg low-level waste repository. The basic scheme, which has yet to be designed in detail, is shown in Figure 4.

CEC approval of the proposal, including a programme of support work by Windscale Laboratory, is now awaited.

<u>Appendix 1</u>

The basic equations solved in the model are given below.

1. HCl Vapour Density

$$\frac{\partial \rho_v}{\partial t} + \frac{\partial \rho_v}{\partial z} \cdot v_v = \frac{-2\pi r_e (T_v - T_f) k_{vf}}{A_v [h_{vf} + h_f (T_v - T_f)]}$$

2. Vapour to Condensate Film Heat Transfer Coefficient

$$k_{vf}(z,t) = k'_{vf} \cdot \rho_v(z,t)$$

3. Tube Fluid Temperature

$$\left[\frac{\partial T_w}{\partial t} + v \cdot \frac{\partial T_w}{\partial z}\right] c_w r_e \rho_w = 2(T_i - T_w) k_{iw}$$

4. Condensate Film Temperature

$$\frac{dT_f}{dt} \cdot c_f t_v \rho_f = k_{vf}(T_v - T_f) - k_{fo}(T_f - T_o)$$

5. Tube and Oxide Outer Temperature

$$\frac{dT_o}{dt} \cdot \rho_m c_m \cdot \frac{1}{3} A_m = k_{fo}(T_f - T_o) - k_{oc}(T_o - T_c)$$

6. Tube and Oxide Centre Temperature

$$\frac{dT_c}{dt} \cdot \rho_m c_m \cdot \frac{1}{3} A_m = k_{oc}(T_o - T_c) - k_{ci}(T_c - T_i)$$

7. Tube and Oxide Inner Temperature

$$\frac{dT_i}{dt} \cdot \rho_m c_m \cdot \frac{1}{3} A_m = k_{ci}(T_c - T_i) - k_{iw}(T_i - T_w)$$

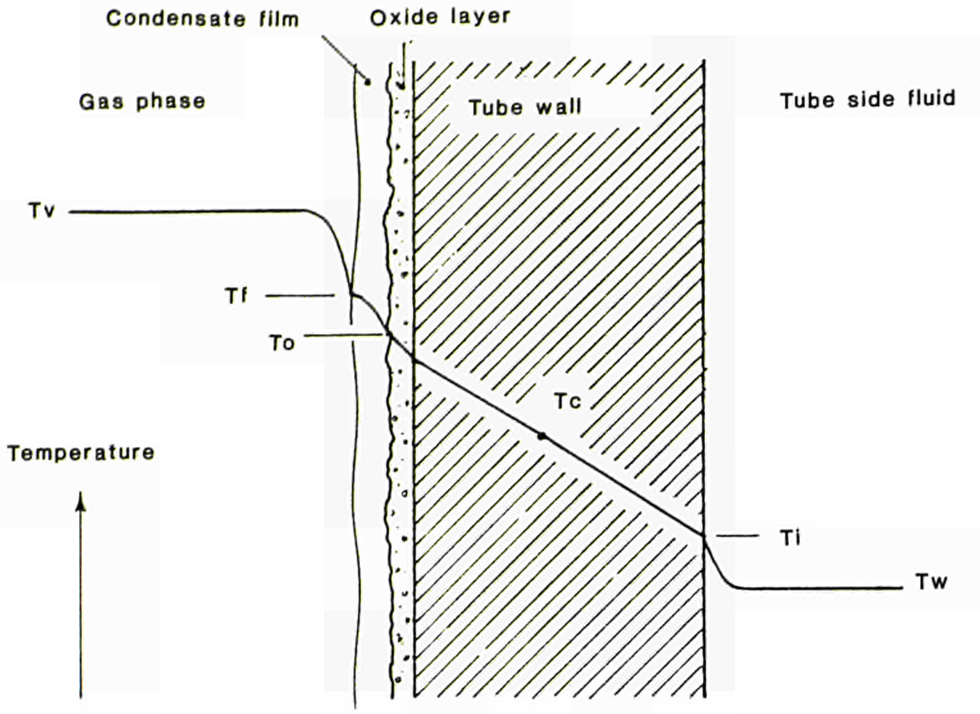


Figure 1. Temperature profile through the heat transfer regions

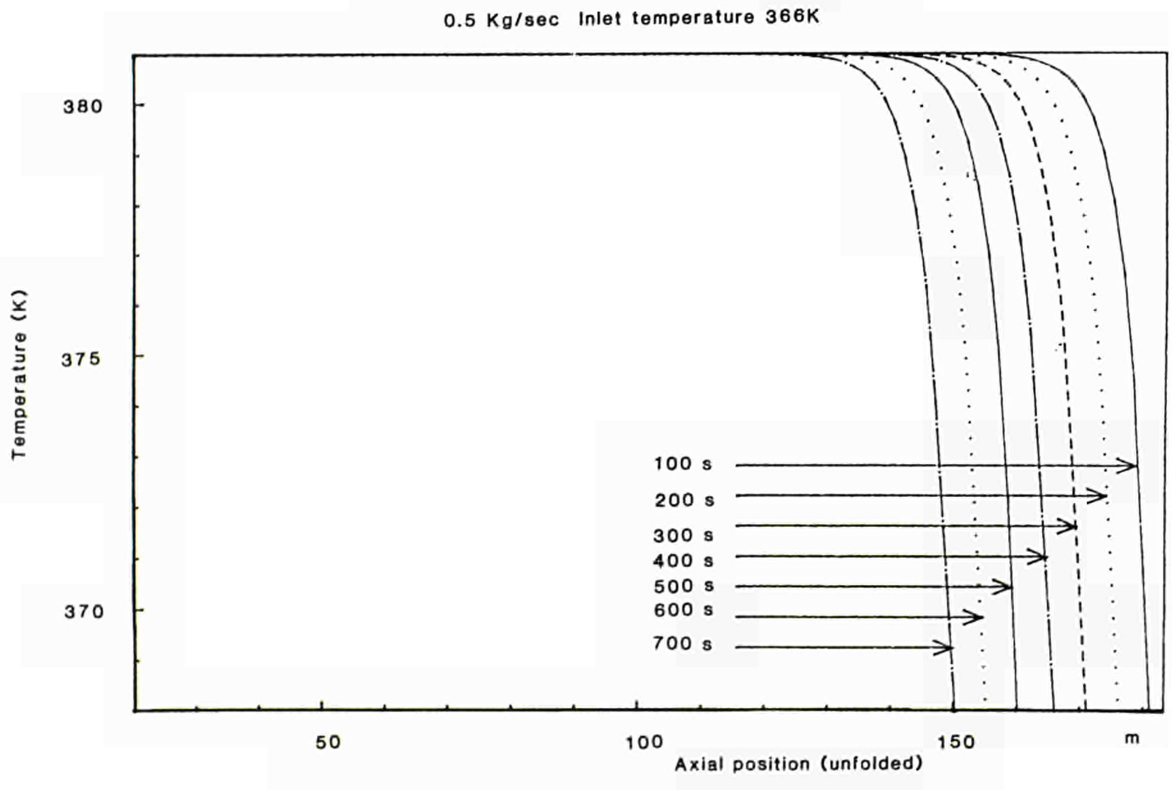


Figure 2. Movement of condensation zone

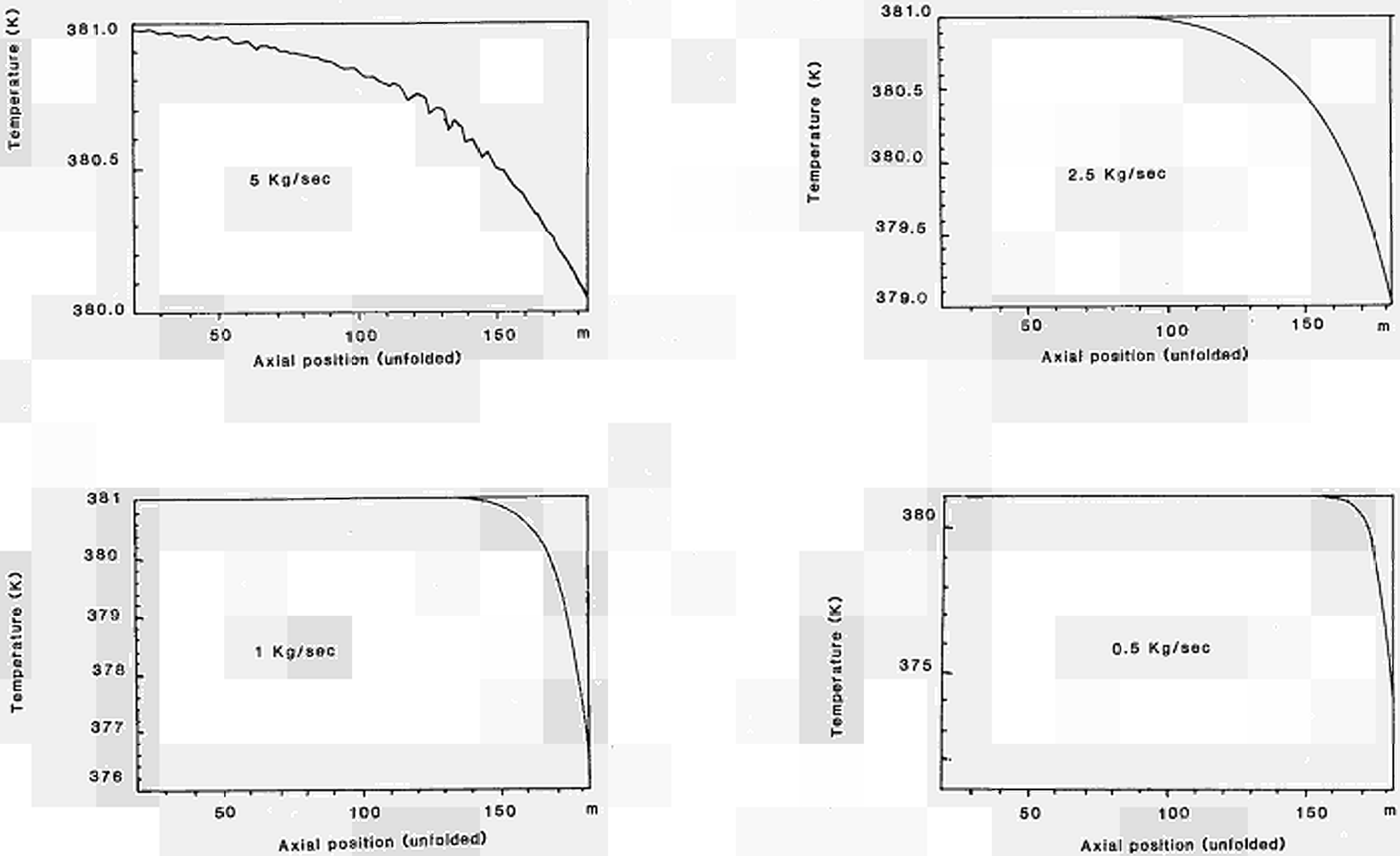


Figure 3. Effect of coolant temperature and flow rate on condensation zone

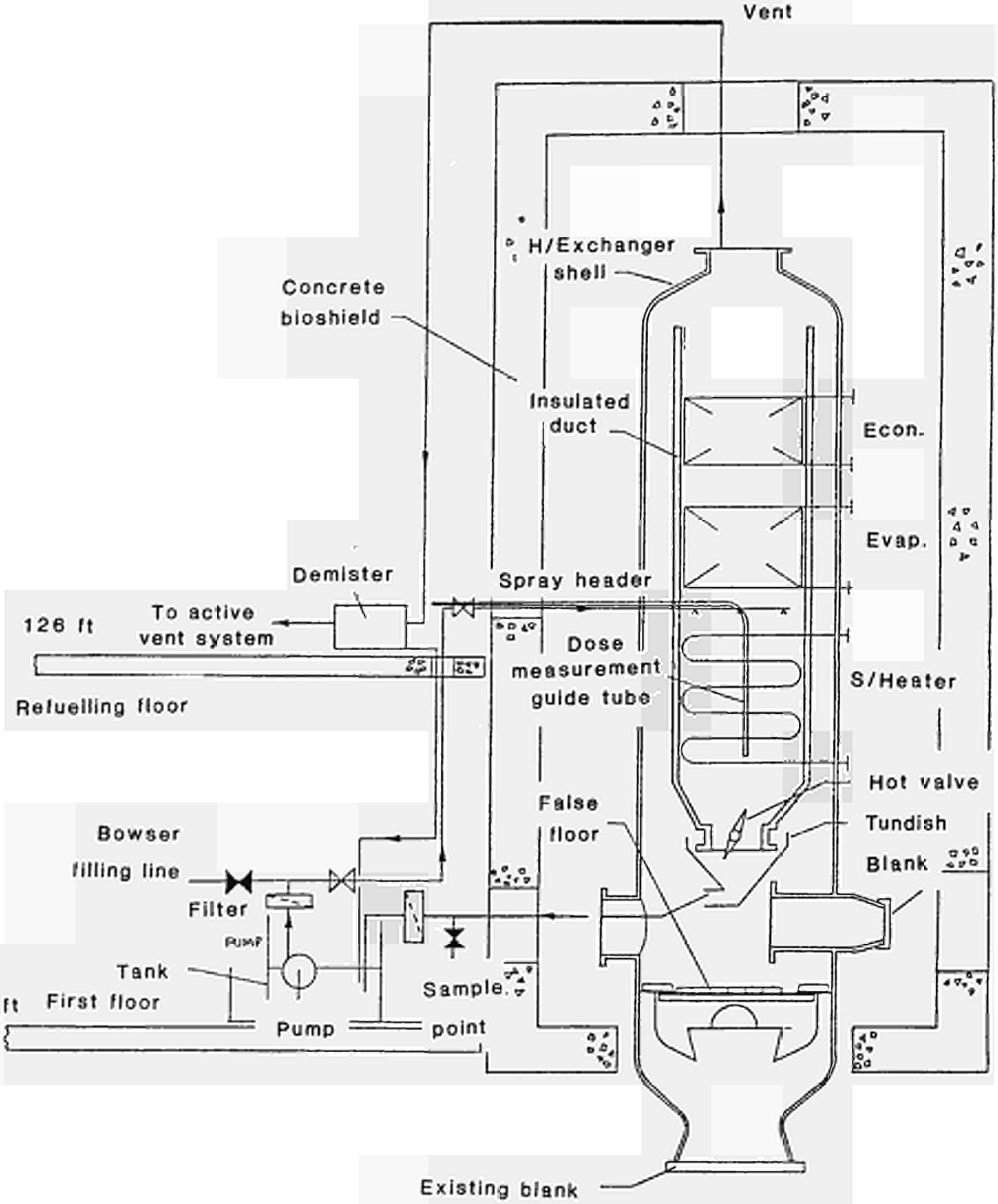


Figure 4. Heat exchanger decontamination

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 1987 started with completion of Post operational Clean-out (POCO) of the "Wet"end of the plant (ie where solutions/slurries were handled) so that the liquors could be handled by a plant due to close down in February 1987.

This was followed by removal of redundant equipment inside and outside the cabinets in this section.
Attention was then transferred to the opposite end of the facility to clean out the ball-mill etc where MO_2 was homogenised. This sequence of operations was dictated by the need to remove the mill and its containment in order to create an exit route for all other large plant items and their cabinets.
On development aspects the value of the Reusable Modular Containment (RMC) has been proven on the ball mill operations which have been lengthy, very highly contaminated and involved much physical effort.
Similarly the ability of Tie-Down coatings to minimise airborne contamination and spread of activity has been invaluable.
In-situ Plutonium assay equipment has assisted in dealing with components as these have been removed but attempts to measure the Ball-Mill and it's cabinet have not yet been completely successful.
Freon cleaning plans continue with design completed and manufacture, hazard study and nuclear safety assessment all in hand. Detailed design and costing for an electrolytic cleaning unit in an existing glove box has been completed whilst a range of tools for in-situ size reduction trials have been obtained.

Progress and Results

1 Planning and Safety Documentation (B1)

Detailed design and procurement of equipment are essentially completed as is the dismantling schedule. Decontamination and dismantling work is proceeding under a series of Plant Modification Proposals, as proposed in the original Preliminary Safety Appraisal, which allow for the continuously changing situation as decommissioning progresses. Appropriate operating and engineering instructions are also prepared and the Dismantling Schedule amended.

2 Clean-Out and Initial Dismantling of Wet Chemistry Suite (B2/B3)

Cleaning of this section was undertaken with dilute Nitric Acid, heated and recirculated and finally used for the geometrically safe storage tanks. Table I details material recovered and manhours and radiation incurred for this and other operations. Radiation levels were typically reduced three to six fold by these operations.
Subsequently all in box and redundant glove box exterior equipment was dismantled and removed - some 90 % of items from outside the cabinets was suitable for shallow land burial given careful segregation, surface activity measurement and simple decontamination.

3 Partial POCO of Dry Powder Facilities (B2)

This section of the plant was refurbished and run to clear itself, redundant drives etc were then removed. Further Initial Decommissioning and cleaning, eg with Freon, will be essential.

4 Emptying and Removal of Ball Mill (B3)

The plant layout requires this to be removed to create an exit route. The Ball Mill however is a horizontal 75 cm right cylinder, made of steel reinforced rubber and containing 900 Kg of balls.
Additionally it was shielded by borated polythene blocks and 2 cm lead sheet located between its casing and glovebox containment.
Final emptying of the mill recovered much MO_2, sieved from the balls, after which a temporary containment (see below) was used to access the mill glove box for removal of shielding and finally the mill itself and its casing.

5 Development Aspects (B1)

Considerable progress has been made in all the development areas for which the Coprecipitation plant decommissioning is regarded as a pilot project. These are discussed below.

6 Freon Cleaning (B1)

Initial trials confirmed the basic concept of low pressure (10 Bar) Freon jetting for removal of gross powder deposits work. Work then proceeded to a detailed design phase, now completed, during which outstanding questions on ventilation philosophy, solvent recovery and filter and pump design were resolved. The prototype unit is now in course of manufacture whilst hazard studies proceed and nuclear safety assessments are carried out on the unit itself (almost completed) and on its application to several glove box containments in the pilot plant.

7 Reusable Modular Containment (RMC) and Tie-Down Coatings (B3)

These techniques have already proven their worth on this project. The large RMC used on the ball mill removal operation coped extremely well with the lengthy operation and rigorous conditions, saving significant wastes and with no structural problems. In fact slight activity leakage at one point was traced to the junction between RMC and Ball Mill cabinet, effected with flexible sheet due to the complex shape, whilst the RMC received a clean bill of health.

Similarly the use of Tie-Down coating within the RMC, within the Ball Mill Cabinet and on items taken from the cabinet, permitted fixation of large amounts of loose activity which would have given very high air counts and spread of contamination, albeit within the RMC. For example loose activity levels on the RMC floor were frequently 3000 cps after a working period and this could be reduced to 10 cps by a single Tie-Down coat.

8 Plutonium Assay Equipment (B2)

All the items of "Decommissioning In-Situ Plutonium Inventory Monitors" (DISPIM) have been delivered and are in use. The gamme and neutron probes have allowed location of some high spots and this work will be extended throughout the plant. They have also given rough measurements of larger waste items, eg the polythene shield blocks, to be made. In-situ measurement of the whole Ball Mill cabinet with the Modular Neutron Coincidence counter is also being attempted, with mill in position and since it's removal. This is a large and complex unit however and analysis of results is not completed.

The miniature alpha probe has also been of value in helping to segregate material for shallow land burial and also in locating leakage of activity as mentioned previously.

9 Electrolytic Decontamination (B1)

The detailed design of equipment and modifications for the existing cabinet has been completed and costed. Meanwhile AERE Harwell have been examining the prospects for a portable unit, capable of movement to other locations in the future. A decision on the most appropriate solution is imminent.

10 Tooling for In-Situ Size Reduction (B3)

Based on experience in UK and elsewhere a range of hand power tools has been obtained for this project and initial trials of their suitability are in hand.

These include :

Orbital jig saw for sectioning Plastic Panels.
Crimp/Shear tool for pipework up to 35 mm Stainless Steel.
Nibbler for stainless plate up to 3 mm.
Reciprocating and circular "cut-off" saws for thicker Stainless and other sections.
Angle Grinder and Portable Bandsaw for difficult materials and irregular shapes.

Table I

SUMMARY OF OPERATIONS, MAN HOURS, RADIATION AND WASTE

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	-	225	N/A	1.0	2.0	-
Engineering Refurbishment	40.5	17.3	1350	600	1.4	-	-
Liquor Recirculation	7	1.9	670	100	-	-	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	-	450	20	-	-	2.9 Kg as Nitrate
Ball Mill and Product Offtake Preparation	20	5	1450	70	0.4	-	1.6
Ball Mill Emptying	1.8	-	100	-	Data not available		26.0
Other Cabinets	3.2	-	650	100	1.4	4.6	2.4

PCM = Plutonium Contaminated Material for Storage

SLB = Shallow Land Burial

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 - December 1988
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 decommissioning of a technical laboratory by removing thin wall surfaces was completed.

The volume of scraps from the building structure and inventory of NUKEM-A as well as the costs were calculated.

To start the work of contaminated metal treatment by melting we contacted a corresponding company and started experiment planning.

Progress and Results

1. Exemplary decommissioning of a technical laboratory (B.4)

1.1 Experimental work

In continuation of the work last year we experienced methods for the removal of contamination of the walls. For determination of specific removal efficiency and costs the walls were divided in testing areas of 1 m^2 each. The methods tested were a hot-air blower and two grinding machines. The removal efficiency of the colour sheets with the hot air blower depended strongly on the type of subsoil. The grinding machine with sandpaper had removal efficiencies of 0.4 - 0.5 m^2/h, significant higher values were reached with a diamond studded grinding head (1.2 - 1.6 m^2/h).

Because of the high investment costs we could not get experience with automatic facilities. Depending on the manufacturer and type of the machines the removal efficiencies are between 10 - 30 m^2/h. The technical performance of the removal at walls can be realized with a basis mobile facility like e.g. BROKK (Atlas Copco), which can be equipped with several automatic grinding heads.

1.2 Comparison of demolition volume and cost

Concerning the construction plans of NUKEM-A facility and the actual estimation of final storage costs in the KONRAD mine we compared demolition volume and costs for two alternative methods:

Method 1: No removal of contamination, total building material is calculated as waste

Method 2:. Removal of contamination and separation of demolition volume in radioactive waste and rubble

Without removal the demolition waste volume is calculated to 2000 m^3. A removal depth of 10 mm leads to a waste volume of 350 m^3. The cost situation is as follows:

Method 1: Demolition: 0.8 million DM, final storage: 10 million DM

Method 2: Demolition: 0.5 million DM, final storage: 1,7 million DM

2. Discussion of a decommissioning concept (B.4, B.9)

According to the calculations and experiments the decommissioning of NUKEM-A is less expensive by removal of thin contaminated layers at walls and floors. To be sure, that after this removal the walls and floors could be treated as not radioactive they have to be measured. During our exemplary decommissioning of a technical laboratory we detected beyond the removed layers a lot of irregularities like cracks or repaired gaps. At these points the proof of removal efficiency of the contamination cannot be given, these irregularities must be treated separately. The estimated additional costs could be higher than the savings of surface removal. During our further work the methods of leaching from Uranium ores will be experienced.

3. Material inventory of NUKEM-A (B.7)

To estimate waste volumes of the plant inventories, all buildings and rooms were controlled and the inventory was listed depending on materials, sizes and weights, separated into inventory of controlled areas with contamination and not radioactive inventory, and into burnable and not burnable materials. Burnable materials were divided into PVC, wood, and others (Polymers, textiles, card board). Not burnable materials were divided into steel, other metals, and others (ceramics, stone, isolation materials, glass). Of an effective area of 8300 m^2 70 % are used as controlled areas, 30 % as supervised areas. The results of the inventory control are summarized in table 1. In total we expect about 700 t of inventory waste. Compared with the demolition waste of 2000 m^3 resp. about 4000 t the treatment of inventory waste is of less importance concerning economic consequences.

4. Metal waste treatment by melting (B.8)

To check the possibility of contaminated metal recycling by melting we contacted Siempelkamp company. The composition of NUKEM-A metals is suitable for their melting process and for the resulting material. An open question which has to be answered during the next reporting period is the treatment license of Siempelkamp für NUKEM's isotope composition (enriched Uranium). As a second way we plan own melting experiments in the lab and technical scale (up to 50 l drums) for the determination of activity distribution (slag, melt) and decontamination efficiency.

Table 1: Mass belance of NUKEM-A inventory (figures in metric tons)

	contaminated	not contaminated	total
burnable:			
PVC	10	9	19
wood	3	2	5
others	4	4	8
total	17	15	32
not burnable:			
steel	540	50	590
other metals	56	6	62
others	28	3	31
total	624	59	683
Total mass:	641	74	715

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 - June 1988
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.

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.

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

Cleanup of the PIVER pilot vitrification facility, including the process cell (No 74) and all the related rooms, continued during the preceding year despite the major problems involved.

Permanent difficulties complicating this task include the following:

- The increasing formalism related to safety and quality regulations for Basic Nuclear Installations (INB).
- The requirement of maintaining a very high level acid solution storage unit operational in the basement of cell 74.
- Reliability problems with a 20 year old obsolete telemanipulator that has sustained severe wear.

Nevertheless, a number of positive achievements were obtained during this period:

- Decontamination of two rooms adjacent to cell 74 with direct pipe connections to the cell.
- Inventory of all the pipe lines in the cell and of the congested layout of the cell floor.
- Development and testing of "Antoine", a pantograph arm telemanipulator unit equipped with a high efficiency shear.
- Drilling of holes in the cell 74 wall and installation of penetrations to set up a work station with two master-slave telemanipulators.
- Preparation of means for removal of bulky wastes through the cell lock.
- Renovation of the "Caroline" robot which was able to maneuver in the cell without incident in December 1987.

Progress and Results

1. High Level Liquid Storage Tank (B.1.)

Maintaining safety-related and other functions for the concentrated fission product solution storage vessel implies major constraints for the PIVER cleanup operations, and has slowed progress to date.

Authorization was obtained at the end of 1987 after submission of a safety report for reception of fission product solutions from the Pilot Reprocessing Plant at Marcoule during the first quarter of 1988, pending completion of work to be done at the beginning of 1988.

The principal tasks undertaken during 1987 included:

- Reconditioning of the essential safety-related functions of the tanks (cooling, pulsator agitation, off-gas removal and treatment).
- Complete renovation of the PICCADOR sampling system.

2. Decontamination (B.2.)

Two rooms adjacent to cell 74 were decontaminated: the glass melting induction furnace power supply room and the roof of cell 74 including facilities for reagent preparation and distribution in the vitrification process. Conventional and contaminated wastes were prepared and certain equipment items were recovered (racks, medium frequency generators, electrical switch cabinets and pumps).

- Examination and Inventory of Cell 74

The high irradiation level in the cell is largely attributable to the presence of numerous operating wastes, principally radioactive glass, lying on the floor. The first task will therefore be to remove these wastes, which are difficult to handle.

The pipe lines were also accurately inventoried, and those which must be left intact to ensure normal operation of the high level liquid storage unit were identified.

3. Preparation for Installation of Handling Equipment (B.3.)

· Master-Slave Telemanipulators

A work station was set up around a window on the front wall of the cell. Mechanical penetrations 357 mm in diameter and 1400 mm long were installed in July 1987 to accommodate telemanipulators. This proved to be a delicate operation: after preliminary work to set up structural, handling and ventilation provisions, two holes were drilled in the cell wall, thus directly interconnecting two zones with very different irradiation and contamination levels.

Most of the wastes generated during this work were not radioactive: these included the drill bit cooling water (1160 liters) and the concrete cores. Some contamination was produced when the drill bit penetrated into the cell itself, but his was limited both in volume and intensity (0.03 $mGy \cdot hr^{-1}$).

· "ANTOINE" Intervention Device

Development work and nonradioactive testing on the ANTOINE intervention device were completed in December 1987.

- A high performance shear was developed with very high cutting capacity for its size, powered by a high pressure (800 bar) compact hydraulic unit that has proven very reliable and can be installed on the device itself.
- Remote handling tongs were developed to hold pipes while they are being cut.
- The equipment was adapted to the requirements of operation in radioactive environment (notably to allow easy replacement of the cutting blades).
- A full range of additional tools (disk grinder, pneumatic tongs, extension for floor-level work, shovel and scraper) was developed.

The following performance has been obtained to date:

- Maximum load: 60 kg
- Maximum cutting dimensions on stainless steel tube:
  - 60 mm diameter
  - 4 mm thick
- Cutting time: approx 30 seconds.

Tests were conducted to identify sequences or motion restrictions and to prepare operating procedures and methods. After personnel training "Antoine" will be ready for entry into cell 74.

· Direct Waste Removal

Provisions for direct removal of bulky wastes were designed and implemented. These include adapting the exit locks and providing them with handling equipment, as well as renovating the floors and stainless steel sheeting with improved means for rolling heavy concrete shells.

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 - December 1987
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 within a prefixed time schedule and in respect to safety and cost.

This work complements contract N° FI1D-0024 "Decommissioning of very large glove boxes" of the present research programme.

B. Work Programme

B.1. Conception of a work procedure, preparation of special equipment, conditioning and dismounting of the glove box.
B.2. Transportation of the glove box parts.
B.3. Dismantling of the glove box parts.
B.4. Conclusive assessment of work.

C. Progress of Work and Obtained Results

Work in this contract has been discontinued for technical reasons. A revised work programme will start mid-1988.

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 - December 1988
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 to the general on 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 present research programme aims at collecting experience by large-scale melting of about 400 t of radioactive metal waste out of the KRB-A decommissioning, up to concentrations of 500 Bq/g.

The work will be executed with an induction-heated furnace (capacity ca. 3t) in the controlled zone of the KRB-A turbine building, probably after some backfitting of the existing filtration systems. The furnace will be leased, and is provided for reuse on another site.

The study will result in a clear statement if the above procedure has a potential for a 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 dismantling operations in the reactor building and for the installation and operation of a melting furnace.

B.4. Execution of the melting test programme, including decontamination for reducing the activity level (if needed), dismounting and conditioning of scrap into feeding drums, installation of the melting furnace and its auxiliary equipment, and execution of the main melting and casting programme, followed by an analysis of the prevailing radiation level during the melting operations.

B.5. Study of the final decommissioning of the melting furnace and the arising secondary waste.

B.6. Conclusive assessment of the usefulness of large-scale melting of metal scrap with a radioactivity level higher than 74 Bq/g.

C. Progress of Work and Obtained Results

Summary

An evaluation of the technical and economical aspects of large scale melting in the controlled area of KRB A has shown that this intent is reasonable. Even licensing seems to be possible. Nevertheless the amount of 400 t which has been contracted with the European Community is not on the economical side because of the high capital costs for the necessary installations.

The break even point in comparison with other processes has been calculated in the range of about 1500 t of melted steel. The advantage of melting arises mainly from volume reduction and therefore high costs for final disposal speak for large scale melting.

Because the installation of the furnace in the controlled area of KRB A results in higher specific costs than expected it has been proposed to install and operate the melting furnace in cooperation with a service company on another site.

Progress and Results

1. Selection of decontamination and cutting methods (B.2)

Because of their high dose-rate for some components of the reactor building a predecontamination is desirable before dismantling if this can be done efficiently and with a reasonable effort. In order to get an information on the applicability of different acids, the effect of decontamination of diluted citric acid, oxalic acid, and of mixtures of citric and oxalic acid and of oxalic and hydrochloric acid was tested. Four stainless steel tube segments contaminated on the inside with cobalt 60 and cesium 137 were treated during half a year at room temperature. The decontamination experiments with diluted acids demonstrated that no sufficient decontamination effect could be achieved by room temperature. The test series have to be continued by higher temperatures and by a variation of different decontamination fluids.

The choice of cutting techniques for the preparation of plant components was derived by the experiences gained during the works in the turbine hall. There is always a great advantage for using mechanical cutting methods, first of all sawing. If mechanical cutting is not possible or over all uneconomical thermal cutting will be used (acetylene or propane for ferritic steel and plasma arc for austenitic steel). The production of radioactive aerosols has to be checked carefully during these activities and suck off filter systems must be used as well as protecting masks.

The segmentation of heat exchanger bundles particularly those of the secondary steam generator (SSG) will be executed by a new technique developed at KRB. The upper portion of the SSG will be dismantled by torch cutting using acetylene or propane with oxygen because the contamination on the secondary steam side is below 1000 Bq/cm^2. The lower portion contains the heat exchanger bundles (about 2000 U-tubes) with an inside contamination of 80000 Bq/cm^2. Therefore the lower part of the SSG will be filled with water (primary and secondary side) and afterwards frozen to ice. A special bandsaw fixed on the outside of the SSG will cut horizontally through the vertical SSG pieces of steel and ice of about 1 m length. After deicing the tubes will be stored in shielded containers. The main purpose of the ice is shielding, keeping the structure and collecting the saw-mill waste. A special bandsaw for diameters up to 2.2 m has been ordered.

2. Planning for the installation and operation of the melting furnace (B.2)

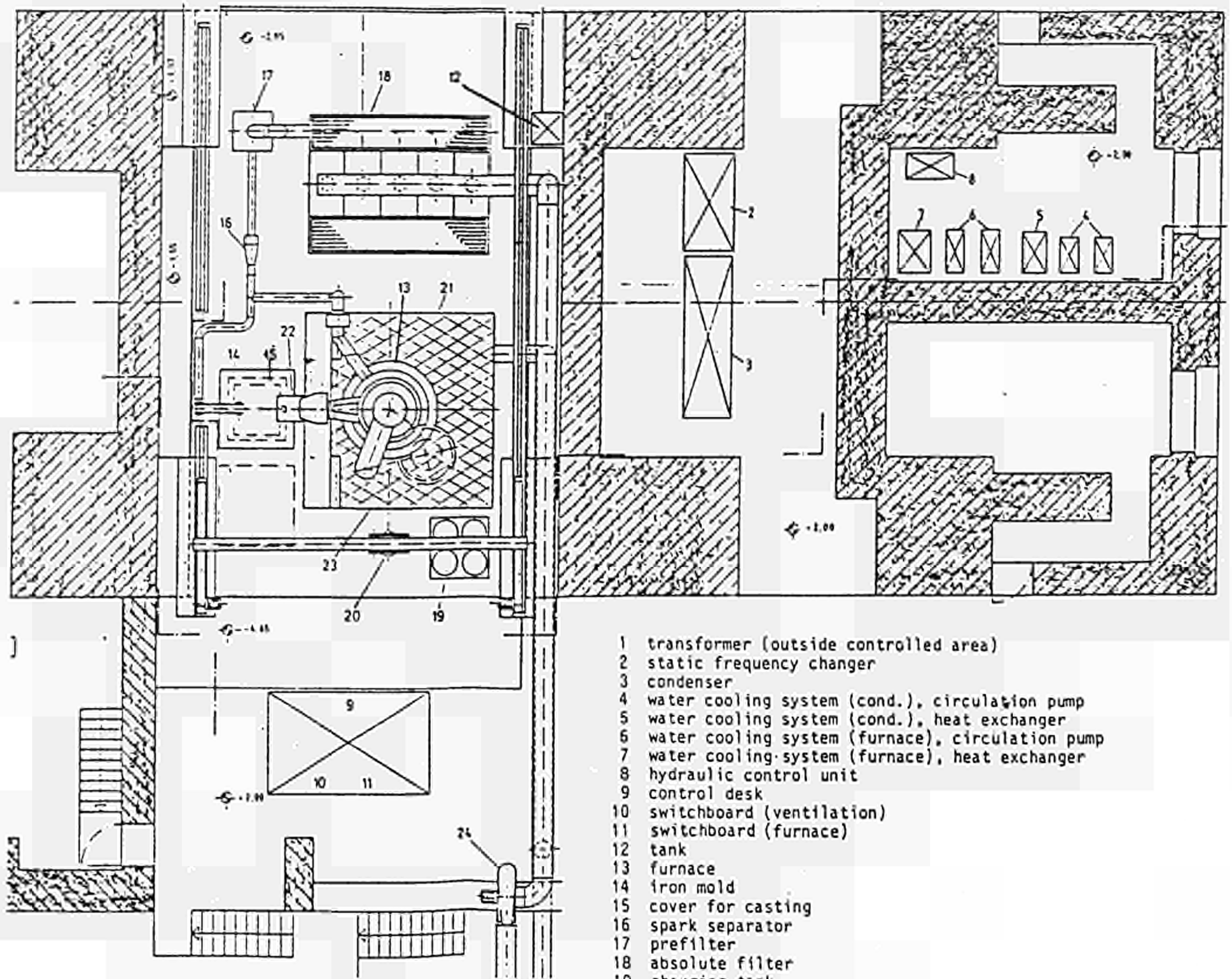
The service companies GNS/Siempelkamp and Noell have finished a detailed plan for the installation and operation of the melting furnace. The technical and economical evaluation of these two concepts demonstrated that a large induction furnace (between 8 and 16 tons capacity) and a double filter and suck off system for the furnace and the compartment air ventilation includes all advantages in an optimum concept. The large furnace saves costs for disassembling and the double filter and suck off system increases safety in case of a ventilation failure. An overview of these main systems is given in figure 1 and 2.

3. Radiation protection (B.2.)

For radiation protection the room of recirculation-loop III had to be cleaned and some components from earlier activities had to be transported out of the working scope.
Around the recirculation-pump a 2 meter high shielding wall with concrete-press-stones and lead-stones was built up and the recirculation-pipe at the entrance of the loop was shielded with lead-plates. Resultant on this measure the dose-rates in the working scope could be reduced to a value of max. 150 μSv/h.

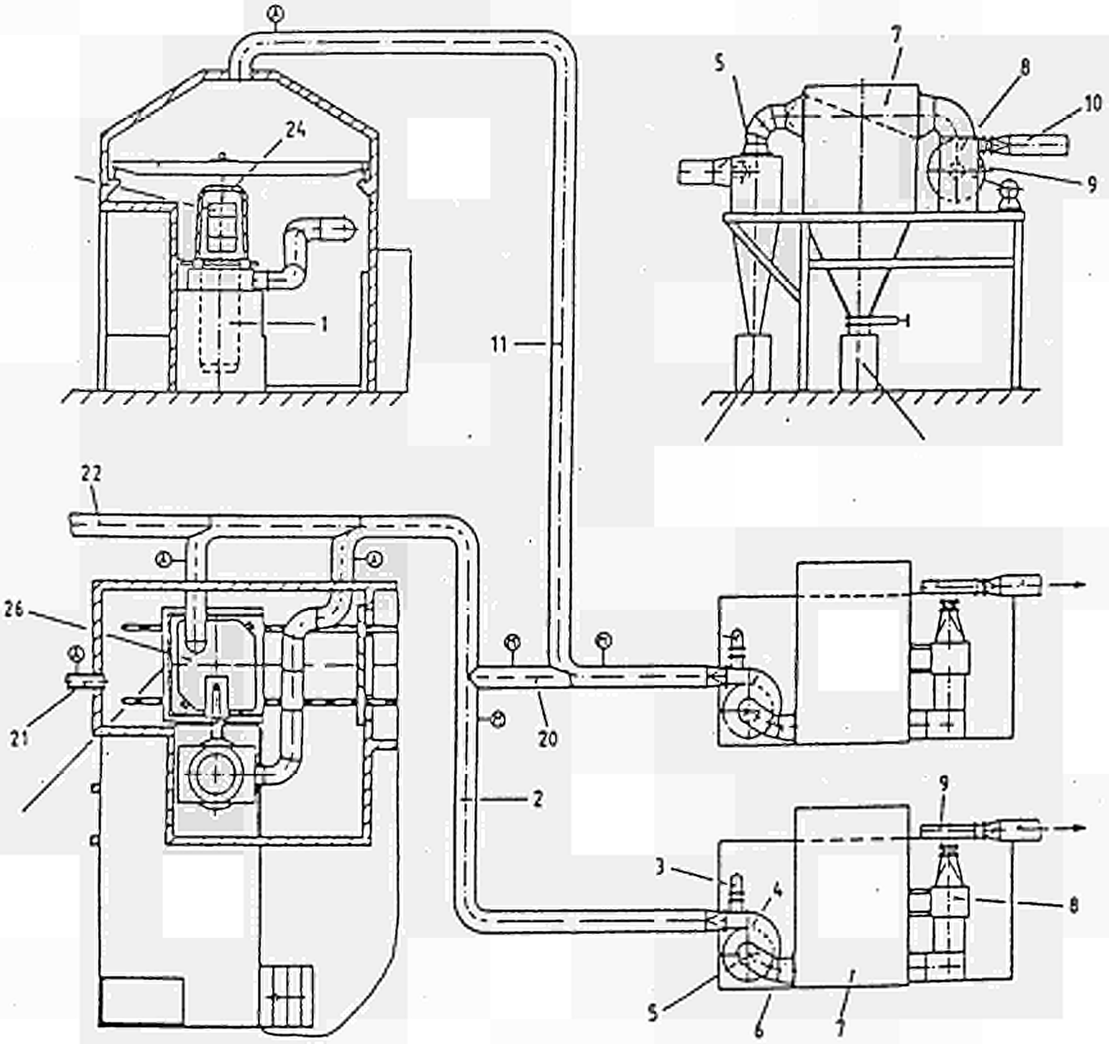
4. License procedure (B.3.)

For the future decommissioning activities in the reactor and auxiliary building KRB has applied for a license by the Bavarian ministery for environmental protection in December 1987. Additional papers with detailed specifications of the applied measures are still in production.



1 transformer (outside controlled area)
2 static frequency changer
3 condenser
4 water cooling system (cond.), circulation pump
5 water cooling system (cond.), heat exchanger
6 water cooling system (furnace), circulation pump
7 water cooling system (furnace), heat exchanger
8 hydraulic control unit
9 control desk
10 switchboard (ventilation)
11 switchboard (furnace)
12 tank
13 furnace
14 iron mold
15 cover for casting
16 spark separator
17 prefilter
18 absolute filter
19 charging tank
20 crane
21 shell
22 pouring spout
23 window
24 ventilator

Fig. 1 Concept for the installation of a furnace at KRB A recommended by the company Noell, Würzburg



1 furnace
2 exhaust tail pipe
3 bypass pipe
4 bypass mixture
5 prefilter
6 flame trap
7 coarse filter
8 absolute filter
9 exhaust-gas ventilator
11 exhaust tail pipe
20 pipe branching
21 fresh-air entrance
22 bypass pipe
24 charging hood
26 container
Ⓜ shutoff valve, motor-driven
Ⓐ shutoff valve, hand drive

Fig. 2 Concept for the installation of a furnace at KRB A recommended by the companies GNS/Siempelkamp

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 - October 1988
Project Leader: P. Gerontopoulos

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 disconnection of all glove boxes belonging to the processing lines specified in the previous Annual Progress Report and their transport to the waste treatment service of CEN-CADARACHE (SPPC-SAR) have been terminated. Final dismantling and disposing of this material into waste drums has to be postponed for technical reasons for about 12 months.

In the case of the experimental air lift transfer system of PuO_2 powders located in LAB B-8 and 010 of SCTCPu, the transport to SPPC-SAR has been preceded by the recovery of important quantities of fissile material using a powder aspiration device constructed to this end.

Additionally to the processing lines specified in the previous Annual Progress Report, a fuel transfer tunnel connecting LAB 2, 3 and 4 of SFECPu has been partially dismantled, decontaminated and conditionned in 100 liter metallic drums. Preliminary estimates based on surface alpha counting show that all metallic wastes generated to date in this particular campaign respond to the swallow land burial limit (370 MBq/T) established in France by the Service Central de Sûreté des Installations Nucléaires (SCSIN). Confirmation of these data will be possible in the second half of the current year using a neutron activation Pu monitoring system already commissioned by SPPC-SAR.

Progress and Results

1. PuO_2 powder transfer systeme located in LAB-8 and LAB 010 of SCTCPu (B.1, B.3)

The single units composing this system, consisted of an air lift powder transfer circuit ($2m^3$), seven glove boxes containing auxiliary equipment ($4.9m^3$) and the connecting ducts ($4.5m^3$). They have been separated from each other and from the auxiliary services of Building 272 (SCTCPu), yielding 79 voluminous waste packages sent to SPPC-SAR for further dismantling and conditioning.

During the preliminary dismantling operations at SCTCPu, plutonium powders corresponding to 1187 g of Pu have been recovered from different sections of the system using an aspiration circuit set up on this purpose. The residual amount of plutonium contained in the voluminous waste packages delivered to SPPC-SAR was estimated to a total of 829 g Pu. An additional quantitiy of 172 g of Pu has been discharged in non voluminous solid waste packages (disposed in 100 liter metallic drums) and liquid wastes. The grand total of 2188 g corresponding to the Pu recovered during the preliminary dismantling operations at SCTCPu and that discharged into the various waste streams is in reasonable agreement with that resulting from inventory data (2292 g Pu). The difference is attributed to the incertitudes of plutonium monitoring in the voluminous waste packages.

The main conclusion to be drawn from this dismantling campaign is that the quantities of plutonium put into the voluminous waste packages are too high and that in the absence of further Pu recovery during final dismantling, the totality of the generated solid wastes have to be disposed in deep geological repositories.

2. Fuel transfer tunnel interconnecting LAB 2, 3 and 4 of Building 258 SFECPu (B.1, B.2, B.3, B.4)

This tunnel (total volume 12 m^3) anchored to the ceilings of LAB 2, 3 and 4 of Building 258 (SFECPu) run over the glove box processing lines located in the above said laboratories for an overall length of about 50 meters and was connected to the underlying processing lines by vertical branches. It was mainly composed of all-metal parallepipedic modules (typically 0,5 m x 0,5 m x 1,5 m) held together by nuts and bolts and, similarly connected, small dimension glove boxes (0,6 m x 0,6 m x 0,8 m) interposed at 13 points of the installation, for control and sorting out purposes. Fuel movement from one point to the other of the tunnel was assured by means of a motorised monorail transport device, and between the tunnel and the main processing line by lifts housed in the vertical branches.

A preliminary estimation made before dismantling at SFECPu (visual inspection using a video camera attached to the fuel transport system, smear tests in manually accessibles areas, external babyline monitoring) indicated a total quantity of about 40 g PuO_2 present in the form of a black submicronic powder homogeneously distributed along the lower part of the tunnel. Preliminary decontamination tests indicated that this powder could be wiped out leaving residual contamination levels compatible with the disposal of the metallic scraps generated in this particular campaign as non TRU wastes. Although all wastes arising from the operation of Fuel Fabrication Complex of SFER-CADARACHE are for safety reasons conservatively considered as TRU wastes, to be ultimately disposed in deep geological repositories, it was thought that a decontamination exercice of this particular installation to non-TRU levels could offer some pratical interest.

Except for a part of the installation contained in LAB. 2 whose dismantling has been postponed because of budget cuttings, the rest has been partially dismantled and transported to SPPC-SAR for final dismantling (cutting down to pieces of typical dimensions 0,25mx0,25m), cursory decontamination (wiping out the Pu bearing powder with cotton moistened with TRIAXOL), estimation or the residual contamination (surface alpha counting), and disposal in metallic drums of 100 liter capacity.

Presently the following data are available regarding a first group of 13 all-metal modules :

- total volume before dismantling .... 5 m^3
- number of metallic waste drums 34
- number of burnable waste drums 11
- volume of the metallic waste after .... 2,6 m^3 compaction and immobilisation in cement (estimated)
- density of the waste (estimated) .... 2 T/ m^3
- total alpha activity .... 4,5 - 450 MBq

The estimation of the total alpha activity is based on surface alpha counting data showing the presence of significant alpha activity (500-5000 Bq/cm^2) after decontamination limited to the lower surfaces of the tunnel (total area $9m^2$).

These data will be verified in the second half of 1988 using a neutron activation plutonium monitoring system already commissioned by SPPC-SAR.

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

The main objective of this year was the conception and the constuction of the pilot facility called Grolabo. This facility was designed to test products for use in decontamination of the Rapsodie coolant systems and tanks.

Progress and Results

The facility is made of glass-reinforced polyester lined with a teflon film able to withstand acid solutions at temperature up to 85 degree Celsius.

Grolabo includes (fig 1) :

. a 60-liter decontamination tank,

. an exhaust line connected to the ventilation system via a bubbler that is used as a hydraulic valve,

. a circulating pump with variable flow rate from 2 to 5 cubic meters per hour, a heating vessel with three immersion heaters,

. connecting pipes, unions and valves,

. a spray and filling system,

. and measuring instruments.

The system also includes two 120-liter effluent treatment reactors made of polyvinyl chloride.

All system equipment is installed over a stainless steel retention vat.

The facility includes separate decontamination units for qualifying two different methods : spraying and immersion on specimens measuring 300 by 300 by 3 mm.

Subsequent plans call for installing a RAPSODIE component, a valve for example, between two flanges to observe the effects of repeated contact with decontaminant solutions at high flow rates in portions of the system exposed to all the RAPSODIE decontamination.

After the test, the solution is drained into one of the neutralization reactors to determine the decontamination factor. After neutralization the contents of the tanks are transferred to the building's radioactive liquid waste system.

The pump draws the solution from the bottom of the decontamination tank and discharges it into the heating vessel from which it returns to the tank. Two systems may be used : spraying or filling. The solution is circulated through the process lines simply by starting the pump.

The three immersion heaters in the vessel maintain the solution at a set point temperature ; temperature control could be obtained by a single heater. A simple on/off regulating system is used.

The failure analysis should result in a reliable system allowing the facility to be operated 24 hours a day with minimum nighttime surveillance.

The instrumentation is based on the failure analysis results and covers the following parameters :

-In the event of a current overload the frequency variator on the pump power supply line trips the pump contactor to prevent restarting.

-Oxidation-reduction potential is measured at the heating vessel inlet with a strip chart recorder.

-Hot spot temperature in the vessel is monitored to detect overheating and actuate a process controller in order to shut off the heating system at a threshold temperature of 90°C.

-The vessel outflow rate is measured with a rotary flowmeter protected by a strainer and equipped with a low flow rate contactor set to operate at 2 cubic meters per hour.

-Fluid level in the decontamination tank with automatic heating shutoff if the level drops below 40 cm.

-Temperature at the bottom of the decontamination tank is used to control the heating regulation system.

-Fluid presence in the retention vat.

Process control functions are implemented in accordance with the failure analysis to shut off the pump and the heating system according to measurements inputs.

Additional fire alarm and radiological incident warning provisions are included at the site surveillance headquarters.

The pilot facility is now operational. Our next work will cover the decontamination of RAPSODIE components in Grolabo under conditions representative of industrial application.

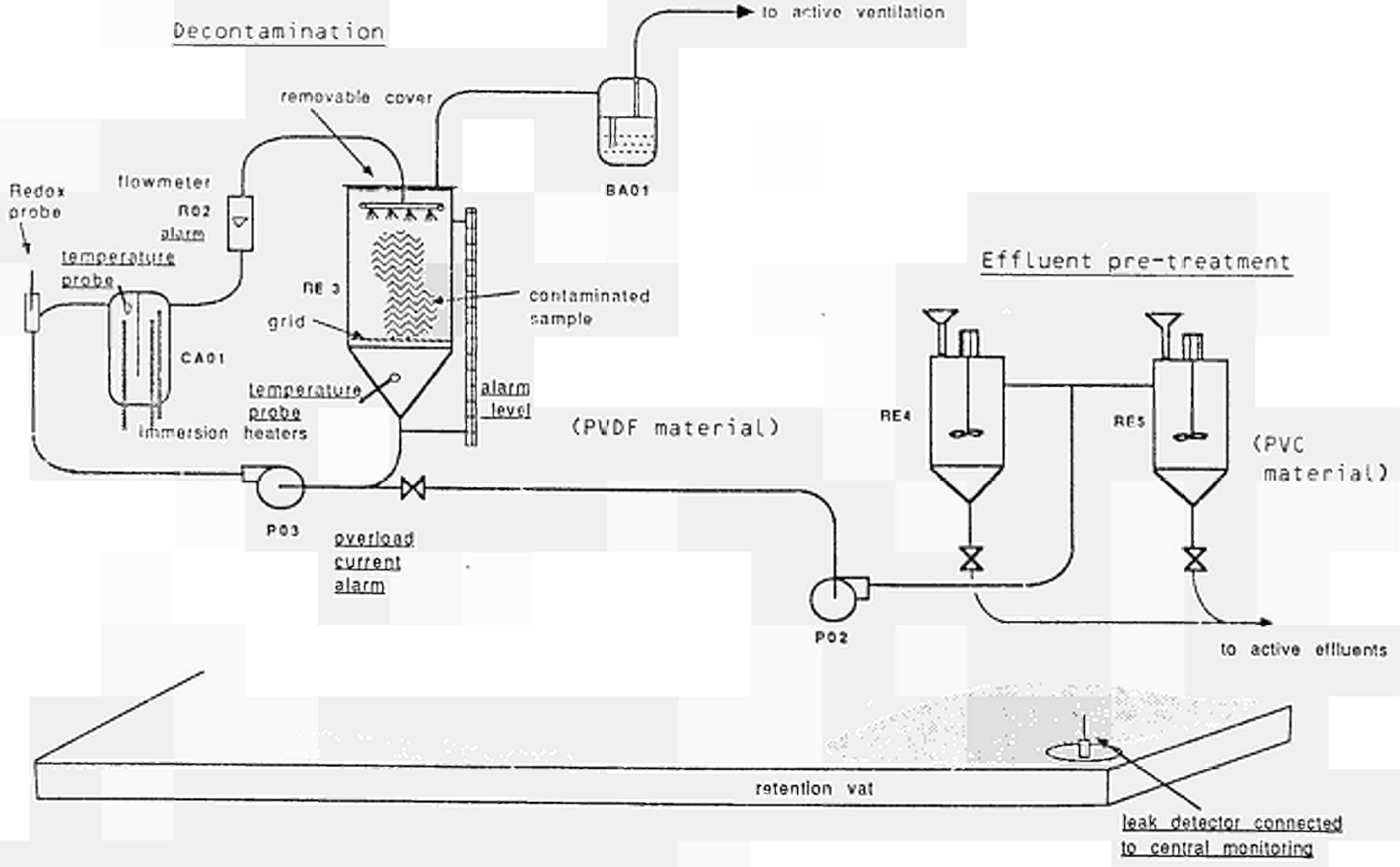


Figure 1: General configuration of GROLABO facility.

8.12 Automated Measuring System for Waste from Dismantling of the KKN Plant, to be Released

Contractor: Nuklear-Ingenieur-Service GmbH, Hanau
Contract N°: FI1D-0062
Working Period: May 1986 - December 1988
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

After preplanning the measuring chamber was specified and manufacturers were called for bids. Münchner Apparatebau (MAB), Munich got the order for the manufacturing. First tests at MAB with a smaller device but with similar detector units showed, that the specified minimum detectable activity (3700 Bq) of the components to be measured can be met with the planned measuring chamber. Documents for preliminary examinations were submitted to the licensing authority.

Progress and Results

1. Preparation of design (B.2.)

1.1 Conception of measuring device

The ordered measuring device is specified as follows:
The measuring chamber consists of the detector system, a data acquisition and control unit and a supporting rack. The detector system has twelve detector cases filled with a liquid scintillator. Each detector case has a sensitive area of 0.3 m^2. The 12 cases are arranged on the outer surface of the measuring tunnel of the supporting rack, so that a large portion of 4 π-geometry will be gained (Figure 1).

Each detector has its own measuring chain, which transmits the counts to the data acquisition unit (Figure 2). This unit preprocesses the counts related to the earlier calibration and stores the preprocessed counts into a microprocessor memory.

Further data processing will be done with a personal computer bidirectional connected with the data acquisition unit. This computer offers a great sophistication and flexibility of programming for the complex data analysis.

Besides the measuring chamber the automated measuring device is composed of an outer shielding, a conveyor track and a data processing unit. The outer shielding reduces the zero counting rate inside the detectors. A scale for the components is integrated in the conveyor track. The conveyor is conducted through the measuring chamber.

1.2 Basis of layout

Basis for the layout of the measuring chamber is the minimum activity (3700 Bq) of a component or a batch of material to be measured which was derived from limit values of the KKN-licence.

These limit values for unrestricted release are 0.37 Bq/g and 0.37 Bq/cm^2. With these limits the minimum activity of 3 700 Bq is associated to a reference mass of 10 kg and a reference surface of 1 m^2. Due to different surface/mass-ratios of components real minimal masses to be measured can go from 8 to 167 kg.

For considering different nuclides and complex geometries of the components to be measured the minimal detectable activity (MDA) of the measuring chamber has been estimated and finally defined to 300 Bq of Cobalt 60.

The potentially free components to be measured with the measuring chamber (clearance measurements) will be presorted taking into account origin, knowledge of operation history and the results of activity measurements on material samples (Figure 3).

1.3 Estimations

For the analysis of measuring data different parameters and effects have to be estimated in advance. Therefore groups of components with similar geometry, surface/mass-ratio and self-shielding effect were created. For these different groups guiding principles for measuring configurations (maximum/minimum weight, number of parts and layers, height of the batch) were developed. Due to

- limiting wall and layer thickness
- hard gamma radiation of the reference nuclide Co 60
- dose build-up of the steel walls and layers

the influence of self-shielding of covered contaminations will be low (> 0,7).

The outer shielding of the measuring chamber will be in the range of 2.5 cm to 5 cm. This corresponds to a reduction of radiation of Co 60 lateral to the chamber by a factor of 0.3 to 0.1. The reduction of zero counting rate will be even higher.

Measurements of the manufacturer with a smaller prototype device with similar detector units confirmed that the specified minimal detectable activity can be gained with high degree of confidence.

2. Provision of equipment (B.2.)

Specialised manufacturers were called for bids for the measuring chamber. Within the frame work of preliminary specifications a broad variation of conceptions was presented. With the bidders of the two most suitable bids detailed placing discussions were accomplished. Finally MAB, Munich got the order to manufacture the specified measuring chamber.

The inferface and data exchange mode between data acquisition unit and personal computer was specified. The conveyor track, the transport and measuring bins and the shielding of the measuring chamber were specified and suppliers were called for bids.

3. Licensing procedures (B.3.)

In July 1987, the licensing authority made a positive decision for immediate start of KKN dismantling after objection by nuclear opponents. Consequently, this site is now available for the large scale assessment of the measuring device. Also the measuring device itself must 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.

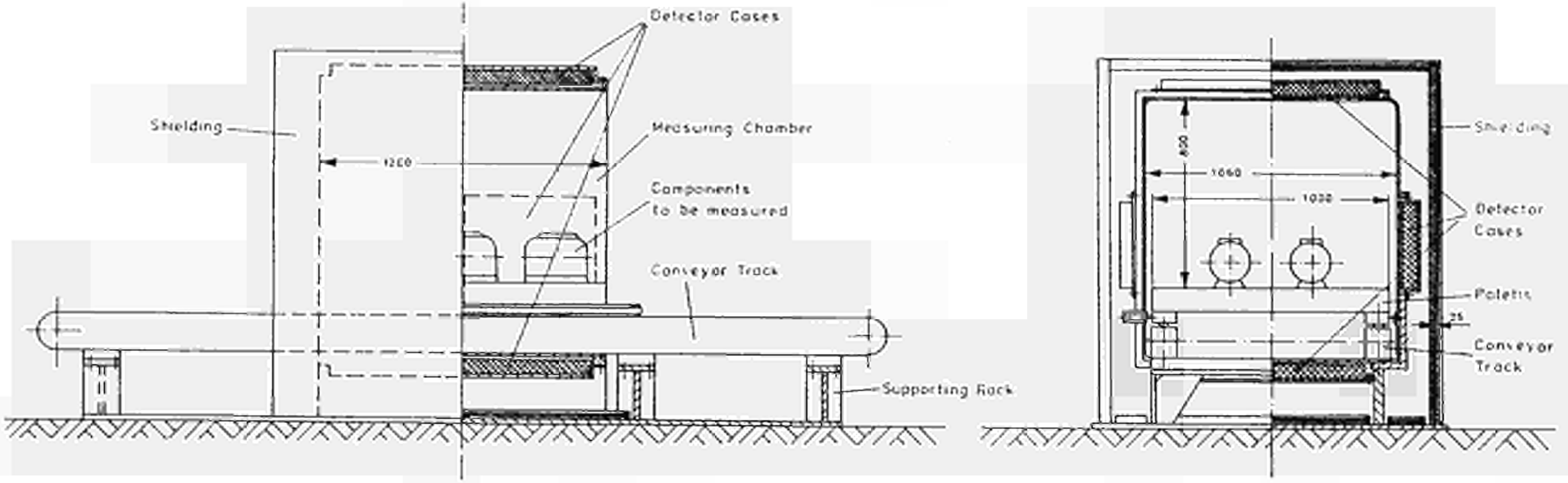


Figure 1 : Preliminary Layout of Measuring Device

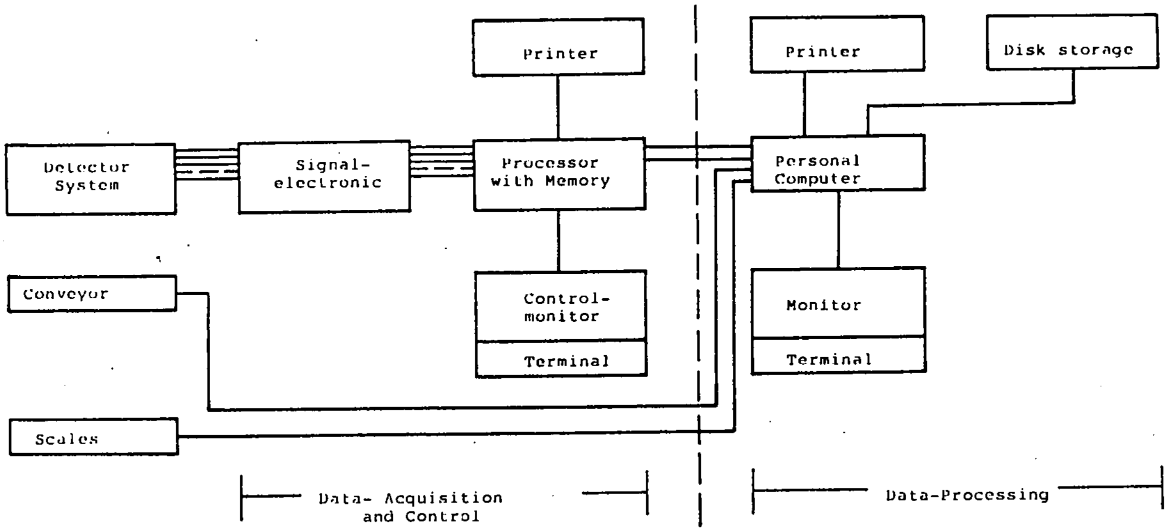


Figure 2: Configuration of Data-Acquisition and -Processing

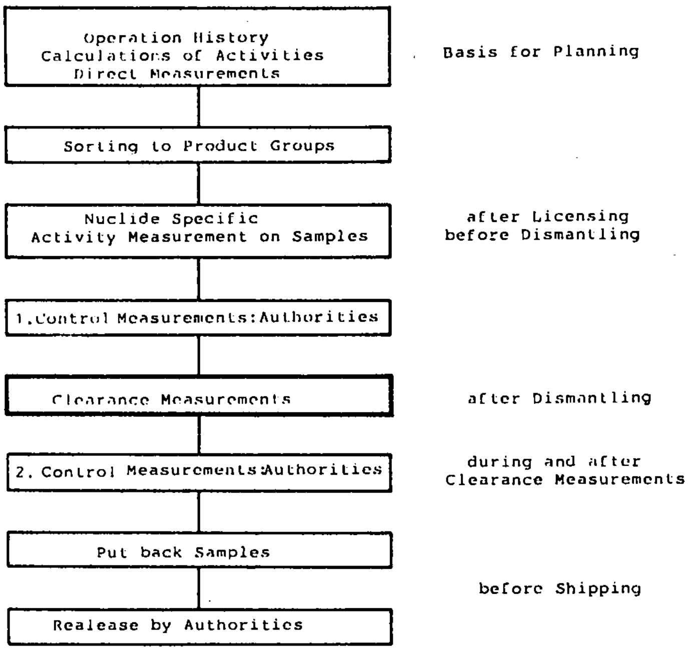


Figure 3: Release procedure KKN-Dismantling

ANNEX I

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

A. Annual Progress Reports

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

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

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

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

B. 1984 European Conference

Schaller, K.H., Huber, B. (ed). Decommissioning of nuclear power plants - Proceedings of a European Conference held in Luxembourg, 22-24 May 1984. Graham & Trotman Ltd, London. EUR 8655.

C. Final contract reports

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

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

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

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White, I.F. et al. (1984). Assessment of management modes for graphite from reactor decommissioning. EUR 9232.

Goddard, A.J.H. et al. (1984). Trace element assessment of low-alloy and stainless steels with reference to gamma activity. EUR 9264.

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Hulot, M. et al. (1986). State-of-the-art review on technology for measuring and controlling very low-level radioactivity in relation to the decommissioning of nuclear power plants. EUR 10643.

ANNEX II

LIST OF PUBLICATIONS RELATING TO THE RESULTS OF THE 1984-88 PROGRAMME ON THE DECOMMISSIONING OF NUCLEAR INSTALLATIONS

A. Annual Progress Reports

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - First Annual Progress Report (year 1985)", EUR 10740, 1986.

"The Community's Research and Development Programme on Decommissioning of Nuclear Installations - Second Annual Progress Report (year 1986)", EUR 11112, 1987.

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.

ANNEX III

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European Communities — Commission

**EUR 11715 — The Community's research and development programme on decommissioning of nuclear installations
Third annual progress report 1987**

Luxembourg: Office for Official Publications of the European Communities

1988 — VIII, 311 pp., 56 tab., 86 fig., 22 ph. — 21.0 × 29.7 cm

Nuclear science and technology series

EN

ISBN 92-825-8899-8

Catalogue number: CD-NA-11715-EN-C

Price (excluding VAT) in Luxembourg: ECU 25

This is the third annual progress report of the European Community's programme (1984-88) of research on the decommissioning of nuclear installations. It shows the status of the programme on 31 December 1987.

The third progress report describes the objectives, scope and work programme of the 69 research contracts concluded, as well as the progress of work achieved and the results obtained in 1987.

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