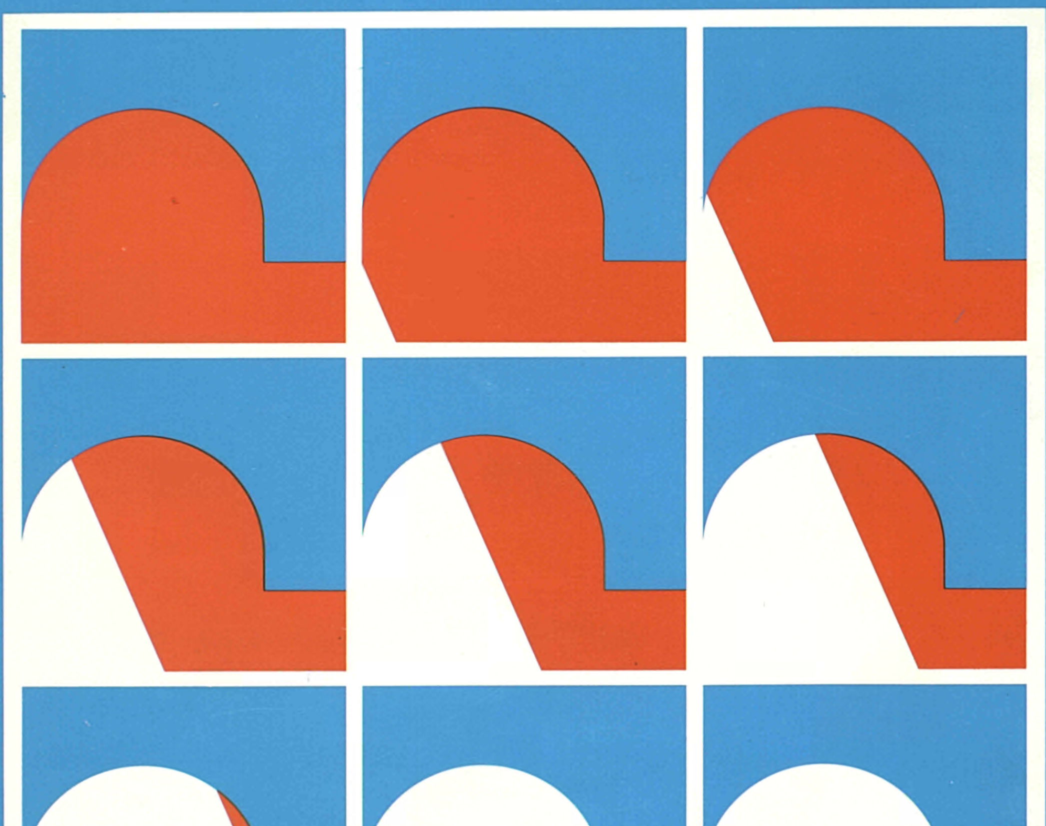
Commission of the European Communities

nuclear science and technology

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

Fourth annual progress report (year 1983)



Report

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Fourth annual progress report (year 1983)

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FOREWORD

This is the fourth progress report of the European Community's programme (1979-1983) of research on the decommissioning of nuclear power plants. It covers the year 1983 and follows the 1980, 1981 and 1982 Reports (Ref. 1, 2, 3).

The Council of the European Communities had adopted the programme in March 1979 (Ref. 4), considering:

> "Certain parts of the nuclear power plants 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 plants."

The programme sought to promote a number of research and development projects as well as the identification of guiding principles. The projects concern the following subjects:

- Project N° 1: Long-term integrity of buildings 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 transport containers for radioactive waste produced in the dismantling of nuclear power plants;
- Project N° 6: Estimation of the quantities of radioactive waste arising from decommissioning of nuclear power plants in the Community;
- Project N° 7: Influence of nuclear power plant design features on decommissioning.

The research was carried out by public organizations and private firms in the Community under cost-sharing contracts with the Commission of the European Communities. The Commission budget for this five-year programme amounted to 4.7 million ECU.

The Commission has been responsible for managing the programme and was assisted in this task by an Advisory Committee on Programme Management, which consisted of experts appointed by the Member States' governments and of Commission officials.*

The 1980, 1981 and 1982 Reports described the work programmes of most research contracts and initial results of the research.

The present report describes the further progress of research and contains a large amount of results. For a majority of the 51 research contracts composing the 1979-1983 programme, work was completed by the end of 1983; the conclusions drawn from this work are in this report.

Readers wishing complementary information are referred to the final contract reports, a number of which have already been published (see References at the end of the present report), and to the Proceedings of the International Conference on the Decommissioning of Nuclear Power Plants that the Commission of the European Communities organized on 22-24 May 1984, in Luxembourg.

Further research and development work is to be carried out under the new five-year (1984-1988) programme on the decommissioning of nuclear installations.

The Commission staff in charge of the programme during 1983 and of editing this report were: B. Huber, K.H. Schaller, R. Bisci and K. Pflugrad. Finally, the Commission wishes to express its gratitude to all the scientists of the contractors who have contributed to this report.

B. Huber
Head of the Programme

S. Orlowski
Head, Division
"Nuclear Fuel Cycle"

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* See Annex

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

It has been proposed that the dismantling of nuclear power plants be delayed for periods ranging from several decades to about a hundred years. Thereupon, radiation having largely died away, the dismantling would be easier and the radiation exposure of the dismantling workers would be less.

In this connection, measures are being studied to determine which ones are necessary to maintain retired plants in a safe condition over long periods. Particular attention is paid to the integrity of buildings and systems which contain the radioactive material (e.g. reactor building, reactor cooling system).

1.1. Degradation of Building Plant and Materials

Contractor: Central Electricity Generating Board, Barnwood, United Kingdom
Contract N°: DE-A-001-UK Work Period: April 1980 - December 1983

1.1.1. Objective and Scope

The objective of this research is to establish the life cycle of existing nuclear power station buildings and ensure that specifications for new station buildings list materials that are suitable for a long life with minimum maintenance. Wherever possible, the research should aim at ensuring that the specified materials attract surface contamination only or induced activity which decays rapidly.

Long life maintenance treatment for retained plant and buildings for safety and security purposes will be researched, to enable future maintenance and surveillance to be kept within reasonable economic limits. The types of nuclear power plants concerned by this research are Magnox reactors and Advanced Gas-cooled Reactors.

1.1.2. Work Programme: See Ref.1, Paragraph 1.1.2.

1.1.3. Progress and Results

Sampling and in situ testing of materials in N° 1 reactor building at Trawsfynydd power station took place during January. The samples taken were subsequently tested by the specialist testing contractor who produced a report in May. The condition of the building was considered to be good for its age and unlikely to suffer from any major deterioration for at least 50 years assuming adequate maintenance is provided.

In-situ tests carried out were ultrasonic pulse velocity tests on concrete, covermeter surveys on reinforced concrete, carbonation tests on concrete, inspection of steelwork for corrosion and fatigue cracks, inspection of precast concrete panel fixings for corrosion, inspection of metal window frames for corrosion.

Samples removed for testing comprised diamond-drilled concrete cores, mastic asphalt samples from roofs, fixing bolts from asbestos cement cladding, retaining bolts from patent glazing, samples of dry pack mortar bedding from precast concrete panels and a ground-water sample.

Laboratory testing of concrete cores included density measurements, compressive strength measurement, ultrasonic pulse velocity testing, aggregate/cement ratio measurement and chloride ion measurement. Mastic asphalt samples were tested for hardness, soluble binder content and aggregate grading. Fixing bolts were examined microscopically for corrosion. The ground-water sample was analysed for sulphate content.

Following an inspection of Oldbury power station reactor buildings in January, detail proposals for the locations and type of samples and the nature of in-situ and non-destructive tests were made by the specialist testing contractor and were subsequently approved. The sampling and in-situ testing of the materials in N° 1 reactor building were successfully completed in May. The samples were tested during June and the contractor's report was issued in August. The reactor building was found to be in good condition and, if given adequate maintenance, unlikely to suffer from any major deterioration for the next 50 years.

At Oldbury, the in-situ tests carried out were ultrasonic pulse velocity tests on concrete, covermeter surveys on reinforced concrete, carbonation tests on concrete, hardness tests on concrete, inspection of structural steelwork for corrosion and fatigue cracks, internal and external examinations of steel cladding panels for deterioration and inspection of metal window frames for corrosion.

Samples removed for testing comprised diamond-drilled concrete cores, mastic asphalt samples from roofs and ground-water samples.

Laboratory testing of concrete cores included density measurement, compressive strength measurement, ultrasonic pulse velocity testing, aggregate/cement ratio measurement and chloride ion measurement. Mastic asphalt samples were tested for hardness, soluble binder content and aggregate grading. The ground-water samples were analysed for pH value and sulphate content.

An inspection of the reactor buildings at Dungeness 'A' power station was made in April, following which detailed proposals for locations and types of samples and the nature of in situ non destructive tests were subsequently approved and sampling and in-situ testing in N° 2 reactor building were carried out during September. Laboratory testing was completed in October and a report produced by the specialist testing contractor. The condition of the building was considered to be satisfactory for its age and unlikely to suffer from any major deterioration for at least 25 years, on the assumption that adequate maintenance will be provided.

In-situ tests comprised ultrasonic pulse velocity tests on concrete, covermeter surveys on reinforced concrete, carbonation tests on concrete, examination of steel cladding panels for deterioration, examination of patent glazing for corrosion and inspection of internal brickwork, door lintels for cracking.

Samples removed for testing were diamond-drilled concrete cores and mastic asphalt samples from roofs.

Laboratory testing of concrete cores included density measurement, compressive strength measurement, ultrasonic pulse velocity testing, aggregate/cement ratio measurement and chloride ion measurement. Mastic asphalt samples were tested for hardness, soluble binder content and aggregate grading.

1.2. Long-term Integrity of Buildings and Systems

Contractor: Commissariat à l'Energie Atomique, Etablissement de la Vallée du Rhône, France

Contract N°: DE-A-002-F   Work Period: January 1981 - March 1984

1.2.1. Objective and Scope

The aim of this study is to improve the knowledge of the aging of nuclear plant and to propose preventive measures for maintaining such plant in a satisfactory condition. The results should make it possible to choose the best decommissioning strategy (deferred or prompt dismantling) and to provide recommendations for the design of new plants.

It is planned to prepare a methodology document to be used for any nuclear power plant of which decommissioning is under consideration. This document has to deal with:

- identification of the plant components the safety barriers rely on, in case of Stage 1 (maintenance of the plant in a satisfactory condition) or Stage 2 (long-term containment) of decommissioning;
- causes of aging and damaging effects, due to aggressive agents to be considered depending on the nature of the components (concrete, steel...);
- measures to prevent or to cure these effects in order to maintain the plant in safe conditions at Stage 1 or Stage 2.

The results expected from this analysis are:

- elements for estimating the maintenance and surveillance cost;
- elements for choosing the decommissioning stage and the delay suitable to achieve Stage 3 (complete removal);
- information for the design of new plants, facilitating their future decommissioning.

1.2.2. Work Programme: See Ref.1, Paragraph 1.2.2.

1.2.3. Progress and Results

The reported work relates to a 900 MWe PWR plant as a reference.

1.2.3.1. Functions to be kept operational

This subject was completed with the study of the confinement and maintenance of contaminated circuits and with the cost estimate of the various decommissioning steps.

An intermediate decommissioning stage specific for the reference reactor, called "Reinforced Stage 1", is being proposed. The operations necessary for this stage and its consequences were described, e.g.:

- circuit draining, setting of barriers between circuits and drying to minimise corrosion effects;
- radiological and thermal consequences of draining; in particular removal of the decay heat from the pressure vessel and core structure activation;
- periodical control and preventive maintenance of circuits and their main components;
- estimation of cost for each decommissioning stage and further yearly costs, except for waste treatment and disposal.

1.2.3.2. Radioactivity inventory

The PWR primary circuit radioactivity inventory was completed with the activity of the pressure vessel and the core structures (mainly cobalt- 60) and the calculation of the dose rate and its evolution with time.

1.2.3.3. Relationships between postponement times and their consequences

The factors to be considered in selecting the decommissioning stage and waiting times at various stages have been listed. These factors were broken down into:

- aging or damage of the barriers or service equipment and components;
- radioactivity decay curve;
- thermal aspects related to the removal of the decay heat from the vessel;
- environmental safety aspects;
- evolution in licensing procedures;
- development and experience of new techniques;
- specific aspects of other reactor types;
- demand for reutilization of reactor site;
- existance of one or several reactors in operation on the site;
- availability of waste storage

and, of course, the specific or general economic aspects.

Some of these factors have been explained by curves pointing out the dates or periods favourable for selection.

1.2.3.4. Specific aspects of other reactor types

The study has been extended to other types of power reactors in operation within the European Community, avoiding uninteresting and long repetitions. In particular, pertinent details concerning components and functions to be maintained in operation after shutdown have been given.

2. PROJECT N° 2: DECONTAMINATION FOR DECOMMISSIONING PURPOSES

In most of the radioactive parts of a nuclear power plant to be decommissioned, the radioactivity is present exclusively as surface contamination. Decontamination is aimed at simplifying the dismantling of these parts or reducing the arisings of the radioactive waste.

The following decontamination techniques are being assessed and developed:

- techniques using chemically aggressive liquid and gel-like decontaminants;
- electrochemical techniques;
- hydromechanical techniques (high-pressure water lance, erosion by cavitation);
- decontamination of concrete walls by flame-scarfing.

Moreover, the composition and structure of the contamination layers, which are formed over a long operation period in the cooling circuit of light water reactors, are being analysed.

The special decontamination problems which may arise after a reactor accident with loss of coolant are also being considered.

2.1. Decontamination of Concrete Surfaces by Flame-scarfing

Contractor: Salzgitter A.G., Salzgitter, Germany

Contract N°: DE-B-002-D Work Period: October 1980 - December 1982

2.1.1. Objective and Scope

The flame-scarfing technique involves the use of a torch, the heat of which causes a thin concrete surface layer to peel off and burns up paint. The purpose of the research is to investigate the efficiency and limitations of flame-scarfing for decontamination by testing the technique on non-contaminated and contaminated concrete surfaces with and without paint. The investigation should give information on fire hazard, aerosol formation and filtering, radiation protection of the workers, feasibility of directly exhausting the combustion products to prevent recontamination, and magnitude of the decontamination factor.

2.1.2. Work Programme: See Ref. 1, Paragraph 2.2.2.

2.1.3. Progress and Results

The research was completed in December 1982 and the final report has been published (Ref. 5).

2.1.4. Conclusions

Initial fact-finding tests, using a flat-section jet torch on inactive coated and uncoated specimen slabs, resulted in the following optimum parameters of flame-scarfing:

- gas pressure C_2H_2: $0.5 \text{ bar} < p < 1 \text{ bar}$
- gas pressure O_2: $6 \text{ bar} < p < 8 \text{ bar}$
- distance between torch and concrete surface: 5 mm
- angle between torch and concrete surface: 30°
- feed velocity for uncoated surfaces: 1 m/min
- feed velocity for coated surfaces: 0.5 m/min

Removal tests on inactive specimen slabs have shown that removal depths of 2.5 to 3 mm could be reached in the case of coarse-grained concrete and removal depths of 1.4 to 1.7 mm in the case of fine-grained concrete.

For the precipitation of the dust and aerosols arising from flame-scarfing, a sucking unit with three filtering steps connected in series was used. The first filter stage consisted of a bulk material barrel filter. As intermediate stage, a box-type filter, containing a pre-filter (type Eu 7, DIN 24 185) and a HEPA filter element (type S, DIN 24 184), was selected. A HEPA barrel filter was used for superfine dust separation. Overall filter efficiencies between 99.991% and 99.999% could be obtained.

Tests on contaminated concrete surfaces were carried out in the solid waste store of the closed-down nuclear power plant Gundremmingen KRB-A. The contamination mainly consisted of ^{137}Cs (90%) and ^{60}Co.

Concrete layers of 4 to 5 mm thickness had to be removed in order to reach the limit value allowing unrestricted further use. To do this, four flame-scarfing treatments were necessary.

The tests showed that during flame-scarfing the activity is concentrating in the slag. An increase of aerosol activity in the building with increasing height due to air movement was measured. The recontamination of the surfaces was negligible.

The research carried out has shown that flame-scarfing is a well-suited decontamination process for concrete surfaces. Further tests are needed to optimize the filter technique, especially to increase the dust-collecting capacity of the pre-filters.

2.2. Erosion of Metal Surfaces by Cavitation at Very High Velocity

Contractor: Alsthom-Atlantique Neyrtec, Grenoble, France

Contract N°: DE-B-010-F   Work Period: October 1982 - October 1983

(Follow-up to contract N° DE-B-003-F)

2.2.1. Objective and Scope

Cavitation is the formation of vapour cavities in zones of a flowing liquid where pressure is low due to high local velocity. Subsequently, the cavities collapse abruptly, and if this takes place near the surface of a solid, the latter may be eroded. In a first study (contract N° DE-B-003-F) the feasibility of a surface erosion technique employing a cavitation flow of very high velocity was shown, using cavitation elements with one-directional flow.

The present contract is aimed at the development of a new type of cavitation element, based on a circular geometry. This geometry eliminates the side-walls of the cavitation stream and, thereby, the problem of their erosion and the associated energy loss.

2.2.2. Work Programme: See Ref. 2, Paragraph 2.2.2.

2.2.3. Progress and Results

The work has been completed in summer 1983 and the final report has been published (Ref. 6).

In 1983, cavitating elements with four slightly different designs using vortex flow were tested with up-stream pressure of 56 bars. Table 1 gives the results, allowing comparison with the results obtained with a circular cavitating nozzle (Table 3 of Ref. 2).

The tests also allowed the evaluation of the erosion resistance of materials employed in nuclear industry. An "erosion coefficient" has been defined as the mass loss per unit energy spent in the process (Table 2).

2.2.4. Conclusions

The aim of this study was to assess the feasibility of using cavitation erosion to remove thin layers of material from contaminated surfaces

Table 1. Erosion tests with vortex cavitating devices

Device N°	Test N°	Specimen material	Stand-off distance (mm)	Downstream pressure (bar)	Exposure time (min)	Volume loss (mm^3)
1	13B	Aluminium	1	14	10	< 0.1
2	14	Aluminium	2	14	10	< 0.1
2	14B	"	1	14	10	< 0.1
3	15	Aluminium	1	14	10	1.5
3	16	"	1	5	10	< 0.1
3	17	"	2	14	10	1.5
3	18	"	3	14	10	1.5
3	19	"	5	14	10	1
4	20	Aluminium	1	14	10	60
4	21	"	3	14	10	10
4	22	Mild steel	1	14	600	0.3

Table 2. Erosion resistance of some common materials

Material	Density (mg/mm^3)	Erosion coefficient ($mg\ min^{-1}kW^{-1}$)	Relative resistance (basis : aluminium)
Aluminium	2.7	1.0	1
Plexiglass	1.2	1.1×10^{-1}	9
Duraluminium (4% Cu)	2.8	6.8×10^{-3}	150
Stainless steel	7.8	4.6×10^{-4}	2200
Mild steel (A 37-1)	7.7	3.9×10^{-4}	2600

such as those encountered in the dismantling of nuclear power plants. As compared with other types of mechanical erosion like high pressure jets and grit blasting, cavitation erosion has the advantage of not producing solid or liquid aerosols.

Various cavitating flow devices were designed and tested in a specially constructed test loop. Aluminium, stainless steel, plexiglass, paint and other materials were used as the target surface. Though the applied pressure did not exceed 60 bar, the most efficient cavitation devices tested achieved measurable erosion rates on hard materials like stainless steel.

Full-scale tests using a nozzle moving over the contaminated surface and a pressure of 300-600 bar may now be envisaged.

2.3. Composition of Contamination Layers and Efficiency of Decontamination

Contractor: Kernkraftwerk Lingen GmbH, Lingen, Germany

Contract N°: DE-B-004-D Work Period: January 1981 - October 1983

2.3.1. Objective and Scope

The purpose of this research is the characterization of, on the one hand, the contamination layers formed in long-term reactor operation and, on the other hand, of the residual contamination after the application of aggressive chemical techniques. These investigations are performed on samples taken from the primary circuit of the Lingen reactor (520 MWth Boiling Water Reactor, shut down in 1977 after nine years of operation).

2.3.2. Work Programme: See Ref. 1, Paragraph 2.4.2.

2.3.3. Progress and Results

The research work was completed and the final report is being prepared for publication (Ref. 7).

2.3.3.1. Characteristics of contamination layers

Following the determination of the oxide quantities and ^{60}Co activity, other nuclides were measured (Table 3).

2.3.3.2. Decontamination tests

In order to achieve complete decontamination of surfaces having been exposed to the primary coolant in water-cooled reactors, removal of the oxides is necessary before dissolving the upper layers of the base material by attack with strong solutions.

The LOMI procedure has been tested for oxide dissolution. When LOMI reagents (LOMI = Low-Oxidation state Metal Ions) are used to dissolve the oxides, the ion Fe^{3+} is reduced to the ion Fe^{2+} in the oxides. The lattice structure of the oxides is destroyed by this process leading subsequently

Table 3. Activity of oxide layers (Reference date 1.9.1983)
(Values are given in Bq/g of oxide and Bq/m^2 of pipe surface)

Nuclide	Half-life (a)	Steam Pipe		Condensate Pipe		Primary Water Pipe	
		(Bq/g)	(Bq/m^2)	(Bq/g)	(Bq/m^2)	(Bq/g)	(Bq/m^2)
^{60}Co	5.27	$2.4x10^6$	$1.9x10^7$	$1.5x10^7$	$1.6x10^8$	$7.9x10^8$	$3.6x10^{10}$
^{55}Fe	2.70	$9.5x10^3$	$7.6x10^4$	$3.8x10^4$	$3.9x10^5$	$3.9x10^6$	$1.8x10^8$
^{106}Ru	1.01	$3.2x10^2$	$2.5x10^3$	$3.9x10^2$	$4.1x10^3$	$1.2x10^5$	$5.6x10^6$
^{137}Cs	30.10	$1.4x10^2$	$1.1x10^3$	$1.4x10^3$	$1.5x10^4$	$3.7x10^4$	$1.7x10^6$
^{134}Cs	2.06			$2.8x10^1$	$2.9x10^2$		

to decomposition of the oxides. In the decontamination tests, the bivalent vanadium ion has been applied as a metal ion with a low oxidation level:

$$Fe^{3+} + V^{2+} \Rightarrow Fe^{2+} + V^{3+}$$

If the oxides contain larger quantities of trivalent chrome ions, then the lattice structure of the oxides is only partly destroyed, because chrome-III-ions, which are not reduced, stabilize the lattice structure. In order to destabilize these lattice places, too, the chrome-III-ion in the lattice can be converted into the soluble chrome-VI-ion by oxidation with potassium permanganate ($KMnO_4$).

The main test result is that complete removal of the oxide layers could not be achieved through the LOMI treatment alone. The oxide layer was not completely removed even after three LOMI treatments in steam-pipe sample N° 1. The remaining parts of the oxide layer, however, adhered only loosely to the base material. They could be removed readily by a light mechanical treatment. This behaviour of the oxide layers could be due to the relatively high Cr_2O_3 contents (15.7% - 18.4%).

The oxide layers were completely removed (condensate pipe sample N°1 and steam pipe sample N° 2) only through a subsequent oxidic treatment with potassium permanganate and a further after-treatment with oxalic acid. However, complete removal of the oxide layers was not achieved if the decontamination of the pipe samples was executed in the sequence oxidation treatment, LOMI-treatment (condensate pipe sample N° 2 and steam pipe sample N° 1).

Therefore, the decontamination of the primary purification pipe sample was carried out according to the treatment cycle LOMI, oxidation, and oxalic acid treatment. Contrarily to the condensate and steam pipe samples, three cycles were needed for complete removal of the oxide layer. This could be due to the thicker oxide layer of the primary purification pipe (45.8 g/m^2) as compared with the condensate pipe (10.8 g/m^2) and the steam pipe (8 g/m^2). The bulk of the activity was removed by the oxalic acid treatment and only a very small part of it was removed by the oxidation treatment.

2.3.3.3. Conclusions

Three pipe samples from the primary steam, primary condensate and primary water loops of the Lingen nuclear power station, were examined to determine the activity distribution in the base material (activity depth profile). The following processes are of importance for the presence of radioisotopes in the base material:

- deposit of radioactive corrosion products by the coolant in the corrosion-damaged locations of the base material;
- grain boundary diffusion;
- very low-level activation of the base material.

As the result of the studies on decontamination for unrestricted release, the following procedure is recommended:

- removal of the oxide layer by the LOMI procedure,
- removal of a (base) material layer of determined thickness using a very diluted mixture of nitric acid and hydrochloric acid.

2.4. Vigorous Decontamination Tests of Steel Samples in a Special Test Loop

Contractor: Ente Nazionale per l'Energia Elettrica, Rome, Italy

Contract N°: DE-B-005-I Work Period: October 1980 - September 1983

2.4.1. Objective and Scope

The aim of this research is to identify, develop and assess vigorous decontamination techniques. The work involves the design and construction of a special test loop (DECO loop) and experiments using contaminated samples from reactors, in particular from the Garigliano reactor, a 160 MWe BWR shut down in 1978 after 15 years of operation. The decontamination efficiency and other important aspects such as treatment of the spent decontaminant, reliability, operational radiation exposure and costs are investigated. Eventually, a system for in-situ decontamination of a reactor component will be designed.

2.4.2. Work Programme: See Ref.1, Paragraph 2.5.2.

2.4.3. Progress and Results

The work has been completed and the final report is being prepared for publication (Ref. 8).

2.4.3.1. Oxide layer investigations

The metallographic and microanalytical characterization of the contaminated oxide layer on AISI 304 stainless steel of the Garigliano BWR plant has been concluded. The conclusions are summarized below.

The morphology and the thickness of the oxide were found to differ markedly over the surface of the examined component part, but the chemical composition was found to be quite uniform. The oxide corresponds very closely to the nickel-ferrite spinel $NiO-Fe_2O_3$, where the Ni and Fe atoms are partly substituted by Mn, Cu and Cr atoms. In particular, the Cu distribution across the oxide can be related to the history of the coolant water, which in turn is strongly influenced by metal composition of the coolant and feed water loops.

Apart from the variations of the minor components, it cannot be stated that a double oxide layer exists; in particular, the inner corrosion layer which in other nuclear reactors was found generally to be rich in Cr, has not been detected.

2.4.3.2. Decontamination tests

The decontamination tests have been pursued both in static and in dynamic conditions. About sixty tests were performed in total.

The DECO loop runs carried out in 1983 on AISI 304 specimens coming from the Garigliano BWR are summarized in Table 4. The evolution of decontamination, corrosion potential and corrosion rate with time, in the runs N° 6 to 8 are shown in Figures 1 and 2.

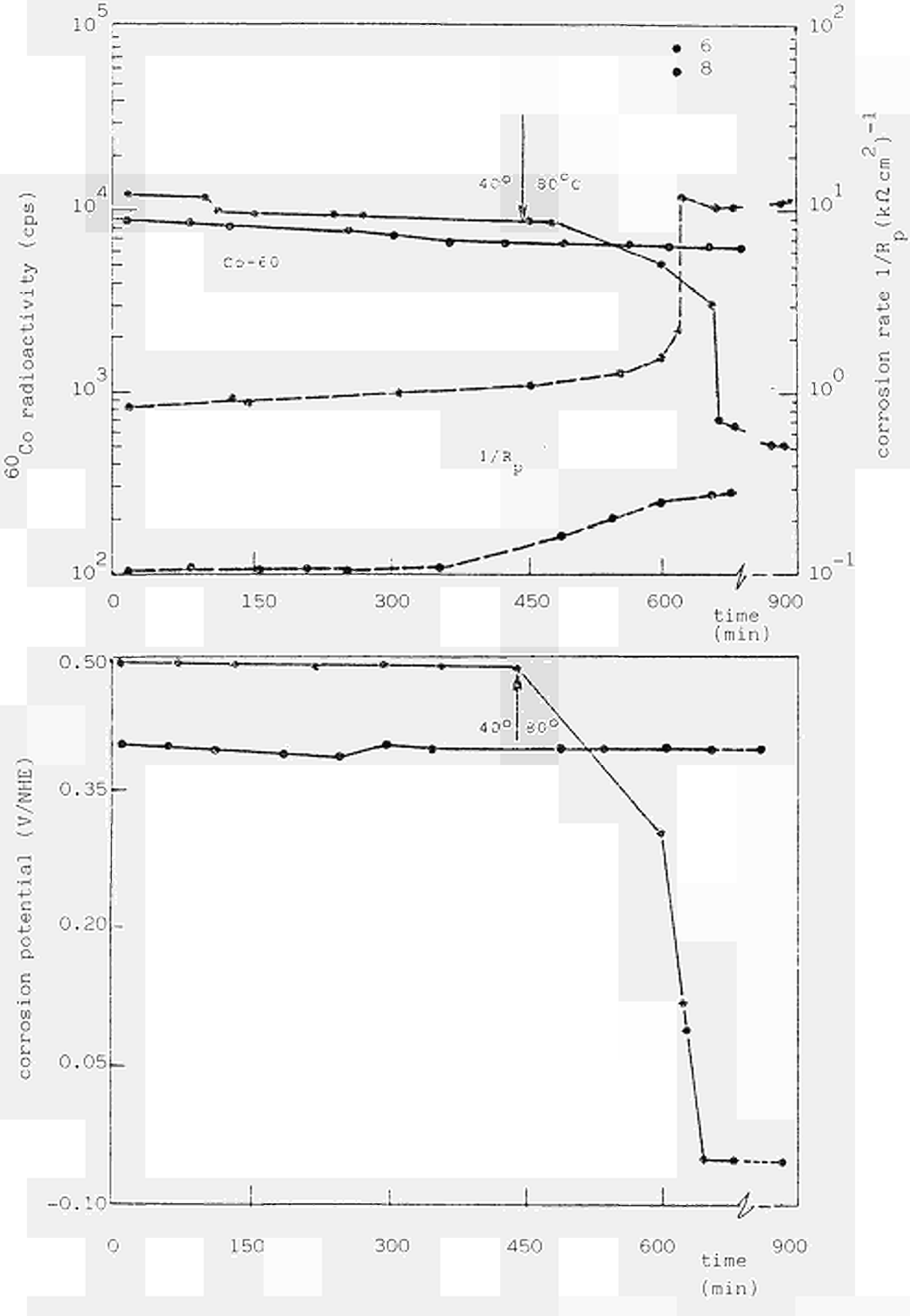

Figure 1. Evolution of decontamination, corrosion potential and corrosion rate, with time, in the DECO loop runs N° 6 and 8.

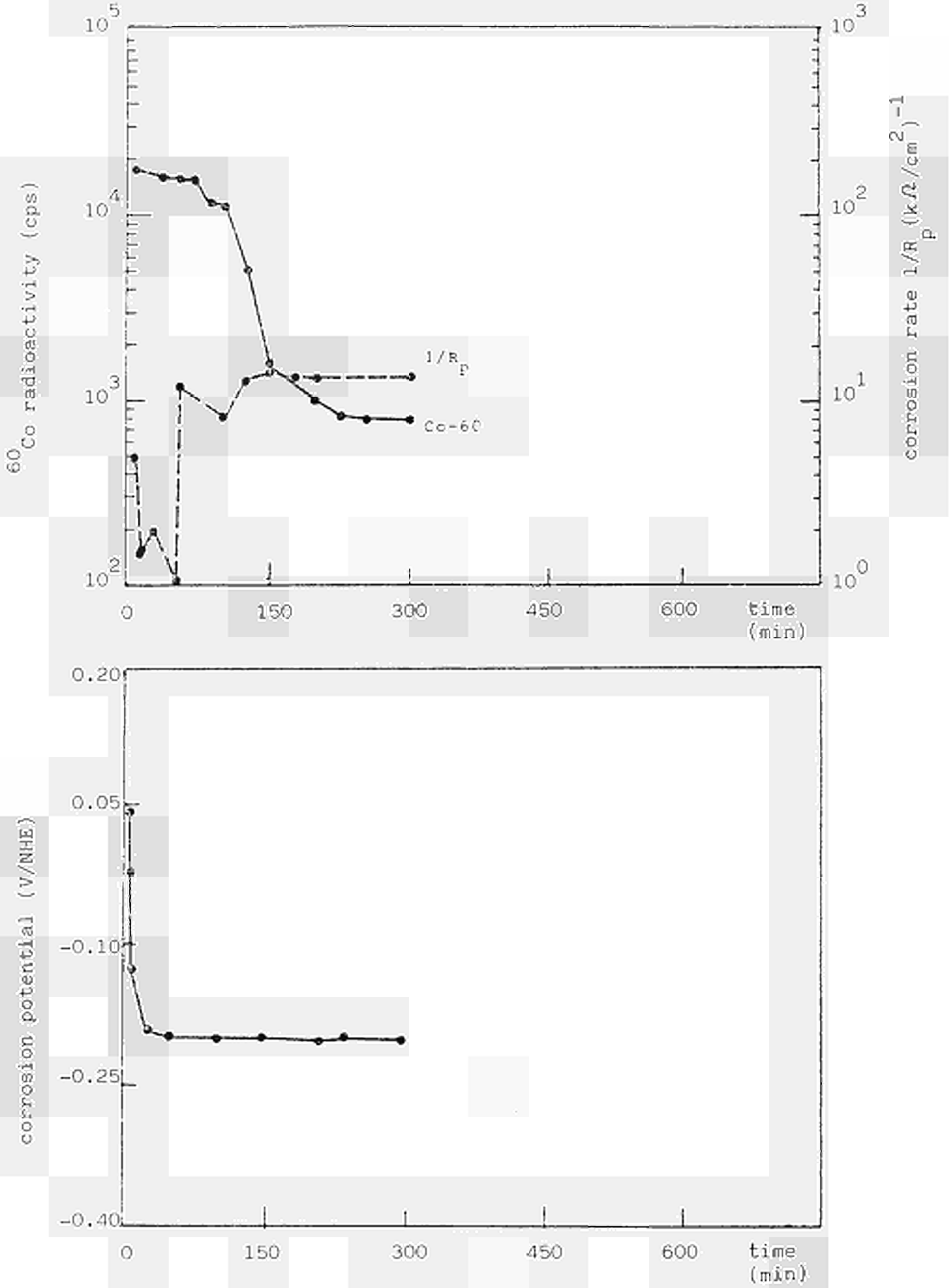

Figure 2. Evolution of decontamination, corrosion potential and corrosion rate, with time, in the DECO loop run N° 7.

Table 4. Conditions and results of dynamic decontamination tests

Test N°	6	7	8	9	10	11	12*
Decontaminant (vol. %)	0.35HF $+5HNO_3$	4.1HCl	0.35HF $+5HNO_3$	0.85HCl	4HCl $+2.5HNO_3$	0.75HF $+2.5HNO_3$	0.75HF $+5HNO_3$
Temperature (°C)	40-80	40	40	40	40	40-80	42
Test time (h)	9+5	4.83	12.50	11.75	2.10	5.5+2	3.35
Flow-rate (m/s)	1	4	4	4	4	4	1-4
Decontamination factor	3670	2610	1.1	2.5	5750	2360	839

* With oxygen controlled at less than 50 ppb.

2.4.3.3. Basic considerations about vigorous decontamination

Basic phenomena concerning the corrosion/dissolution of AISI 304 stainless steel were investigated in laboratory tests which have been extensively described in Ref. 9. Many potentiodynamic polarization curves both with stainless steel and platinum electrodes in the examined solutions have been recorded. These curves allowed to explain the strange behaviour of the 4 vol.% HCl + 13 vol.% HNO_3 solution, for which the potential stagnates at typical oxide values and dissolution does not occur.

2.4.3.4. Investigations on decontaminated materials

The characterization of the decontaminated specimens was aimed both at determining the final contamination level and evaluating the condition of the decontaminated surface.

Metal layers were removed and counted, beginning from the face opposed to the contaminated one. The ^{60}Co radioactivity profile across the metal and the metal oxide interface is given in Figure 3. The data show that the bulk of the metal is not activated and the final surface contamination levels are within the limits for unrestricted release.

Though the metal surfaces are generally very clean and shiny after vigorous decontamination, some differences according to the nature of the solutions are found. In particular, pits appear to be a distinctive feature of the specimens treated with the HF + HNO_3 solutions. No intergranular attack due to the decontamination was detected.

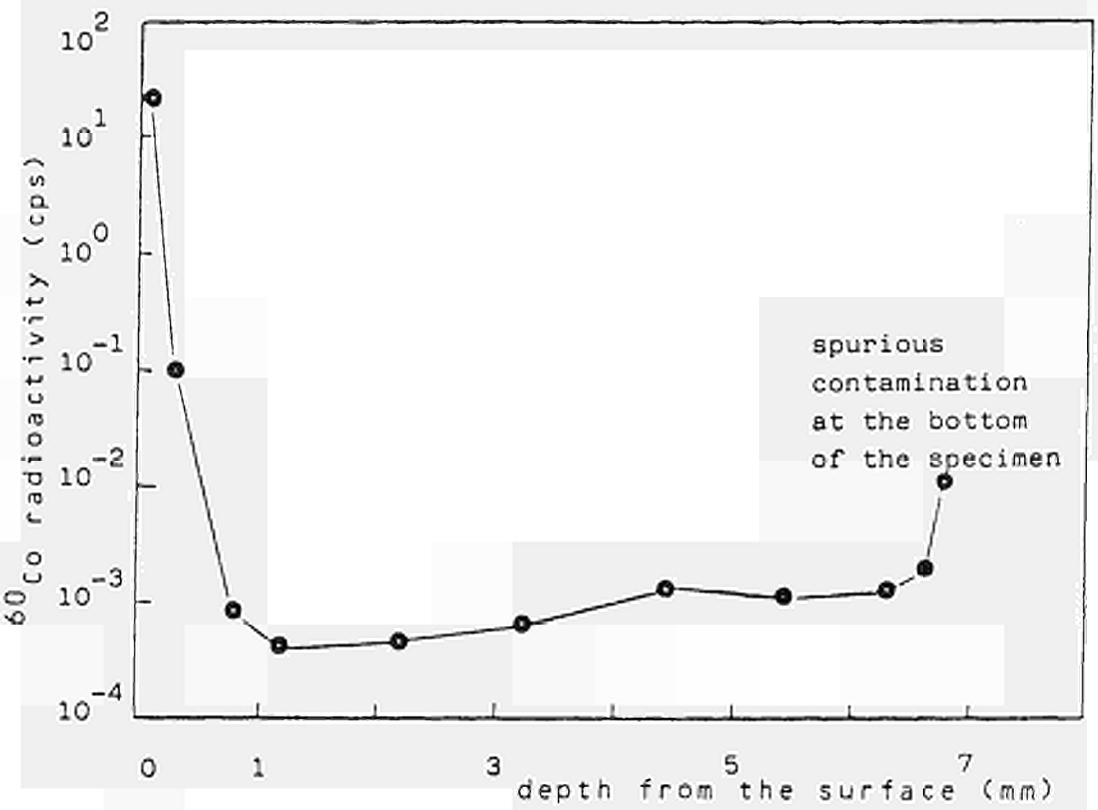

Figure 3. ^{60}Co radioactivity across the metal and the metal/oxide interface

2.4.4. Conclusions

The following treatments are considered to be most efficient and to achieve the total removal of the oxide and very high decontamination factors:

- 4-5 vol.% HCl, at 30°C in 5-6 h;
- 0.7-1 vol.% HCl, at 80°C in about 4 h;
- 1.5 vol.% HF plus 5 vol.% HNO_3, at 30°C in 4-6 h;
- 0.3-0.5 vol.% HF plus 2.5-5 vol.% HNO_3, at 80°C in about 5 h.

High flow rates are not necessary, but a good re-circulation of the solution is needed.

The treatment of the secondary waste has been quantified. Decontamination of a 10 m^2 surface would typically produce 0.5 to 3.0 kg of dry waste, corresponding to 1.6 to 10 kg of concrete-conditioned waste.

Finally, the preliminary design for vigorous decontamination of the regenerative heat exchanger of the Garigliano plant has been completed.

2.5. Development of Economical Decontamination Procedures

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen, Germany

Contract N°: DE-B-006-D   Work Period: October 1980 - December 1983

2.5.1. Objective and Scope

The main objective of this research is to identify, develop and assess the most suitable decontamination procedures for decommissioning purposes with particular reference to BWRs. The work involves:

- supplementary contamination measurements, in addition to existing ones, in the KRB-A nuclear power plant (a 237 MWe BWR shutdown in 1977 after 11 years of operation);
- decontamination experiments on samples taken from different areas of the primary circuit of the KRB-A reactor;
- evaluation of the experimental results in respect of both the KRB-A reactor and a 1200 MWe BWR.

2.5.2. Work Programme: See Ref. 1, Paragraph 2.6.2.

2.5.3. Progress and Results

Following the development of a suitable decontamination method based essentially on electropolishing with a 70% phosphoric acid solution, a large-scale application test was performed at the KRB-A Gundremmingen Power Plant (Ref. 10).

In order to reduce the restricted area of the plant before safe enclosure, dismantling of the large quantities of slightly contaminated components in the machine house would be desirable. As the radioactive material arising after decontamination is to be stored in the reactor building, this material should not exceed 10% of the initial quantity. As a demonstration project, 100 t of selected parts from the primary steam, condensate and feed-water loops were treated. The selected parts were insofar representative for about 75% of the total quantity, as their geometry allows measurements for proving that the unrestricted release limit (0.37 Bq/cm^2 in this case) is observed. About 25% of the material consists of a small diameter tubing, irregularly shaped components, etc., for which an accepted contamination measurement procedure does not yet exist.

The inner-surface contamination of water- or steam-ducting components ranged between 10 and 1000 Bq/cm^2; surfaces in contact with other contaminated media, like oil or air, showed contamination levels of less than 10 Bq/cm^2. In the demonstration test, 65 t of material were treated by the electrolytical method; for 35 t of material with loosely adhering contamination, current methods like steam cleaning were sufficient to reach the limit for unrestricted release.

The secondary waste from regeneration and treatment of the phosphoric acid was less than 2 wt% of the electrolytically treated parts. Adding small parts, which are not decontaminable economically, and other slightly contaminated items like filters, clothing, plastic covers, after volume reduction by incineration and compression, 8.5 t of low-level radioactive waste were left over for an initial quantity of 100 t.

Extrapolating this result cautiously to the 2000 t inventory of the machine house, the radioactive waste resulting from decontamination would be less than 200 t, not needing any shielding for conditioning and transport.

2.6. Development of Gel-based Decontaminants

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France

Contract N°: DE-B-007-F   Work Period: February 1981 - September 1982

2.6.1. Objective and Scope

Existing decontamination techniques employ usually abundant quantities of liquid decontaminant and, consequently, produce large volumes of radioactive effluent. Less decontaminant is required, if the decontaminant is applied by means of a gelatineous carrier substance.

The objective of this research was to develop a decontamination technique using highly effective gel-based decontaminants, which is capable of ensuring:

- a reduction of the time during which personnel is exposed to radiation;
- a reduction of the contamination of a component, thereby enabling it to be decontaminated more thoroughly after its transfer to the decontamination bays;
- the decontamination of vertical walls;
- a reduction of the quantity of radioactive effluents resulting from decontamination.

2.6.2. Work Programme: See Ref. 1, Paragraph 2.7.2.

2.6.3. Progress and Results

The final report has been published (Ref. 11).

2.6.4. Conclusions

Glycerophtalic, glycerophosphoric gel, silica gel and diopside were tested as decontaminant carriers. Adherence qualities of the gel-based decontaminating compounds were measured. Viscosities were studied as a function of time, temperature and pH. Samples of stainless steel, mild steel, aluminium, copper and plexiglass have been contaminated in laboratory, then comparatively treated by applying gels or by dipping into corresponding decontaminant solutions.

This study revealed a range of compounds of gels, which efficiently decontaminate surfaces of various materials contaminated by fission products or plutonium.

2.7. Metal Decontamination by Chemical and Electrochemical Methods and by Water Lance

Contractor: Commissariat à l'Energie Atomique, CEN Cadarache, France

Contract N°: DE-B-008-F Work Period: January 1981 - December 1983

2.7.1. Objective and Scope

The aim of this research was to develop highly effective methods for the decontamination of steel for decommissioning purposes.

Chemical decontamination methods have been studied with the aim to provide very active scouring baths enabling the contaminated surface to be laid bare, without fears of corrosion damage. In order to limit the concentration of the chemicals employed, the chemical action has been accelerated or amplified by electrolytic action.

Decontamination by high-pressure (700 bars) water lance will also be studied, including the "hardening" of the liquid jet by the addition of solids (salts of low or slow solubility or inert abrasives) and the combination of chemical treatment and water lance.

2.7.2. Work Programme: See Ref. 1, Paragraph 2.8.2.

2.7.3. Progress and Results

All decontamination tests carried out in 1983 used samples from the emergency feed-water piping of the KKI Boiling Water Reactor.

2.7.3.1. Decontamination by high-pressure jet

Samples covered with hematite were sprayed with chemical decontaminants and subsequently submitted to the high-pressure jet. A pressure of 150 bars is sufficient to reach decontamination factors (DF) of 40. A slight increase in pressure improves the DF by a factor of 1.5 for a target-nozzle distance of 5 cm, and a factor of 2 for a 10 cm distance.

2.7.3.2. Chemical decontamination

The solutions which are most effective at room temperature were compared by combining their action with ultrasonic cleaning (Table 5).

Table 5. Combined action of chemical agents and ultrasonics on BWR samples

Chemical agent (mol/l)	Temperature (°C)	Time (h)	Activity (cps) initial	Activity (cps) residual	D F
Sulphuric acid (2)	35	1	4300	2180	1.9
		2		217	19.8
		3		< 5	> 860
Hydrofluoric acid (2)	50	1	4675	236	19.8
		2		< 5	> 935
Hydrochloric acid (2)	30	1	3740	1870	2
+		2		23	162
Hydrofluoric acid (0.5)		3		< 5	> 748
Hydrochloric acid (2)	40	1	3927	1246	3.1
+		2		22	178
Hydrogene peroxide		3		0.5	> 785
Formic acid (2)	30	1	5610	2180	2.5
+		2		1750	3.2
Formol (0.1)		3		< 5	>1122

In order of decreasing efficiency these solutions include:

- the hydrogene halides (hydrofluoric acid, hydrochloric acid) alone or combined with an oxidizer (hydrogene peroxide),

- sulphuric acid,
- formic acid in the presence a reducing agent (formol).

When used together with an ultrasonic generator, all these reagents achieved nearly complete cleaning with residual activity levels not exceeding 10^{-10} Ci/cm^2.

2.7.3.3. Electrolytic decontamination

Electrolytic decontamination is effective in sulphuric or formic acid. The lowest residual activity levels were obtained using a two-step process consisting of electrolytic reduction to embrittle and remove the oxide layer, followed by oxidation to eliminate the final trace of contamination by erosion of base metal (Table 6). Radiochemical purification of the depleted sulphuric solution is possible using sodium hydroxide at pH 10-11 to precipitate the dissolved iron.

Table 6. Electrochemical decontamination of BWR samples

Chemical agent (mol/1)	Electrolytic condition		Activity (cps)		D F
	Polarity	Time (min)	initial	residual	
Sulphuric acid (2)	-	0	7,500	-	
	-	10	7,500	90	83
	-	20	90	65	115
	+	30	65	37	203
	+	40	37	37	203
Oxalic acid (0.16)	-	0	13,657	-	
	-	10	13,657	8,690	1.6
	-	20	8,690	7,957	1.7
	+	30	7,957	3,500	3.9
	+	40	3,500	3,500	4.6
Formic acid (1)	-	0	10,990	-	
	-	10	10,990	257	43
	-	20	257	170	65
	+	30	170	120	92
	+	40	120	48	229

2.8. Delegation of an expert to the US-NRC* in relation to the clean-up of the TMI-2 plant

Contractor: Studiecentrum voor Kernenergie/Centre d'Etude de l'Energie Nucléaire S.C.K./C.E.N., Mol, Belgium

Contract N°: DE-B-011-B Work Period: June 1983 - December 1983

2.8.1. Objective and Scope

Clean-up of a nuclear power station after an accident which caused dispersion of radioactivity in normally non-active areas, differs from decontamination after normal exploitation. In particular, the quantity and nature of the radionuclides are different, contamination of normally clean areas leads to access problems, and large volumes of gases and liquids are to be disposed of.

Unit 2 of the Three Mile Island nuclear power station at Middletown, Pennsylvania, U.S.A. had undergone such an accident in 1979, and valuable information is produced there about the extent of contamination and the clean-up procedures.

In order to transfer such information to the Community, the contractor assigned an expert for a six-month period (June-December 1983) to the NRC/TMI Program Office which is installed close to the power station and is responsible for review and approval of the clean-up procedures and programmes submitted by the plant owner and his contractors. This was the second delegation of an expert to TMI-2 performed under the present Community research programme, the first one having taken place in 1980-1981 (Ref. 2, paragraph 2.1).

2.8.2. Summary of Experience Gained

2.8.2.1. Treatment of accident-generated waste water

From June 1981 to May 1982, the 2,600 m^3 of high-level waste water in the reactor building sump was decontaminated by use of the so-called "Submerged Demineralizer System" (SDS) consisting of zeolite ion-exchange vessels connected together. The processed water is stored on site and used for flushing purposes.

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* United States Nuclear Regulatory Commission.

After that, the SDS was used to process 1,390 m^3 of reactor building decontamination water and 111 m^3 of tank decontamination water.

Processing of a total of 2,230 m^3 of reactor coolant according to a feed-and-bleed cycle from May 1982 to July 1983 lowered the cesium concentration in that water from 518 to 16.3 GBq/m^3.

A total of about 3 x 10^{16} Bq of Cs and Sr was trapped in 14 zeolite vessels which have been sent to Hanford for vitrification tests.

2.8.2.2. Clean-up of the reactor building and of the auxiliary and fuel handling building

In March 1982 the polar crane, the two upper floors, the refueling canal, the D-rings and missile shields and various bridges, hatches and stairwells were flushed in the framework of the so-called Gross Decontamination Experiment. Emphasis was put on flushing with water without chemicals because of the option of using the SDS to process the spent decontamination water. However, strippable coatings and acid or detergent cleaning were also tested locally.

The results of the Gross Decontamination Experiment were fair. Efficiency of the flushing was limited by use of primitive equipment, poor control of personnel movements causing recontamination, and failure to sequence operations correctly. Dose rates were lowered to the 30-400 mR/h range except on the 305' floor where hot spots of up to 1,300 mR/h were left.

Further dose reduction efforts were attempted, including local decontamination and shielding of important radiation sources. Steady recontamination through airborne radioactivity limited the efficiency of these works.

Decontamination efforts were also pursued in the Auxiliary Building, using remotely-operated equipment in some cubicles. Special emphasis was put on the management of the make-up and purification demineralizers, the organic resins of which contain some 600 TBq/m^3 of ^{137}Cs.

2.8.2.3. Reactor core status

From July to December 1982, the Reactor Coolant System was depressurized and three control rod lead screws were removed to allow insertion of a closed-circuit television camera in the space between the plenum assembly and the core. This first visual inspection revealed extensive collapse of the fuel, the bulk of which was transformed into a rubble bed. The precise distribution of the debris in the various parts of the primary loop and the sludge in the reactor building sump is still unknown, and drastic measures (3,500 ppm of boric acid in reactor coolant water, 2,000 ppm in reactor building decontamination water) are taken to prevent criticality. Radiation levels at contact of the removed lead screws were in the range of 10 - 50 R/h gamma and up to 2,000 rad/h beta.

On 25 July 1983, the system was depressurized again. Transition from reducing to oxidizing conditions increased cesium leaching drastically, and after some weeks all benefit from the previous cooling water processing (see Paragraph 2.8.2.1.) was lost. Radiation fields in the range of 8 - 40 R/h gamma and 300 rad/h beta were measured in the space between the head and the plenum.

Various samples of the deposits on the plenum surface were tested for pyrophoricity; the results showed that the water level could be safely lowered to about 0.3 m below the upper edge of the plenum, allowing "dry" radiation measurements and a new television inspection in good conditions.

The space between the plenum and the rubble bed was mapped using an ultrasonic technique. This technique confirmed the results of the television inspections, showing many fuel rod stubs protruding from the debris or hanging at the plenum but very few, if any, intact fuel assemblies left at the core periphery.

2.8.2.4. Spent fuel pool refurbishment

The TMI-2 spent fuel pool is divided into two parts: the "A" pool which contains six tanks once used to stage Reactor Building Sump water before treatment in the SDS, and the smaller "B" pool which contains the largest part of the SDS system. Present plans call for restoration of the "A" pool to its pre-accident configuration; it should then be used to store the fuel debris canisters from the defueling operations before shipment to a suitable disposal site.

Refurbishment operations include modifications to the SDS piping, removal of the tank ventilation, removal of radiation shields from above and around the pool, decontamination of the tanks, sealing and removal of the tanks, and pool-liner repairs. Works began in spring 1983, and the removal of the last tank is scheduled for mid 1984.

2.8.2.5. Reactor defueling

The Reactor Building polar crane has been refurbished and tested at no load. Load testing is planned for March 1984.

The reactor vessel head should be lifted in September 1984, and the plenum in late 1985 or early 1986, after completion of the water clean-up systems and flooding of the refueling canal and the "A" fuel pool. Removal of the fuel could then be attempted by means of various suction devices. The SDS should still be used to process some of the defueling water.

3. PROJECT N° 3: DISMANTLING TECHNIQUES

For the removal of a nuclear power plant, thick-walled steel components (e.g. reactor pressure vessel) and reinforced concrete structures (e.g. reactor shielding) must be dismantled. Here, the radioactivity demands particular requirements such as remote operation, minimum dust formation and air cleaning.

The following techniques are being examined and developed:

- thermal techniques such as plasma-arc and oxygen cutting and cutting by laser beam;
- mechanical cutting techniques such as sawing;
- explosive techniques for the dismantling of concrete structures;
- cutting of metals by intergranular fissuration.

3.1. Plasma Techniques for Cutting Mineral and Metal Materials

Contractor: Ansaldo Meccanico Nucleare, Genoa, Italy

Contract N°: DE-C-002-I Work Period: October 1980 - December 1983

3.1.1. Objective and Scope

This research relates to plasma-arc and thermal lance cutting techniques and its basic aim is to provide more information on the processes, to bring the various cutting techniques more closely into line, from a technical and economic standpoint, with the various possible applications in the dismantling of nuclear power plants, and to improve the safety and reliability of these techniques.

The aforesaid techniques were studied independently of the type of power station in which their use is envisaged; however, where specific examples had to be considered, the contractor, as a licensee for Boiling Water Reactor stations, referred to this type of plant.

3.1.2. Work Programme: See Ref.1, Paragraph 3.2.2.

3.1.3. Progress and Results

During 1983, the tests and analyses related to the plasma torch and the oxygen lance techniques were completed.

3.1.3.1. Plasma Cutting

In 1983, the activity focused on the cutting tests and in particular on the qualitative and quantitative aerosol analyses. On the basis of these analyses, suggestions on how to optimize a cutting process for minimizing the emission of harmful substances were put forward.

The cutting tests were performed in air and underwater (at a depth of 130 mm) with ferritic and austenitic steel.

Suction was secured as follows:

- For the atmospheric cutting, by a stiff smooth pipe connection with the exhaust channel; the chamber was closed; inlet was secured by two channels ending under the tests plate to be cut.
- For the underwater cutting, by a plexiglas bell placed concentrically above the plasma torch, placed on its base in the water for capturing all gas bubbles coming up.

The measurement procedures were:

- Electric measurements using an aerodynamic particle sizer for measuring particles from 0.5 µm to 15 µm and an electrical aerosol analyser for particles from 0.01 µm to 1.0 µm;
- Gravimetric measurement procedures using a gravicon impactor and a gravicon device.

For both steels, the maximum plate thickness was 150 mm in air and 103 mm underwater.

The aerosol emission decreases sharply if the cutting speed is increased and depends also on the kind of steel. In fact, for cutting thicker plates (100-150 mm) in air, the aerosol emission from ferritic steel is about 300% higher than the emission from austenitic steel.

During underwater cutting, there is a lower emission for ferritic steels, up to about 50 mm thickness, whereas for thicker plates the emission is superior.

The comparison of the values from cutting tests in air and under water shows that a water depth of only 130 mm reduces the aerosol and dust emission by factors up to 500, with practically the same cutting performance. The particles suspended in water were easily filtered through a sand filter. For both steels, the largest fraction of aerosol particles ranged from about 0.1 to 0.25 µm, when cutting underwater, and from 0.35 to 0.55 µm in air.

The quantity of dust is reduced by a factor 3 (from 28 to 9 g/m for austenitic samples of 20 mm thickness) if argon-hydrogen cutting gas is used instead of argon-nitrogen.

3.1.3.2. Thermal Lance

An operational study was performed to compare the manpower required for dismantling reinforced concrete structures in a nuclear power plant using the thermal lance technique or the blasting technique. The results showed that the thermal lance technique requires about 50% more man-hours than the blasting technique and does not damage the surrounding structures.

3.1.3.3. Analysis of radiological protection and safety aspects

An analysis of the radiological aspects related to the plasma-arc technique was developed, considering the possible use of the plasma torch for cutting metal structures directly inside the vessel or after their removal outside the plant in an ad-hoc facility.

Due to difficulties in handling the instrumentation of the plasma torch apparatus, the results of the tests performed with the plasma torch at the ENEA Casaccia Centre were not significant.

3.2. Diamond-tipped Saws for Cutting Concrete Structures

Contractor: Central Electricity Generating Board, Barnwood, United Kingdom

Contract N°: DE-C-003-UK Work Period: April 1980 - December 1983

3.2.1. Objective and Scope

The objective of this research is to develop a suitable diamond saw capable of cutting away remotely the inner 1 m activated layer of a reinforced concrete biological shield or pre-stressed concrete pressure vessel. Since the dose rates within these structures will be too high to permit manual work for practical periods, the saw must be capable of being remotely controlled and operating reliably for long periods. In addition, the cooling system must be designed to be efficient but produce the minimum practical amount of slurry.

This research concerns all types of nuclear power plants.

3.2.2. Work Programme: See Ref. 1, Paragraph 3.3.2.

3.2.3. Progress and Results

3.2.3.1. Testing and sawing on a horizontal concrete floor

The initial testing of the saw with the 2.5 m blade cutting into a concrete floor, reinforced with 18 mm steel bars culminated with the successful removal of a 600 mm cube. The time required to both saw the cube profile and snap out the block was in the order of 90 minutes.

With the blade sawing through 760 mm of reinforced concrete and 240 mm of sandstone bed rock, thus producing a kerf 1 m deep, the rate of advance through the 18 mm reinforcing steel and concrete was 200 mm/min and through the concrete alone 300 mm/min, the sawing rates being 12 m^2/h and 18 m^2/h, respectively.

In this test, the saw was arranged to produce four kerfs: two parallel kerfs 600 mm apart and 600 mm deep, and subsequently crossing these with two parallel kerfs, also 600 mm apart and 600 mm deep, in order to establish the 600 mm concrete cube profile.

Having established this profile, wedges were evenly spaced over a 600 mm length to snap out the lower face of the cube.

3.2.3.2. Testing and sawing on the vertical face of a wall

The equipment for carrying out the sawing of the wall consisted of a mobile crane with stabilizing outriggers and telescopic boom to which was attached the circular saw supporting frame. Fig. 4 shows the general arrangement of the sawing head. The crane boom was able to position the sawing head in both horizontal and vertical planes and the saw frame was pivotted to allow the frame to position itself parallel to the wall surface resting on the two driving wheels and one castor wheel. The crane and boom were able to exert a force onto the frame, and hence the saw, and the driving wheels then traverse the saw frame along (or down) the wall with the castor wheel trailing.

Vertical sawing down the wall was easily achieved. Even when the drive couplings failed during a vertical cut the gravitational forces were sufficiently well balanced to allow the saw to continue in a straight vertical line guided and controlled by the crane driver.

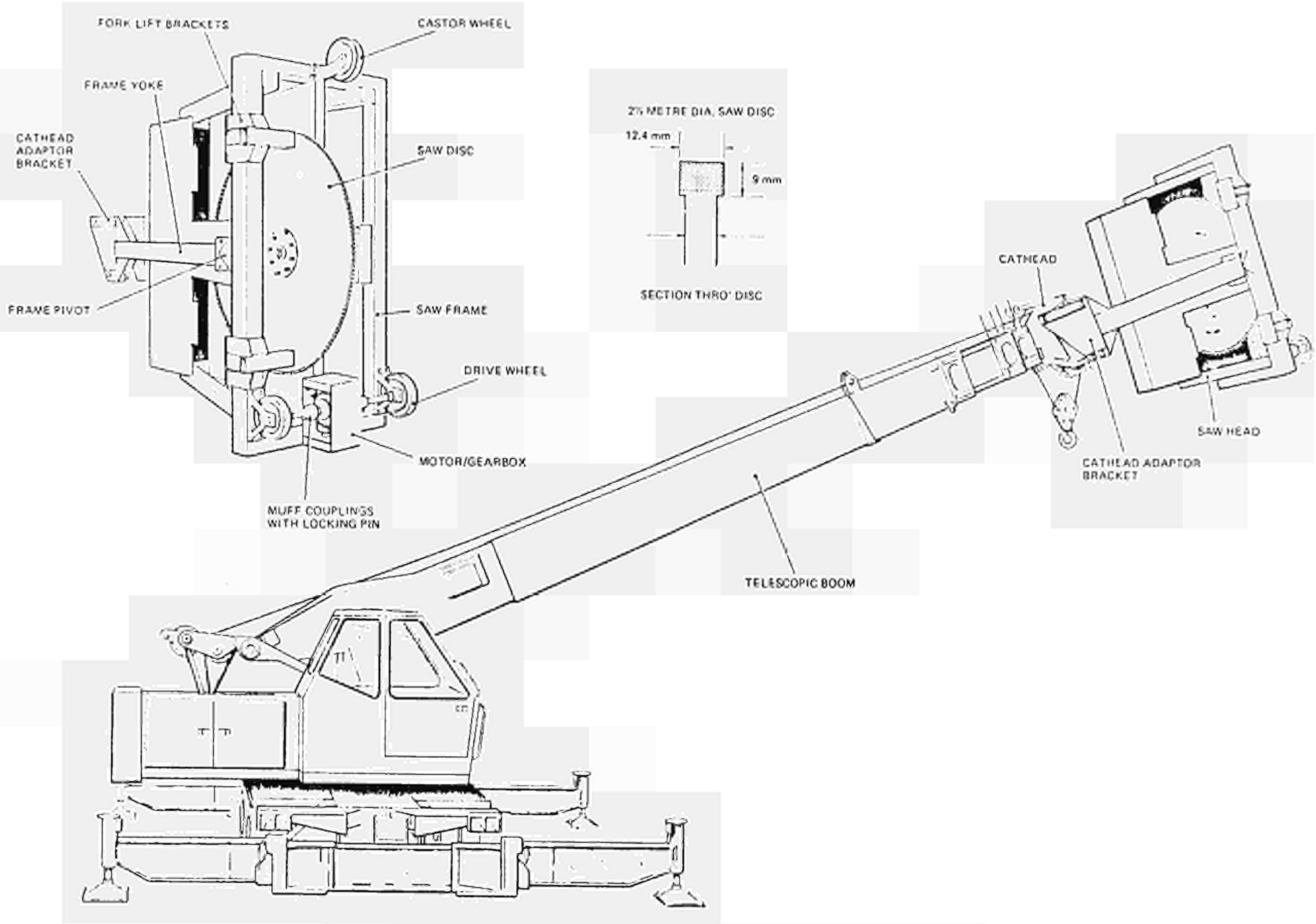

Figure 4. General arrangement of sawing head

When sawing horizontally across the wall, the out-of-balance forces caused diversions from a straight path. This was rectified by the addition of counterbalance weights on the saw frame and also adjustment of the castor wheel radius.

The fastest sawing rate was achieved on the final horizontal cut across the wall. A full plunge was carried out in 7 min without stalling the drive or approaching an overload condition. This is equivalent to a sawing rate of 15 m^2/h.

The maximum traverse speed over a distance of 1 m with the sawing disc plunged 1 m deep into the concrete was approximately 160 mm/min. This is equivalent to 10 m^2/h or a kerf 1 m deep and 10 m long every hour.

3.2.3.3. Concrete cube removal

After the two horizontal and two vertical cuts had been completed to establish the 1 metre cube profile, small wedges were inserted in the upper edge of the cube at equal distances along the 1 metre length. These were evenly tapped in to strain and crack the rear face of the cube. This was not successful and resulted only in spalling the corner edges of the concrete. Larger wedges with broader faces were then used and these were successful in snapping the rear wall of the cube.

With the concrete cube now released, the technique for complete removal from the wall was to withdraw the cube onto a fork life truck and lower it to the ground. The front wall of the cube was drilled and a lifting eye plugged into the cube. A cable attached to the eye and the fork lift truck mast was used to withdraw the cube from the wall until it could be fully supported by the life forks. Unfortunately, restricted access for the truck caused a misalignment of gravity centres between the cube and the truck and the cube fell onto the test area base. The cube split along a place which was identified as a construction joint about 200 mm from the top of the cube.

3.2.4. Conclusion

The objective of this phase of the project was successfully achieved and was also within the programme and cost estimates.

The limited size of the test wall prevented any practice runs to ascertain optimum speed conditions, saw wear rates and operator experience. Therefore, the positioning of the saw and the sawing action were the primary concern of the test team. The rate of sawing and possible improvements in technique and application were secondary to achieving the objective without damage to personnel or equipment.

The sawing rates quoted may not be the best possible. It is believed, however, that whilst the plunge rate achieved was very near to the optimum, the traverse rate could be almost doubled with further development. The wear rate and life of the saw disc could not be accurately estimated from the work carried out by the saw throughout the testing. At the end of the testing the blade had performed approximately 100 m^2 of sawing and the diamond tips appeared half worn. In general tip wear is constant throughout its full depth, hence a blade of this type could be expected to saw in excess of 200 m^2, which is three eight-hour shifts, before re-tipping.

3.3. Plasma-oxygen Cutting of Steel Pressure Vessels

Contractor: Salzgitter AG, Salzgitter, Germany

Contract N°: DE-C-004-D Work Period: April 1980 - December 1983

3.3.1. Objective and Scope

The purpose of this research is the development of a technique based on a combination of plasma and oxygen torches, capable of cutting up from the clad side thick sections of low-alloy steel clad with stainless steel, occuring in the pressure vessel of light water reactors.

The work comprises cutting tests on thick (up to 600 mm) inactive specimens, optimization of cutting parameters and off-gas filtering system, and a concept study of dismantling reactor pressure vessels.

3.3.2. Work Programme: See Ref. 1, Paragraph 3.4.2.

3.3.3. Progress and Results

The development of a combined cutting torch which consists of a plasma torch and an oxyacetylene torch has been continued.

3.3.3.1. Characterization of aerosols and testing of filter systems

The objective was to analyse the dust and aerosols and to examine selected filter systems with regard to their dust storage capacity and their degree of separation.

In order to obtain detailed information on the particle size distribution of the dust and aerosols arising in the course of thermal cutting, extensive preliminary investigations were carried out on ferritic and austenitic steel plates of 20, 40 and 80 mm thickness. The same tests were then performed using the combined plasma-oxyacetylene torch on thick, clad plates (150, 300 and 600 mm).

The cutting tests, aimed at collecting and analysing the dust from single and combined cutting, were performed in the gravity position and with a suction flow rate of 3000 m^3/h. The dust and aerosols were collected in the suction line of the test rig both ahead of and behind a filter stage or filter combination. Measurements were taken using the EM 100 dust collecting probe of Sartorius Membranfilter GmbH, and the Electrical Aerosol Size Analyser (EAA 3030) of Thermal Systems Incorporated. These were then evaluated by gravimetric and particle size analysis. The measured dust concentrations allowed the separation degrees of the filter stages and combinations to be determined.

The filter system comprised a bulk material pre-filter, an intermediate box filter, which included a first filter of the EU 7 classification and a second filter of the S class, and a final barrel-type filter with a class S element. The first tests showed that the barrel-type filter used as a pre-filter had inadequate dust storage capacity. Therefore, tests with this filter were abandoned.

The following median values were determined for the particle size distribution during the individual cutting tests:

Cutting process	Material	Median values of particle size distribution (μm)		
		Raw gas	Gas filtered once	Pure gas
Plasma	Austenitic	0.24	0.13	0.01-0.02
Plasma	Ferritic	0.13	0.075	0.01-0.02
Oxyacetylene	Ferritic	0.23	0.13	

The tests showed that the least amount of dust is developed when cutting ferritic steel.

Evaluation of the particle size distributions determined for the combined cutting of thick components showed that the median value in both the raw gas and the gas after one filter pass equalled about 0.24 μm.

Maximum values of the same level can be explained by the fact that the separation degree of the pre-filters does not depend on particle size. In the pure gas, the particle size distribution has a median value of about 0.1 μm.

At 99.999%, the overall separation degree for the tested filter combinations can be viewed as satisfactory. In contrast to this, service life is not adequate.

3.4. Dismantling of Concrete Structures and Metal Components Using Laser

Contractor: FIAT TTG SpA, Torino

Contract N°: DE-C-005-I Work Period: April 1981 - April 1983

3.4.1. Objective and Scope

The aim of this research is to study laser techniques for the following applications in particular:

- drilling holes in prestressed and ordinary reinforced concrete for placing explosive charges with a view to demolition;
- cutting reinforced concrete structures;
- cutting thin and medium-thick metal components.

The laser cutting technique will be studied independently of the type of nuclear power station in which its use is envisaged; however, where specific examples have to be considered, Pressurized Water Reactor stations and Gas-cooled Reactor stations will be referred to.

3.4.2. Work Programme: See Ref. 1, Paragraph 3.5.2.

3.4.3. Progress and Results

Until May 1983, the 15 kW carbon dioxide laser was not in working condition. In June, tests were resumed, in particular using 10 kW laser power, the copper needle nozzle of 1.5 mm diameter, and an oxygen assistant gas pressure of 7 bars. The following tests were performed in these experimental conditions.

3.4.3.1. Cutting and drilling of concrete

Cutting tests on steel-reinforced concrete were carried out. The average cutting depth obtained was 94 $\pm$ 14 mm at 0.02 m/min velocity and the maximum cutting depth was 160 mm at 0.01 m/min velocity. The large standard deviation is mainly due to quality variation of the concrete samples used. The kerfs were regular and 3 to 6 mm wide. Laser cutting of steel-reinforced concrete was easier than cutting of non-reinforced concrete. Furthermore, holes of 6-10 mm diameter and 80 mm depth were drilled, using 1 or 2 impulses of 5 seconds.

In conclusion, cutting of concrete and steel-reinforced concrete poses no problems except for the high dust production. Drilling is possible, but the holes obtained were too small to introduce an explosive charge for dismantling.

3.4.3.2. Cutting of steel

Cutting tests were carried out on three types of specimen, i.e.:

- 40 mm thick plates of low carbon mild steel (grade Fe 42 C);
- 45 mm thick plates of stainless steel (grade AISI 304);
- 26 mm outer-diameter bundle pipe of Inconel 600, in two superposed rows.

For the stainless steel specimens, the average cutting depth obtained was 35 $\pm$ 5 mm at 0.2 m/min velocity. For a given power level and cutting rate, the average depth of penetration is 20 to 30% higher in mild steel than in stainless steel.

For the Inconel 600 bundle pipe, the velocity needed to cut was 0.1 m/min and 0.07 m/min for the two pipe rows with offset and vertically alined axis, respectively. The cut was irregular and unclean.

3.4.3.3. Analysis of by-products

Two series of measurements were carried out to characterize laser cutting emission by-products. The first measurement was based on isokinetics sampling, the second on global weighing. All experiments used the same box containing moving workpieces and having an input hole for the laser beam and an output hole for the suction of cutting by-products.

During the first series of tests, an isokinetics emission sample was taken on the exhaust pipe and the following measurements devices were used: a centrifugal separator, a particle counting system based on optical scattering information, a beta-ray balance to measure differential filter densities, organic vapour and nitrogen oxide analysers. Emission volume flow was 1350 m^3/h and dust mass flow varied from 20-30 g/h for steel to a maximum of 500 g/h for concrete. Granulometric distribution measurements showed more than 60% of particle number between 0.3 and 0.5 µm. Organic vapour concentrations were less than 0.16 ppm and the nitrogen oxides emission was less than 2 g/h without any environmental problem.

During the second series of tests, the weight of airborne particles produced during laser cutting of concrete was determined. This determination was based on the difference between the concrete weight variation after cutting and the weight of large particles collected on the walls of the box. For a standard cutting condition (10 kW, 0.02 m/min), the average rate (1850 g/h) was nearly three times higher than the rate measured in earlier experiments. The discrepancy was attributed to the limitations of the first measurement method at large (>5 µm) and small (<0.3 µm) particle diameters. In order to better understand by-products of small diameter, which are difficult to filter out, a third experiment was programmed.

3.5. Explosive Demolition Techniques for Concrete Structures

Contractor: Taylor Woodrow Construction Ltd, Southall, United Kingdom

Contract N°: DE-C-006-UK   Work Period: January 1981 - March 1983

3.5.1. Objective and Scope

The objective of this research is to optimize and assess explosive techniques for the demolition of the radioactive concrete structures of nuclear power plants in respect to safety, radiation protection and costs.

The research is directed mainly at the biological shields of early Magnox reactors and the prestressed concrete pressure vessels (PCPV) of later Magnox and Advanced Gas-cooled Reactors. Relevant 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.

3.5.2. Work Programme: See Ref. 2, Paragraph 3.6.2.

3.5.3. Progress and Results

3.5.3.1. Cratering tests - single shots

Edge effects (2 tests): Earlier tests had indicated that the size of the models being used might be too small with edge effects influencing the results. To investigate this, two large diameter models, but with characteristics similar to previous models, were tested. These tests indicated that the results from the smaller models may have been exaggerated by boundary conditions. It was therefore decided to use the larger diameter model characteristics for future tests.

Method of charge placement effects (3 tests): It was decided to investigate what effects the method by which charges were buried into the models may have on the cratering characteristics. Three large diameter models were built, on which charge holes were provided from the face to be cratered. All other charateristics were kept similar to previously tested models. These tests indicated that the front-face charging method used here does not modify behaviour significantly nor cause reductions in the size of crater. This introduces the possibility of charging a structure, e.g., a biological shield, from the inside.

3.5.3.2. Multiple firings on discs

Charge separation effects (4 tests): Four separate single shots were fired on an existing large diameter model having a crater from a previous test. The separation of these single shots from the centre of the existing crater was varied in order to optimize on the charge separations to be adopted in multiple firing tests. These tests indicated that ridges are formed in the cratered region between the charge hole positions. The heights of these ridges decrease with decrease in charge separation. From these tests a charge separation and depth of burial was selected to be used in subsequent multiple firing tests.

Cascade multiple firings (2 tests): A group of six multiple shots was fired in cascade around a crater on an existing target. The separation distances and delay times between the charge holes were kept constant. The test was continued by firing three further shots in the cratered area using similar separation distances and delay times.

From these tests an indication of the probable effects of cascade firings on the results was obtained. It also became evident that cascade firings could introduce misfires into the programme.

Simultaneous multiple layer stripping (6 tests): A group of seven multiple shots were fired simultaneously in the cratered region of an existing target. This was followed by two and three separate combinations of groups of multiple shots to strip off layers from a second and third target respectively. These initial tests indicated no detrimental effects from firing the shots simultaneously. It was also indicated that layers can be stripped off from the target surface using the proposed methods.

Multiple separation tests (2 tests): Earlier tests had indicted that when multiple shots are fired using charge holes with large separations, then ridges are formed in the cratered area. Two further tests were therefore carried out in an attempt to reduce the ridge heights. It was found that very flat bottomed craters can be produced by firing charges at smaller separations from one another.

3.5.3.3. Cratering tests on rings

Single shots (3 tests): A test was carried out on a ring to check the effects of curvature on the crater formation. The indications from the crater formed was that there was little influence from curvature on the results. Two further tests were carried out on the ring in which charge holes parallel to the face to be cratered were used. The results were similar to those obtained using charge holes perpendicular to the face to be cratered with slightly reduced crater dimensions.

Simultaneous multiple (4 tests): Three tests were carried out on a selected sector of the ring to strip off layers of concrete from the inner face. Different charge hole arrangements were used to investigate the effects of such differences on the craters produced. These tests indicated the charge hole arrangement, separation and depths of burial to be used for the tests on the complete cylinder. A further test was carried out to investigate the effectiveness of firing sequentially to strip off concrete using predrilled charge holes. This was achieved by overdrilling some holes to a deeper depth and firing shots only in the shallower charge holes. It was concluded that the remaining parts of the overdrilled holes were good enough to be used as charge holes for the next layer stripping test and that the method can be adopted in the cylinder test.

3.5.3.4. Cratering tests on complete cylinder

Simultaneous multiple layer stripping (4 tests): A sector over the complete height of the model was selected for testing. The first, second and third layers were cratered off using multiple groups of charges at pre-determined depths of burial and charge separations. The fourth layer stripping was carried out on only the middle ring of the cylinder.

For this test the middle ring sector was divided up into two areas, in each of which a different charge hole arrangement was used. This test was designed to optimize on the charge hole arrangement that would produce a very flat bottom crater.

These test results indicated possible economic and structurally safe cutting techniques that can be used to strip off activated concrete from the inner face of biological shields without losing containment integrity.

Multiple separation tests (6 tests): These tests were introduced into the programme to investigate the effects of different charge separations and different concrete strengths on cratering in curved surfaces. For these tests, the top and bottom rings which had respectively the lowest and highest characteristic strengths were used. For each ring, three different separations were used in separate groups of twelve charges. As with the flat discs, these tests indicated that the flatness of the crater bottoms was improved as charge separation was reduced and a marginal increase in crater volume was obtained with increase in concrete strength.

3.5.3.5. Concrete stripping and cutting tests

Fourteen tests have been carried out with various designs of linear shaped charges (LSCs). The designs of the LSCs varied in wedge liner angle and thickness. In addition, the standoff distance between the front face of the shaped charge and the concrete target was varied. The results indicated an improvement in performance with a reduction in standoff distance. Comparisons between the various test results suggested that the thicker liner LSCs gave a noticeable improvement in cutting characteristics over thinner liner LSCs.

3.6. Cutting of Steel Components by Intergranular Fissuration

Contractor: Commissariat à l'Energie Atomique, CEN Grenoble, France

Contract N°: DE-C-007 Work Period: July 1982 - December 1983

3.6.1. Objective and Scope

The aim of this study is to develop a new method for cutting metal components of nuclear power plants, which produces virtually no secondary waste. In this method, a controlled intergranular fissure is produced in a heated area of the component by the addition of a molten material which gives rise to the formation of brittle compounds.

The research will mainly be concerned with the following points:

- detailed study of the process;
- identification of the maximum achievable cutting thickness;
- preparation of a manual of procedure for the application of the process.

3.6.2. Work Programme: See Ref. 3, Paragraph 3.7.2.

3.6.3. Progress and Results

The development of the method for cutting components of nuclear power plants by intergranular fissuration was continued.

A systematic experimental study was carried out in order to determine the fissuration mechanism and the influence of the work-piece geometry. This study involved, e.g., the analysis of the thermal field, determination of the crack propagation criteria and interpretation of the mechanism. In order to improve the results, the point-by-point temperature measurements were replaced by continuous measurements by means of an infra-red camera (thermography). From the experimental data of temperature, the stress field generated by non-uniform heating was calculated using a computer code based on the finite elements method and the assumption of elastic material behaviour. These calculations clearly showed the traction and compression zones and satisfactorily justified the assumptions made.

The cutting tests were performed on austenitic and ferritic steel plates with three types of heating, i.e., using a Tungsten Inert Gas (TIG) torch, a Metal Inert Gas (MIG) torch and high-frequency induction heating.

For a thickness of more than 20 mm, induction heating (3 to 10 kHz) was used in order to avoid melting of the workpiece material, to obtain a correct thermal field and the additional material progression into the work-piece. The maximum cutting thickness achieved was 50 mm for stainless austenitic steel (AISI 304L) and 60 mm for ferritic steel (A 52 French standards). Metallurgical investigations of the fissured zone were carried out.

Feasibility tests for remotely controlled cutting were performed. For this purpose, a fissuration head to be fitted in the "slave" grip of a manipulator controlled directly by the "master" was manufactured. Preliminary tests carried out on stainless steel plate of 4-5 mm thickness gave good, clean cuts.

3.7.1. Review of Systems for Remotely Controlled Decommissioning Operations

Contractor: Gesellschaft für Systemtechnik mbH, Essen, Germany

Contract N°: DE-C-008-D Work Period: May 1983 - April 1984

and

Contractor: Ateliers de Construction Electriques de Charleroi, Charleroi, Belgique

Contract N°: DE-C-009-B Work Period: May 1983 - April 1984

3.7.2. Objective and Scope

In order to limit radiation exposure and occupational hazards during decommissioning tasks in nuclear facilities, and to allow interventions at an early stage after shutdown of a nuclear facility, remotely controlled and, during execution of specific operations, autonomous systems are needed. For different tasks the mechanical support will often be different, but sub-systems for monitoring, tracking, distance measurement and so may be common.

The objective of this joint study is to analyse the decommissioning tasks in nuclear facilities, to review the applicable existing equipment, to carry out the first conceptual studies of remotely operated equipment needed and finally to identify the areas where further research is needed.

3.7.3. Work Programme

The operations to be taken into account are all types of procedures arising in the decommissioning of nuclear facilities. As the dismantling is normally delayed for a number of years, in most cases the radiation fields are of low intensity as compared to those existing in normal reactor operation. Nevertheless, measurement programmes must be carried out at a very early stage after reactor shutdown and, in situations of accident, early decontamination or dismantling operations could be needed.

The nuclear facilities to be considered are fuel fabrication plants, reactors of the types existing in the member countries of the European Community, reprocessing plants, and all facilities for treatment, packaging and interim storage of radioactive waste. The main applications to be considered are:

Monitoring: visual inspection of components or rooms, position measurements, measurement of radiation fields.

Decontamination: methods like washing with water, possibly containing chemicals, water-lance, blasting, electropolishing and flame-scarfing could be applied.

Dismantling: cutting of activated vessels and internals by mechanical or thermal techniques, drilling of holes in concrete, cutting of concrete by mechanical or thermal techniques, cutting of contaminated tubing and other equipment, including possibly treatment and transport of debris.

The specific areas and the main responsible organizations for each area are as follows.

a) Review of already performed remotely controlled operations in decommissioning (GST and ACEC): The available information on existing experience will be collected. This search should not be limited to decommissioning, but also take into account related fields, like maintenance and repair of nuclear installations.

b) Definition of the decommissioning tasks and of the techniques which may possibly be applied (GST): The tasks to perform in decommissioning will be assembled and the possible techniques will be listed. Examples are cutting of steel vessels by mechanical or thermal methods, or decontamination of concrete surfaces by chemical, mechanical or thermal methods.

c) State of the art of existing equipment applicable to decommissioning tasks (ACEC): Techniques and systems already applied in the nuclear and chemical fields and in automated assembly lines will be analysed for their potential of application in decommissioning. In operator-linked action, the main points will be data transmission, feedback in master-slave operation, positioning, treatment of visual information and tracking systems. In autonomous action, existing hard- and software for automatic and dynamic positioning, sensoring, calculator-controlled processing in pre-defined sequences, motion supervision of subsystems and control of process will be the main points. In all cases the limitations of procedures and sub-systems will be indicated, e.g., effects of radiation fields (need for shielding), dust and aerosols, temperature, mechanical loads, and speed of data transmission.

d) System analysis of all tasks suitable for remote handling and specification of corresponding hard- and software (GST): A system analysis of all tasks and techniques defined above will allow the arrangement of tasks corresponding to the need for, or the usefulness of, remote operation. Certain interventions can be done only by remote operation (strong radiation fields); for other types of task, remote handling will allow reduction of individual exposure, and there is the possibility of reducing the collective exposure of a group of workers at a rather low cost, for example by remotely controlled treatment, packaging and transport of radiactive waste. For the tasks which are most suitable for remote operations, a detailed specification of performances for the corresponding system will be drawn up (range of loads, environments, surface or volume to be covered, etc.).

e) Conceptual design of the most promising systems (ACEC): A conceptual design of the most needed (and not yet existing) systems, as specified above, will be done. Design will be based on existing elements in micro-electronics, signal analysis and data transmission. High reliability and easy repair will be important considerations, and a cost estimate will be included.

f)Needed developments and recommendations for future work (GST and ACEC): Following the evaluation of tasks and the analysis of existing components, the need for not yet existing equipment and components will be identified. The recommendations for future research and development should cover all fields related to remote operation in decommissioning; however, possible future development should not be investigated for advanced control and data transmission systems, which will be treated in parallel in another contract, and for specific memory chip or computer development.

3.7.4. Progress and Results

During 1983, the contractors concentrated on tasks a,b,c and d.

With regard to task "a", the decommissioning operations performed in various nuclear facilities (decontaminated and/or dismantled to various degrees) were reviewed. Systems used, under construction or envisaged, for dismantling the reactor vessel or internal components were described. It was noticed that numerous operations were performed manually.

With regard to task "b", a synopsis of the tasks occurring in decommissioning of nuclear power plants and applicable decommissioning techniques were discussed and tabled together with their main characteristics and technical data.

Many of the techniques still under research relate to the cutting of the thick-walled reactor vessel and biological shield. The positional arrangements of apparatus, components and structures are important for the development of remote-handling systems. Therefore, drawings of layouts and details of various plant types were compiled.

With regard to task "c", a survey of the existing remote-controlled or autonomous devices with their principal characterisitics was carried out. The devices examined are relevant not only to the nuclear fields but also to the industrial, oceanic, mining, military and civil protection fields. They are classified as follows:

- stationary industrial robotic systems;
- mobile robotic devices (vehicles);
- control systems;
- subsystems such as vision equipment, telemanipulators, sensors and collision protection equipment.

With regard to task "d", all tasks and techniques were examined for their need, usefulness and requirements with respect to remote operation. For this end it is necessary to define and typify those systems, task areas, plant-specific and task-typical work which is best performed by the various forms of remotely controlled operations. Many tasks, especially cutting the thick-walled main components, can only be performed by mechanized techniques because of sheer size, apart from radiological limitations. Fixed installed process machines and automates are best suited for many tasks. The most difficult adaptation is needed for remotely controlled working sequences from a mobile task platform.

Most essential is a detailed task analysis within the plant and work contents in order to specify the concept and the corresponding hard- and software requirements. The definition of performance profiles includes specifications of techniques used, positions worked from and work sequences within the task.

4. PROJECT N° 4: TREATMENT OF SPECIFIC WASTE MATERIALS: STEEL, CONCRETE AND GRAPHITE

In the dismantling of nuclear power plants, large amounts of radioactive steel, concrete and - in gas-cooled reactors - graphite will arise. This waste shall be volume reduced and suitably conditioned for disposal.

The following research works are being done:

- 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 immobilize the radioactivity;
- comparative assessment of various modes of treatment and disposal of radioactive graphite.

4.1. Assessment of Management Modes for Graphite Waste

Contractor: National Radiological Protection Board in association with: Central Electricity Generating Board, United Kingdom Atomic Energy Authority and British Nuclear Fuels Ltd; United Kingdom

Contract N°: DE-D-001-UK Work Period: October 1980 - December 1982

4.1.1. Objective and Scope

The aim of this research is to yield more precise data on the radionuclide composition of the graphite waste arisings, to provide new information on the relative merits of a variety of disposal options, and to provide guidance on the selection of the most appropriate scheme. The disposal options examined are incineration, sea dumping, deep geologic disposal and shallow land burial.

4.1.2. Work Programme: See Ref. 1, Paragraph 4.1.2.

4.1.3. Progress and Results : See Ref. 1, Paragraph 4.1.3.

4.1.4. Conclusions

The models used for the calculation of the radiological impact of shallow land burial have been revised, and new estimates have been made. The conclusions given in the 1982 Report (Ref. 3, Paragraph 4.1.4.) have therefore been re-worded as follows.

The radionuclide inventories of Magnox and AGR graphites 10 years after the end of reactor life were calculated. The main contribution is from activation of the carbon itself and the stable impurities.

A lesser contribution to the total inventory may arise from fission of natural uranium present as an impurity in the graphite. A measurement on a single sample, and other indirect evidence, suggest that the uranium level is unlikely to exceed 0.1 ppm. Even at this level, the contribution to the total radiological impact of disposal would be relatively minor.

Surface contamination of the graphite by circuit crud or by volatile fission products both depend on the design and operating history of the particular reactor. While these contributions to the total radionuclide inventory and the radiological impact of disposal cannot be estimated in a generic study, they will probably be relatively small.

Three packaging methods for graphite were described: drum containers, shielded overpacks and large concrete containers. Each is logistically feasible and could be used for either sea or land disposal, though the type at present used for sea dumping would probably be less suitable for land disposal.

Cost estimates for each of the three packaging methods showed that any of the three could be implemented at acceptable cost. Since the estimates were made on different bases, they cannot be used to make valid intercomparisons between the packaging methods. However, it appears that larger packages are more cost-effective. Further studies are required in this area.

Unirradiated graphite in large blocks retains its mechanical integrity at pressures up to 900 bar. This finding was confirmed during leach tests on small samples of irradiated graphite at 450 bar and 2.5°C, the conditions on the deep ocean bed at a depth of 4 km. Simulated seawater was used as the leachant, and parallel tests were carried out under ambient laboratory conditions using simulated seawater, simulated groundwater and demineralized water. In all cases, ^{14}C had the lowest leach rate and ^{133}Ba or ^{134}Cs the highest; the effects of the different leaching conditions were found to be small.

Ocean-bed disposal of graphite is predicted to lead to acceptably low potential individual doses.

Two forms of deep geologic disposal of graphite have been considered: at an inland site in clay, and at a coastal site in shale.

For a wide range of assumed leach rates and hydrogeologic parameters, either type of generic site could lead to acceptably low potential individual doses.

In the case of shallow land burial it would be necessary to restrict building on the site for some decades after disposal, until the ^{60}Co content has appreciably decayed; such restrictions are quite feasible. The long-term radiological impact of graphite disposal by shallow land burial requires further investigation, mainly because of the present difficulties in modelling the behaviour of ^{14}C in soils and shallow ground-water. These investigations must be on a site-specific basis.

A feasible alternative to direct disposal of packaged graphite is incineration, followed by disposal of a smaller volume of ash. But incineration presents logistical difficulties additional to those of direct disposal: either several incinerators must be built at different sites, or the graphite must be transported to a smaller number of centralized incinerators; the corresponding costs could vary widely. The radiological impact of incineration would arise almost entirely from the atmospheric discharge of ^{14}C and is greater than that of many other options, especially in the early term. Incineration does not seem a desirable alternative to direct disposal, unless the reduced volume of ash is judged to override all other considerations.

All the graphite management modes considered in this generic study merit further investigation. The choice of the optimum management mode must be made in some specific context, and the optimum will depend on the specific circumstances. When making such a choice (possibly with the aid of cost-benefit analysis) further information would be needed on the time-varying probabilities that predicted individual and collective doses would be received, and on the individual dose rates at which collective doses might accrue.

4.2. Treatment of Contaminated Steel Waste by Melting

Contractor : British Steel Corporation, Sheffield, United Kingdom

Contract N° : DE-D-002-UK   Work Period : January 1982 - December 1984

4.2.1. Objective and Scope

Large quantities of radioactive steel waste will arise in the dismantling of every nuclear power plant. Melting is a promising method of conditioning this waste, with the purpose of volume reduction and radioactivity immobilization. On the other hand, with a view to recycling the steel, the possible decontamination effect of melting has to be known.

The aim of this research is to investigate the melting process and its product, by-products and radiological impact, by experiments spanning from the laboratory to the industrial scale with steel scrap originating from contaminated reactor components.

4.2.2. Work Programme : See Ref. 2, Paragraph 4.2.2.

4.2.3. Progress and Results

The melting work this year was mainly at a larger scale than before. Six pilot plant melts, between 2 and 4 t, were made in an electric arc pilot plant. Three other melts were made at laboratory large scale in an induction furnace, two of these being 500 kg and the other 250 kg.

4.2.3.1. Melts for determination of activity concentration

Five melts (electric arc pilot plant) on activated/contaminated components taken out of nuclear power station have been carried out. The purpose was general observation of all melt products and specific slag control of Cs radioactivities. The furnace system (Figure 5) comprises fume extraction system with the filter bags, fume sample points and high sensitivity radiation detectors. Sampling of metal, slag, off-gas and discharged hopper dust cake was undertaken throughout the operations.

Melting of CEGB Magnox reactor fuel chute pieces (Table 7). The pieces were predominantly activated by ^{60}Co but slightly ^{60}Co contaminated. There was no transfer of radioactivity to the slag or any presence in the dust cake. Only in an 18-minute period during melt-down was any level of radioactive present in the duct fume. The radioactivity concentration in the ingot was uniform. All radioactivity concentrations have been determined by comparison with accurately dispensed amounts of radioactivity of the same kind and measured in a counting unit under the same specimen geometry.

Melting of AERE Winfrith SGHWR steam turbine diaphragm (Table 7). There was no measurable transfer of radioactivity to furnace slag and virtually none carried over to the hopper dust cake. There was an extremely

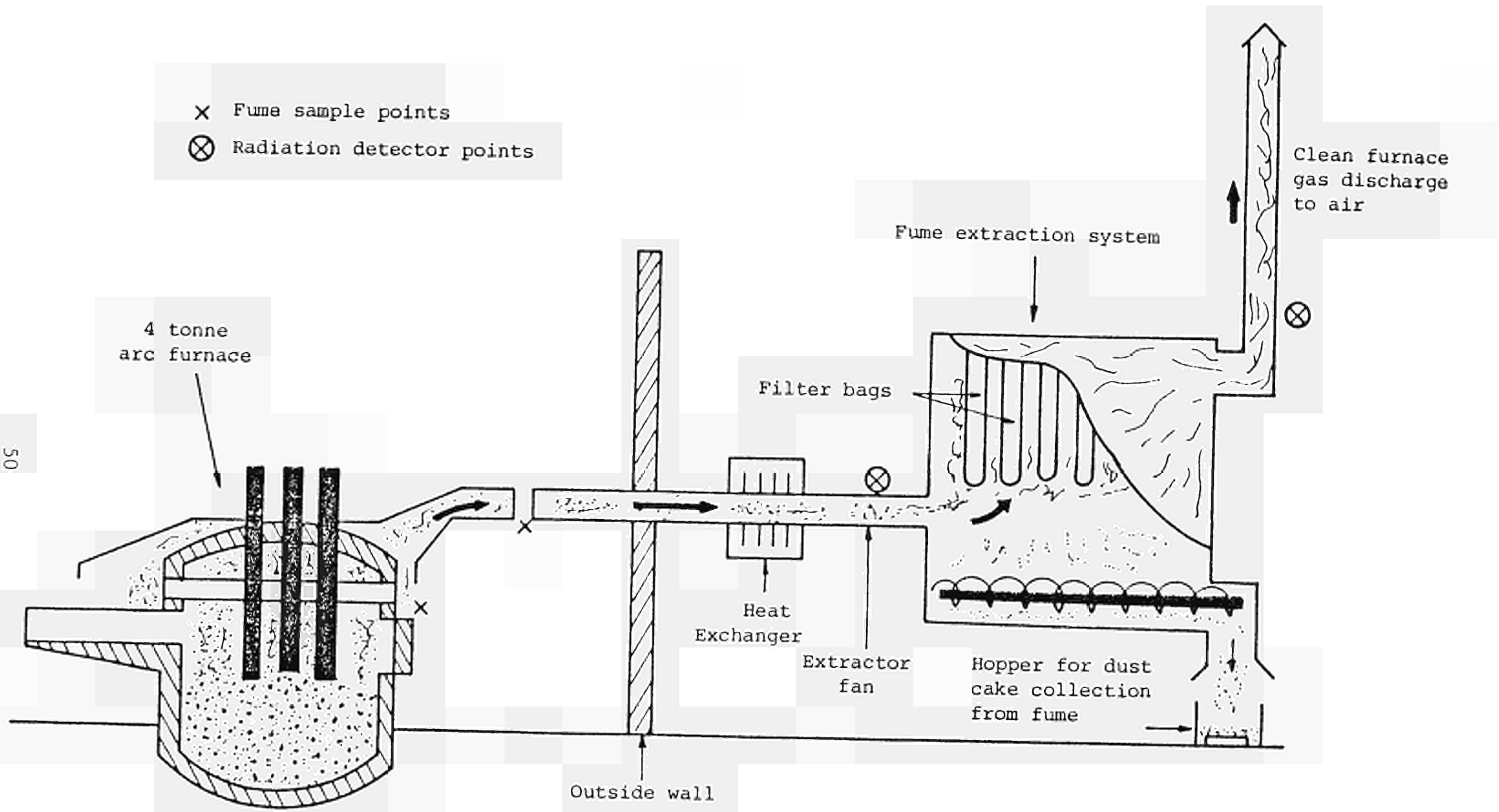

Figure 5. Grangetown pilot plant: electric arc furnace system (4 t)

small level of radioactivity in the furnace duct fume but only during melt down. The product ingot had uniform radioactivity concentration.

Table 7. Radioactivity data for pilot plant melts

	Maximum surface dose rate ^{60}Co (mrad/h)	Total product ingot weight (kg)	Product radioactivity concentration		
			Product ingot (μCi/g)	Furnace duct fume (μCi/ml)	Extractor dust cake (μCi/g)
CEGB Magnox reactor fuel chute material (770 kg)	3.5	3000	6.2×10^{-5}	4.5×10^{-10}	Zero
AERE - SGHWR steam turbine diaphragm (645 kg)	3.0	2000	3.6×10^{-5}	$< 10^{-10}$	3.0×10^{-7}

Melting of AERE Winfrith SGHWR cover plates. Ten of these plates, having been decontaminated by phosphoric acid at Winfrith, were judged to have about equal amounts of ^{60}Co and ^{137}Cs contamination remaining, with ^{134}Cs present at about 10% of either. Activity levels between plates and between faces of the same plate differed widely. The average level was 6.3×10^{-6} μCi/cm².

During the melt of seven of the plates (Table 8-a), the slag was made up to 20% silica - as high as the basic furnace lining would withstand. Also, the lime value was reduced to the lowest workable level. To follow Cs retention, slag samples were taken at 10-minute intervals through the melt, showing that a great deal of total radio-cesium was retained in the slag for 10 min (80 μCi for an assumed slag mass of 100 kg). Thereafter, most was rapidly lost (15 μCi after 20 min), the residues declining slowly (5 μCi after 60 min). Correspondingly, activity was recorded in the duct fume and was high in the hopper dust cake immediately after the melt. The hopper dust cake showed a prolonged radioactivity decline for subsequent non-radioactive casts of the kind also observed for the calibration melts (Paragraph 4.2.3.2.). The measured total activity (^{60}Co) of the metal product (200 kg) was 460 μCi after melt, corresponding to a specific activity of 23×10^{-5} μCi/g.

Table 8. Melting of SGHWR cover plates and Gundremmingen control rod drive tubes. Slag activity as a function of sampling time

Slag Sample N°	Sampling Time after addition of contaminated items (min)	Specific activity (μCi/g)	Total activity (μCi)
a) Seven plates (126 kg)			(Slag mass 100 kg)
1	10	8.0×10^{-4}	80
2	20	1.5×10^{-4}	15
3	30	1.0×10^{-4}	10
4	40	0.8×10^{-4}	8
5	50	0.6×10^{-4}	6
6	60	0.5×10^{-4}	5
7	70	0.4×10^{-4}	4
8	80	0.4×10^{-4}	4
Metal product (2,000 kg)		2.3×10^{-4}	460 (^{60}Co)
b) Three plates (54 kg)			(Slag mass 100 kg)
1	8	8×10^{-5}	8
2	16	5×10^{-5}	5
Slag changed			
3	22	2×10^{-5}	2
4	32	2×10^{-5}	2
5	38	2×10^{-5}	2
6	45	2×10^{-5}	2
Metal product (1,800 kg)		39×10^{-5}	702 (^{60}Co)
c) Control rod drive tubes			(Slag mass 100 kg)
1	5	1.3×10^{-4}	6.5
2	20	0.4×10^{-4}	2.0
Slag changed			
3	36	9.0×10^{-6}	0.045
4	46	1.2×10^{-5}	0.6
5	53	5.5×10^{-6}	0.275
Metal product (2,562 kg)		14.5×10^{-5}	371 (^{60}Co)

The three remaining cover plates were melted in the same manner as above. In order to utilize the initial retention of Cs in slag, the slag was removed as early as possible after melt-down. Table 8-b shows that about 60% of the Cs present at 8 min remained at the unavoidably delayed 16 min slag removal time.

Melting of KRB Gundremmingen BWR control rod drive tubes. These pieces had 6 to 8 mrad/h surface dose rate falling to 0.7 mrad/h at 1 m. Contamination varied widely in value. It was mainly ^{137}Cs with some ^{134}Cs and ^{60}Co. For the slag, radio-cesium amounts were slightly lower than for the previous melts, but the pattern for radio-cesium was repeated with a lower fraction of this (35%) fixed in the removed slag (Table 8-c).

4.2.3.2. Calibration melts

Two melts on metal plate material were carried out with known amounts of Cs radioactivity, applied as simulated contamination for the purpose.

The first melt was made in the 4 t electric-arc pilot plant. 25 µCi of ^{137}Cs as chloride was dispersed into a steel protective paint and applied to 250 kg of thick steel plate which was added to 2.5 t of furnace charge. There was no transfer of radioactivity to the steel at any time and no retention of radioactivity in the furnace slag; furnace fume activity was measurable but very low ($<10^{-10}$ µCi/ml). The hopper dust cake had a radioactivity concentration of 2.10^{-5} µCi/g (only 4% of the added radioactivity). Cesium chloride volatilizes below steel-making temperature, the Cs then re-depositing in the cool parts of extractor filter system with eventual flushing through to the external hopper.

The second melt was made in a 250 kg laboratory induction furnace. 0.1 µCi of ^{134}Cs as hydroxide was distributed over the surface of 2.5 kg of clean steel plate and melted with 250 kg of non-active scrap in an induction furnace. This form of application of Cs radioactivity to the plate was a significant step towards realism as regards the in-reactor conditions. Also, an acid slag was used. The contaminated plate was added with the steel liquid pool established and with complete slag cover. In this optimized melting procedure all Cs radioactivity stayed in the slag phase (Table 9).

These are melts with controlled additions of accurately known amounts of radioactivity for external dose quantification from various surface areas and section thicknesses in steel plate. Material from these melts

Table 9. Calibration melt induction furnace
(0.1 μCi of ^{134}Cs applied to 2.5 kg of steel plate)

Sample	Specific activity (μ/Ci/g)	Total mass (g)	Total activity (μCi)
Slag removed from furnace	2.8×10^{-5}	2,500	0.070
Slag remaining on furnace wall	2.8×10^{-5}	750	0.021
Steel	background	250,000	-
Dust	background	(30 ml)	-

also provided calibration standards for metal samples from the melts reported in Paragraph 4.2.3.1. There is a general lack of data or suitable formulae for calculation of dose rates external to steel with uniform dispersals of radioactivities. This particularly applies to thick sections where various self-shield effects have to be taken into account. Since reclaim of steel involving radioactivity dilution is a main prospect for the present work, it is important to verify such doseage for benefit versus detriment assessments.

Two melts were made in which 125 μCi each of ^{60}Co and ^{192}Ir were two-stage diluted by induction furnace into 500 kg each of steel ingots. These were rolled into various shapes and thicknesses of steel plate and provided the basis for various above surface dose rate measurements. These accurately known dilutions provided cross reference standards for activity concentration determination in all the product metal of the foregoing melts.

4.2.3.4. Radiological Safety

A complete schedule of radiological safety has accompanied the work. Material generating any dose rate in excess of 0.75 mrad/h or with removable contamination in excess of 10^{-4} μCi/cm² was handled in controlled conditions by qualified workers. All persons involved in furnace melting operations wore film badges and radiation dosemeters producing nil readings. The environmental air monitors running during all melting operations showed no measurable concentration of any radionuclide arising from the metal waste. Measurements in the fume in the furnace duct had a highest value of 6×10^{-9} μCi/ml. This contained fume therefore at no time had radioactivity levels greater than whole-time permissible environmental air.

4.2.3.5. Comments

Radioactive components, 1.65 t in all, have been reclaimed within UK legislation provisions in the several melts during 1983, without any radioactivity contamination or inhalation to members of the melting personnel involved in the work.

Surface monitoring, where complex shapes are involved, is extremely inaccurate. Thus two plates were judged equal in ^{60}Co, but the uniform melt products showed them to be in a ratio of 2:1. For ^{137}Cs the actual ratio was 10:1 where surface monitoring had indicated they were equal. The research work also showed:

- that ^{60}Co will almost certainly finish in uniform dilution in the metal;
- that Cs radioactivities can be made to stay substantially in slag, even in practical steel-making furnaces.

Note that all radioactivity concentrations have been determined by comparison with accurately dispensed amounts of radioactivity of the same kind and measured in a counting unit under the same specimen geometry.

4.3. Cobalt Removal from Steel by a Melting Process

Contractor: Seri Renault Ingénierie, Bois d'Arcy, France

Contract N°: DE-D-004-F Work Period: June 1981 - February 1983

4.3.1. Objective and Scope

The aim of this research was to assess, by means of theoretical studies and tests under non-radioactive conditions, the feasiblity of extracting cobalt and, possibly, nickel from steel (austenitic steel in particular). Such a process would be of particular interest if it reduced the radioactivity of the base metal to a level which makes it possible to recycle that metal.

4.3.2. Work Programme See Ref. 1, Paragraph 4.3.2.

4.3.3. Progress and Results

Complementary melting tests on a 10-20 kg scale were carried out, oxidizing the liquid iron by a special oxygen lance.

Results obtained gave a transfer to slag of about 15% of cobalt and 5% of nickel, confirming the unsatisfactory earlier results (Ref. 3, Paragraph 4.4.3.). A preliminary project study of an industrial facility for the recycling of 4,800 t per year of stainless steel was carried out. Such a plant should have:

- two 10 t capacity induction furnaces with an installed power of 3 000 kVA and an instantaneous melting rate of 2.4 t/h;
- two hoppers for raw materials, such as flux and stainless steel;
- weighing systems to allow adequate loading;
- a gantry crane of 20 t capacity, 12 m width and 24 m length;
- a cooled oxygen lance;
- a dosing system for quartz distribution;
- an activity control system to check the activity of residues to be removed from the plant.

The plant would be completely remote-controlled. The estimated total investment costs and operating costs (based on non-active conditions) are (mid 1983 values) $24x10^6$ FF and 3 760 FF/t, respectively.

4.3.4. Conclusions

The study has considered two different methods, i.e.: Electro Slag Refining process application and controlled oxidation with slags. The first method was abandoned after initial experiments at laboratory scale had shown that cobalt phosphides (compounds stabler than nickel or iron phosphides) are not soluble enough in slags to be of interest. The second method, consisting of melting stainless steel with slags under controlled oxidation, did not lead to satisfactory results either, mainly because chemical reactions rates were too low, giving a transfer to the slag of 10-20% of the cobalt. This leads to a high activity level of the slag, precluding its re-use

Progress on the controlled oxidation technique might be obtained with industrial-scale melts (300 to 1 000 kg) in appropriate installations.

4.4. Treatment of Concrete with Silicate Solutions to Prevent Dusting

Contractor: Taylor Woodrow Construction Limited, Southall, United Kingdom
Contract N°: DE-D-005-UK   Work Period: October 1981 - February 1984

4.4.1. Objective and Scope

The aim of this research is to study the use of coatings to prevent the dissemination of radioactive dust produced during the demolition of nuclear concrete structures. Whilst coatings will be considered broadly and in terms of cost effectiveness, silicate solutions and sodium silicate in particular, will have first consideration.

It is expected that treatment will be required of:

- rubble varying in size between that of a pea to the order of a cubic metre and covered with a layer of more or less adherent fine dust of particle size of the order of 0.3 mm and less;
- bulk quantities of dust, varying in size, but with a maximum of about 2 mm.

4.4.2. Work Programme: See Ref. 2, Paragraph 4.5.2.

4.4.3. Progress and Results

4.4.3.1. Material properties affecting dust treatment

Further investigation of the effects of pH, age and nature of concrete has followed the initial investigation of the effect of concrete type on the properties of the treated dust; three more concrete dusts, all containing Pulverized Fuel Ash (PFA) as a cement replacement, were investigated. Two of these concretes were mixes used in nuclear power stations.

The results of the testing, when compared to those obtained previously, again showed the effect of pozzolanic cement replacements in giving slower setting times than 100% Ordinary Portland Cement (OPC) concretes. These have been attributed to the lower lime contents of PFA concretes.

Penetration of silicate solution into dust. Damping of the dust with water to give a moisture content of about 2% was found to increase both the rate at which silicate solutions penetrate a bulk mass of dust and the final depth of penetration. It is suggested that this behaviour is due either to the moisture reducing the reaction rate of dust and silicate or to the easier penetration into the dust mass because of previous wetting. Variation in silica/alkali ratio had little or no effect on penetration rates and depths.

Binding of dust to rubble. The binding of dust to rubble (concrete pieces between 2.36 mm and 20 mm) was successfully achieved using a sodium silicate solution with a solids content of 33.4%. The effectiveness of binding was found to be sensitive to the solids content of the solution with the lower solids content solutions being less efficient binders.

The use of soluble silicate powders. Preliminary investigation into the use of soluble silicate powders, in a process of dry dust mixing followed by wetting, was successful in producing a hardened system. However, after trying the process with different concrete dusts, it became obvious that very accurate gauging of the dust with silicate powder and water would be required. For this reason, further work has not been attempted.

4.4.3.2. Process engineering

A pelletizing process to deal with bulk quantities of concrete dust has been assessed. A desk study of pneumatic collection and tranport was sufficient to identify design parameters and sources of equipment. Sufficient confidence in its feasibility was thereby generated. The most critical part of the development was the mixing and pelletizing of the dust with silicate solutions. A number of trials were done at the works laboratories of pelletizing machine makers.

A dilute phase suction air conveyor will be used to pick up the concrete dust and convey it to a separator from which the pelletizing machine will be fed. The return air flow taken back to the reactor vessel space will avoid the need for very fine filtration of the air used for conveying. The operating parameters of the air velocity, solid to air ratio, and pipe diameter have been considered and it is found that pneumatic conveying plant of the required size poses little difficulty. Feasibility of pick-up, transport and separation from the air-stream is not in doubt.

Development work on the pelletizing process was concentrated on existing machines working in the solid phase. A number of trials were carried out on existing trial equipment, and showing that a pelletizing process in the solid phase is feasible. Other potential methods of pelletizing would have required more original equipment and the development of this was not considered to be necessary. Equipment which was tested included:

- a pan agglomerator;
- an internal roller pelletizer;
- briquetting machines (with external indented rollers).

Either briquetting or a pan agglomerator were found to be suitable for this waste material. Samples of the pellets produced are being subjected to tests for compressive strength and abrasion resistance.

The feasibility of the pelletizing process has thus been demonstrated. Further detail will need to be established in a pilot plant in preparation for any reactor decommissioning programme.

4.5. Coating of Materials to Protect Against Corrosion, Fix Contamination and Avoid Powder Formation

Contractor: Nucleco S.p.A., Milan, Italy

Contract N°: DE-D-006-I   Work Period: July 1981 - June 1983

4.5.1. Objective and Scope

The dismantling of nuclear power plants produces large quantities of concrete and metal parts, with varying degrees of contamination. For such parts it would be useful to have suitable coating techniques to:

- fix surface contamination,
- prevent corrosion of metal parts,
- reduce dust during processing and handling of concrete.

The objectives of this research are:

- identification of low-cost and easily-applied coatings by means of experiments on commercially available products;
- assessment of the coating quality by measuring mechanical strength and corrosion and irradiation resistance;
- conceptual design of coating equipment.

4.5.2. Work Programme: See Ref. 1, Paragraph 4.4.2.

4.5.3. Progress and Results

Epoxy paints charged with polyolefines and anti-corrosive agent, were found to have excellent anti-corrosion properties and good applicability on steel and concrete, even with non-prepared surfaces.

The most appropriate epoxy paint found (Titania EP.23) was tested for its nuclear properties. Surface contamination containment, determined by dry smearing and water leaching, was found to be good, particularly for specimens coated with two layers of paint (total thickness 160 μm); the release of contaminant (natural uranium) in water started after five days of dipping. Radiation resistance, up to the absorbed dose of 600 Mrad, was found to be fairly good (visual control). Decontaminability of the irradiated and non-irradiated samples is poor.

Other paints with characteristics similar to Titania EP.23 and existing on the national market are as follows:

- an epoxy corrosion proofing zinc phosphate primer (Fonkor) superposed by a thick epoxy protective (Korall);
- an epoxy-bituminous paint (Eticoat) with good elasticity and adhesiveness.

These paints were characterized for chemical and mechanical resistance, giving an expected duration of their protective efficiency of 1 to 5 years.

4.5.4. Conclusions

As regards the practical application of epoxy paints in radioactive zones, the airless spraying method, where atomization of the product is obtained by ejection under high pressure, seems to be the most suitable for the coating of metals and concrete. Practical application of epoxy Titania EP.23 paint by an airless pump on surfaces (80x85 cm) contaminated with natural UO_2 powder has been demonstrated. A painting plant with intervention of operators in contaminated areas should be composed of drums, pneumatic transfer pumps, a mixing tank with pneumatic stirrer and a trailer-mounted airless pump.

4.6. Immobilization of Contamination on Metals by Coating with Thermosetting Resins

Contractor: CEA, CEN Grenoble, France

Contract N°: DE-D-007-F   Work Period: January 1983 - February 1984

4.6.1. Objective and Scope

This research is a follow-up of contract N° DE-D-003-F with Ecopol (Ref. 3, Paragraph 4.3.). The work performed in the basic contract showed the possibility of application of epoxy paints by an electrostatic technique on various metal samples (lids, plates, tubes). The average thickness obtained with one layer was 0.03 mm. The maximum thicknesses obtained were 0.07 mm with one layer, 0.14 mm with two layers and 0.18 mm with three layers. Tests on artificially contaminated metal samples showed that smear contamination could be reduced by a factor of 200 with one layer (0.03 mm) and by a factor of 10^4 with two layers (0.06 mm).

The proposed research work concerns tests with more representative, larger samples (metal tubes and plates) and the assessment of a new type of coating obtained by superposing different layers of thermosetting resins. Particular attention will be paid to the effectiveness of immobilization of contamination, mechanical resistance and to the diffusion coefficient of representative radio-elements.

4.6.2. Work Programme

Tests will be performed in an existing installation in order to optimize the projection parameters for multi-layer coats obtained by alternate electrostatic projection of pure thermosetting resins and by an airless paint projection of charged epoxy and elastomeric paints.

The mechanical properties will be assessed with particular regard to transport and handling conditions (resistance to shock, crushing, compression and friction). Two types of test will be performed, i.e.:

- tests on normalized samples, with regard to the general requirements for plastics and paints;
- tests on representative components, with regard to the standards for packages and transport.

The diffusion of radio-elements through pure epoxy paints will be assessed. A test cell will be constructed and tests will be performed on samples contaminated, e.g., with ^{137}Cs and ^{60}Co. The diffusion coefficient will be determined for various thicknesses and compared with the coatings used for the storage of low and medium level waste.

The range of application of the procedure will be assessed by tests on representative tubes and plates of up to 1 m length or 0.15 m diametre, and of various surface conditions (e.g., smooth, rough, slit surfaces and surfaces covered with rust, grease, paint, dust, etc.).

4.6.3. Progress and Results

The first step of research work performed, has been the development of a new technique for the projection of multi-layer coatings, to avoid dissemination and prevent diffusion of radionuclides on various types of surface of representative samples.

Tests were performed with several types of apparatus and metal sample. Several paints were tested, i.e.:

- epoxy paints suitable to fix contamination and prevent diffusion of representative radionuclides, with a thickness ranging from 70 to 500 µm;
- epoxy fiberglass charged resins for coatings with high mechanical characteristics, with a thickness of up to 1.9 mm;
- elastomeric coatings with a thickness of up to 10 mm.

Tests were carried out with an electrostatic paint device for the projection of pure epoxy paints and with an airless paint device for the projection of charged epoxy paints and elastomeric paints. The coatings were projected on tubes and flat sheets of a maximum length of 0.5 m or 0.15 m diameter and various surface conditions (cleaned, rusted, sanded, greasy and painted). Measurements were made in order to determine the thickness of coatings, the diffusion parameters with the radiotracers ^{60}Co and ^{137}Cs in special diffusion cells, the fixation factor of contamination and the mechanical resistance.

Diffusion tests showed, for 100 µm epoxy layers and after 200 days, a released activity, relative to the initial activity, of less than 2×10^{-5}

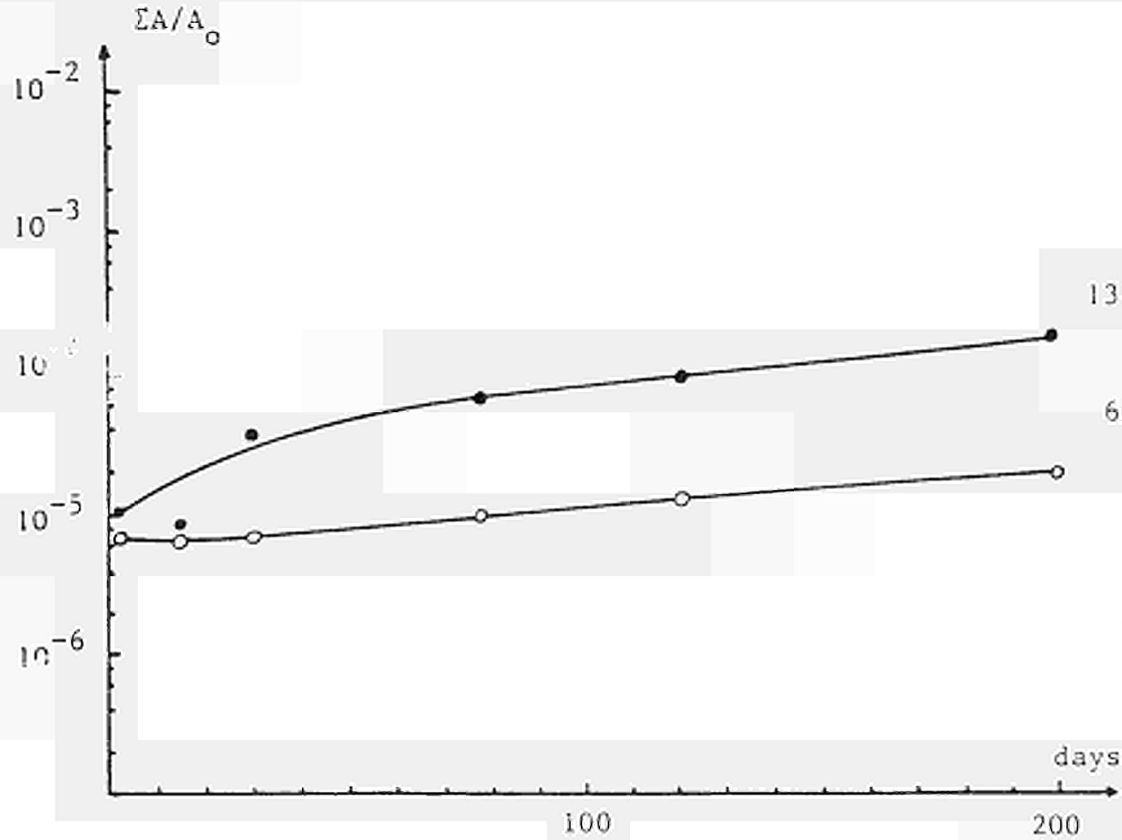

Figure 6. Cumulative activity released through 100 µm epoxy membranes

and 2×10^{-4} for ^{60}Co and ^{137}Cs, respectively (Figure 6). These results indicate that epoxy layers of the type studied would be suitable for temporary storage.

The second aim of the research work was to develop suitable coatings for protection against mechanical shocks. An elastomer coating was selected whose composition can be modified to appropriate hardness. Metal pieces were coated with a 5 mm thick coating using a commercial spray apparatus (airless). Tests to determine the appropriate thickness and composition of the product, as well as compressive, tensile, impact and scratch tests were initiated. The application of this paint process to real contaminated waste items up to 1 m long is planned.

4.7. Design of an Installation for the Melting of Radioactive Steel Waste

Contractor: Siempelkamp Giesserei GmbH, Krefeld, Germany.

Contract N°: DE-D-008-D   Work period: July 1983 - September 1984

4.7.1. Objective and Scope

There are presently in Germany about 3 000 t of radioactive metal waste from the refurbishing of nuclear power plants and 5 000 - 10 000 t from decommissioning. The objective of this research work is to design an installation for the melting of such waste, in order:

- to achieve a substantial reduction (by a factor of about five) of the number of waste drums and of the required storage capacity;
- to immobilize the radioactivity (e.g., reduce leach rates) and to improve the safety of transport;
- possibly, to reduce raw material requirements by recycling of the metal (e.g., use for type B waste casks and shieldings).

The melting installation will be designed for erection within the premises of nuclear power plants and for successive re-use in several plants. It would be remotely operated and capable of treating 1 000 t of waste metal within two months, producing disk-shaped ingots of a weight of 1.5 t and loading them into containers.

The following issues that are essential for the feasibility of the plant will be studied in particular:

- off-gas production and filtering;
- distribution of the radioactivity on the ingot, the furnace lining, the slag, the air ducts and the filter;

- construction and operation costs for the melting installation.

4.7.2. Work Programme

The following design and evaluation studies will be performed:

- specification of requirements;
- design of a special loading container capable of being tightly connected to the top of the melting furnace;
- dimensioning of the melting furnace;
- lay-out of the furnace venting and off-gas filtering system;
- lay-out of the room ventilation and air filtering system;
- detailed description of the operating sequence, including requirements for radiation protection and documentation;
- consideration of possible malfunctions and the necessary interventions;
- detailed estimate of the installation costs and operating costs.

The experimental work will include the following activities:

- preliminary off-gas measurements using existing melting furnaces to be provided with a suction cover;
- manufacturing of the special loading container and function testing;
- melting tests with non-radioactive metal feed, using the special loading container and an existing melting furnace; measurements of off-gas and aerosol quantities;
- manufacturing of a special re-usable casting die and casting tests; manipulation tests (transfer of the ingot from the die into a container by means of a crane).

4.7.3. Progress and Results

First investigations focused on the design of the exhaust system. Since data about the gas volumes produced during the melting operation are lacking, it was decided to melt metal in an existing crucible induction furnace of 8 t capacity, and to measure the gas volumes produced. For these tests, decontaminated scrap was purchased from the nuclear power stations of Gundremmingen (50 t), Obrigheim (30 t) and Biblis (15 t). The activity of these materials is below unrestricted release levels. Design of a provisional loading container was terminated, and test components were manufactured. Contamination by paint and synthetic materials have been simulated.

First melts were carried out in December 1983 with the main object of examining the chosen filter system. Approximately 20 t of decontaminated scrap metal (carbon steel grade St-37) were molten. These tests showed that safe enclosure of the material to be melted, i.e., the generation of vacuum in the melting chamber, could be guaranteed. The gas leaving the filtering system was free of particles. The non-nuclear contamination released during the melting consisted mainly of soot-like particles, the bulk of which was removed in the pre-filter. The filtering system will be re-dimensioned, taking account of the kind and volume of that contamination.

5. PROJECT N° 5: LARGE TRANSPORT CONTAINERS FOR RADIOACTIVE WASTE PRODUCED IN THE DISMANTLING OF NUCLEAR POWER PLANTS

Studies have shown that it is desirable to transport the radioactive waste resulting from dismantling of certain major reactor components in larger units than those currently used for other types of radioactive waste, in order to reduce the required amount of cutting and, consequently, the personnel radiation exposure and the decommissioning costs. The size and weight of the shipping units should at least be such as to take full advantage of the normal transport facilities.

5.1. System of Large Transport Containers for Waste from Dismantling Light Water Reactors and Gas-cooled Reactors

Contractor: United Kingdom Atomic Energy Authority, Winfrith, United Kingdom

Contract N°: DE-E-001-UK Work Period: September 1981 - December 1983

and

Contractor: Transnubel, Dessel, Belgium

Contract N°: DE-E-002-B Work Period: January 1981 - August 1983

5.1.1. Objective and Scope

The aim of this research is to carry out a system study to define the type of large containers needed to transport and/or dispose of the radioactive wastes produced in the dismantling of nuclear power plants within the European Community (Pressurized Water Reactors, Boiling Water Reactors and Gas-cooled Reactors).

5.1.2. Work Programme: See Ref. 2, Paragraph 5.1.2.

5.1.3. Progress and Results

5.1.3.1. Transport regulations

The review of international agreements, national legislation and codes of practice indicates the technical dominance of the 1973 edition of the IAEA Regulations for the Transport of Radioactive Materials, Safety Series N°6. The IAEA Regulations are so comprehensively integrated into the structure of international agreements, national legislation and codes of practice, that it would be technically and commercially wrong to establish designs which conflict with these Regulations. Although the transport categories were not drawn up in relation to the requirements of decommissioning, and may not be ideal, it may prove difficult to licence designs in various member states which do not conform to the IAEA Regulations. It must be recognized that the design, manufacture and approval of large decommissioning containers, particularly Type B, could require many years to bring to fruition and it is therefore recommended that the designs incorporate the requirements of the 1984 draft where these are more rigorous than the 1973 issue.

There is a growing acceptance of international agreements which have been derived in large part from the IAEA Regulations. In particular, for rail transport the International Regulations concerning the Carriage of Dangerous Goods by Rail (RID), for road the International Agreement for Carriage of Dangerous Goods by Road (ADR), for inland waterways the International Agreement on Carriage of Dangerous Goods by Inland Waterways (ADN), for countries bordering the River Rhine the regulations for Safe Transport on the River Rhine (ADNR), and for Sea Transport the Inter-Governmental Maritime Organization (IMO) recommendations.

The review of the national legislative requirements, recommendations and codes of practice for the member states of the EC, indicates that there are individual national requirements which must be considered at this design phase for proposed containers. According to the mode of transport to be adopted, non-nuclear limits especially on weight and size, as well as specific nuclear limits and conditions, must be considered. National considerations on testing and licensing must therefore be considered on a case basis according to the mode and rate of transport, prior to manufacture and application.

A set of recommendations has been formulated as regards the steps which need to be carried out in the design and licensing of new transport containers for decommissioning waste.

5.1.3.2. <u>Means of transportation</u>

The means of transportation have been studied and found to depend principally on the following factors:

- the dimensions of the load;
- the weight of the load;

- the type of packaging (use or not of a container; in the positive case, the type of container);
- the location of the power station;
- the location of the waste treatment facility and the final waste disposal site.

Waterway transportation is especially interesting for the individed large and/or very heavy components, as well as for containers with large dimensions and heavy weights. The transportation of such loads between the power stations and the transhipment location can be made by road providing that:

- the distance is not too long;
- the road is constructed for heavy loads;
- the transhipment location is equipped with an appropriate crane or a landing stage allowing roll-on/roll-off transport.

On the other hand:

- the average speed is very slow (2 km/h) - which rules out the use of returnable containers, taking into account the cost of hire;
- an inland waterway will have a certain number of locks. The dimensions of the locks can be a problem when special vessels or pontoons are to be used;
- the only regulation existing and in application in some countries is the ADNR;
- the competent authorities may be concerned since inland waterways can be the source of drinking water.

Road transport is convenient, providing the dimensions and the weights are in accordance with the regulations and for infrequent transport schedules. Bearing in mind these restrictions, it can be concluded that the consignment of large and heavy loads will always be considered as exceptional requiring approvals, possible police and health physics escorts, slow average speed (20 km/h) and expensive special truck hire. In consequence, it can be concluded that road transport of very large containers can only be performed over a short distance, for example between a reactor power plant and a transhipment area.

Rail transport is interesting providing the following conditions are fulfilled:

- the weight loading per axle of the railway wagon is not higher than the authorized limit. In most of the European countries, the limit is 20 t/axle;

- a railhead is available - if not, a short road haul has to be performed;
- the dimensions of the load are within the appropriate railway profile gauge, taking into account is length.

Rail transport has the following advantages:

- there is no specific regulation;
- the average speed is relatively high (60 km/h);
- transport is relatively cheap and whole trainload operation can be envisaged for decommissioning waste transport;
- no police escort.

It can be generally concluded that the rail option seems to be the most interesting solution, giving the possibility to transport components up to 110 or 130 t, depending upon the permissible axle loading, over a long distance in a relatively short time.

5.1.3.3. Transport costs

Two European countries were selected for study, namely Belgium and the United Kingdom and it was assumed that the model repository would be about 160 or 640 km from the hypothetical reactor being decommissioned. Typical costs were estimated for road and rail transport and, in the case of Belgium, inland waterway costs were evaluated.

To obtain an indicative cost for transporting the decommissioning waste from a reactor, it has been necessary to estimate the number of loads required using the generic types of container evolved during this study. The results have been evaluated for a 1 300 MW(e) European PWR and a 1300 MW(e) European BWR. In each it has been assumed that all the radioactive decommissioning waste is transported by rail except for a short 16 km road link between the reactor and a railhead.

The costs for both types of reactor are broadly similar, as shown in Table 10.

5.1.3.4. Radiological detriment

It has been assumed that the decommissioning waste will be transported principally by rail with short road hauls at each end of the journey. A 1981 survey of the transport of radioactive material within the UK indicates that the dose received by rail staff in the course of their duties

Table 10. Estimated cost for transporting decommissioning wastes from a 1 300 MW(e) PWR and a 1 300 MW(e) BWR to a repository

Mode	Distance (km)	Cost per journey (ECU)*	Number of loads	Number of journeys	Cost (ECU)*
1 300 MW(e) PWR					
a) road followed	16	265	106	106	28,000
by rail	160	20,400	106	16	326,000
				Total	354,000
b) road followed	16	265	106	106	28,000
by rail	640	47,800	106	16	765,000
				Total	793,000
1 300 MW(e) BWR					
a) road followed	16	265	101	101	27,000
by rail	160	20,400	101	15	306,000
				Total	333,000
b) road followed	16	265	101	101	27,000
by rail	640	47,800	101	15	717,000
				Total	744,000

* Conversion rates used: 1 ECU = 0.565 Pound Sterling = 46 Belgian Francs

is small. The collective dose equivalent for rail staff involved with transport of fuel flasks is about 0.5 man-rem/year. The critical staff (the slingers) are unlikely to receive a dose in excess of 10 mrem.

For the purpose of the study, a theoretical population distribution was assumed, which was representative of an average UK situation. It was also assumed that each of the reference designs of container is filled to its limiting capacity with decommissioning waste (i.e., filled to the limits permissible under the IAEA Transport Regulations).

The estimate of the increased risk from the transport of decommissioning waste given in Table 11 is relatively insignificant compared with normal risks encountered during day to day life.

Table 11. Health impact of waste transport to public per container movement (using road and rail)

Distance and route direction	Component of risk ($\times 10^{-8}$ health effects)*			Total collective risk
	Moving container		stationary container	
	Road	Rail		
160km North	0.2	18	66	84
160km East	0.5	9.6	55	65
160km South	4.6	10	85	100
160km West	1.5	16	29	46
640km North	0.5	39	141	180

* Risk Factor = 1.65×10^{-2} Sv^{-1}

5.1.3.5. Preliminary concept

Two generic types of large transport container are finally proposed:

- large returnable cast iron or cast steel square section Type B containers;
- large non-returnable steel reinforced concrete boxes conforming to the Low Specific Activity/Low Level Solid transport category.

Work on both of these designs has been carried out so that the general forms of such containers are established.

The recommendations from the overall study envisage taking these concepts to the stage of the preparation of manufacturing drawings, detailed costs, specifications, quality assurance and quality control documents. This set of tasks also involves the preparation of scale models for the performance of the regulatory tests prescribed by the IAEA.

5.1.3.5. Preliminary design data

The main components of a Type B container are:

- a body of nodular cast iron having externally a rectangular section of 1.7 x 1.7 m, equipped with trunnions;
- a cast stainless steel liner;
- a shielded inner lid in cast stainless steel;
- a stainless steel cavity liner;
- two rectangular shock absorbers at each end.

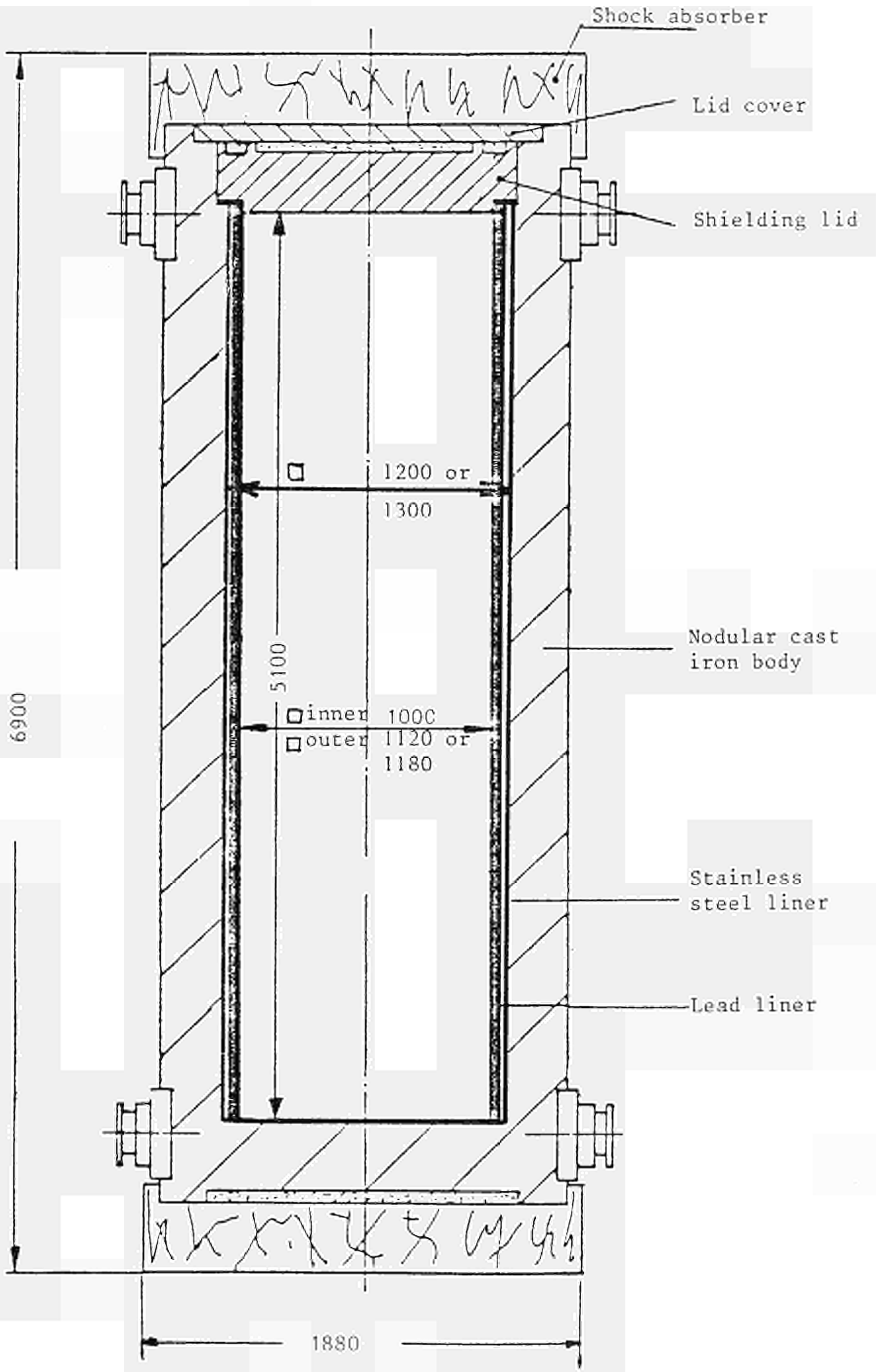

Figure 7. Large B(U) packaging for the transport of decommissioned reactor components (Type TNB 0152 and 0153).

An inner shell can be added or removed in function of the radiation level of the transported component (Figure 7). The shielding is provided in thicknesses of 350 mm of nodular cast iron, 250 mm of nodular cast iron + 50 mm lead (removable), 200 mm of nodular cast iron + 80 mm lead (removable). The cavity wall liner (5 mm) in stainless steel, as well as additional liners and the shielded lid, could be easily decontaminated. The weight of the container (loaded) would be a maximum of 110 t.

5.1.3.6. Evaluation of the fabrication cost

The fabrication cost of a new flask is assessed at 20 - 25 $x10^6$ BF, excluding the costs for the design, the tests and competent authority approval . A flask with a removable inner shielded shell would be more expensive than one fully made with nodular cast iron.

6. PROJECT N° 6: ESTIMATION OF THE QUANTITIES OF RADIOACTIVE WASTE ARISING FROM THE DECOMMISSIONING OF NUCLEAR POWER PLANTS IN THE COMMUNITY

The aim of this project is twofold: to estimate the quantities of radioactive waste arising from the dismantling of nuclear power plants, taking into account the time of production of the waste, and to define reference strategies for decommissioning.

Concerning the estimation of the waste quantities, large uncertainties are subsisting on the quantities of durably contaminated materials, especially of structures outside the primary circuit. Other projects of the research programme in progress should contribute information on this subject. While awaiting this information, Project No. 6 has been limited to the following areas:

- study of the activated masses of the biological shield;
- study of radiation levels related to the timing of dismantling of reactors after final shutdown.
- development of a methodology, to evaluate radiological consequences of the management of very low-level radioactive waste from decommissioning;
- review of techniques for measuring very low-level radioactivity.

6.1. Activation Products in the Biological Shield of the Lingen Reactor

Contractor: Kernkraftwerk Lingen GmbH, Lingen; Germany

Contract N°: DE-F-001-D Work Period: July 1981 - December 1982

6.1.1. Objective and Scope

The quantities of activated waste arising from the dismantling of the biological shield of reactors to be decommissioned are only approximatively known. Normally, existing data on the composition of the concrete relate to the mechanical and thermal behaviour of the material and, accordingly, are not precise enough for detailed computations of activation. Consequently, it is necessary to include pessimistic assumptions into activation calculations, leading to over-estimates of activation and of the quantity of concrete above the limit for unrestricted release.

The aim of this research work was to perform measurements on concrete samples from the biological shield of a closed-down nuclear reactor, directed to chemical elements making major contributions to the activity of the concrete. This analysis, together with neutron-flux and activation computations, will give a better knowledge on the volume of activated concrete in biological shields.

6.1.2. Work Programme: See Ref. 2, Paragraph 6.1.2.

6.1.3. Progress and Results

The experimental work was completed at the end of 1982 and the final report has been published (Ref. 12). Only the determination of trace elements in concrete is to be added (Table 12) to the results of measurements reported in the 1982 Report (Ref. 3, Paragraph 6.1.3.).

Table 12. Activation analyses for trace elements in concrete from the biological shield of KWL (Ca in %, all others in ppm)

Element	Sample 1	Sample 2	Sample 3
Calcium	4.5 ± 0.5	4.8 ± 0.5	5.7 ± 0.6
Cobalt	2.7 ± 0.3	3.0 ± 0.3	2.9 ± 0.3
Samarium	4.6 ± 0.5	5.7 ± 0.6	6.4 ± 0.6
Europium	0.75 ± 0.07	0.99 ± 0.10	1.04 ± 0.10
Barium	222 ± 22	247 ± 25	242 ± 24
Cesium	0.9 ± 0.1	0.9 ± 0.1	0.9 ± 0.1
Nickel	10.5 ± 1.0	8.9 ± 0.9	10.5 ± 1.0
Iridium	<0.003	<0.003	<0.003

6.1.4. Conclusions

In the planning the decommissioning of a nuclear power plant, knowledge of the activation depth of the biological shield is of particular importance. In order to avoid the need for expensive measuring programmes in future projects, the basic requirements for activation calculations using a computer programme have been established by means of detailed sampling and measuring.

Measurements and calculation are in reasonable agreement (Figure 8). Some of the preliminary results served to control and correct the input data, especially that of element weight fractions, and they also contributed to control and correct some measured values by repeating some measurements.

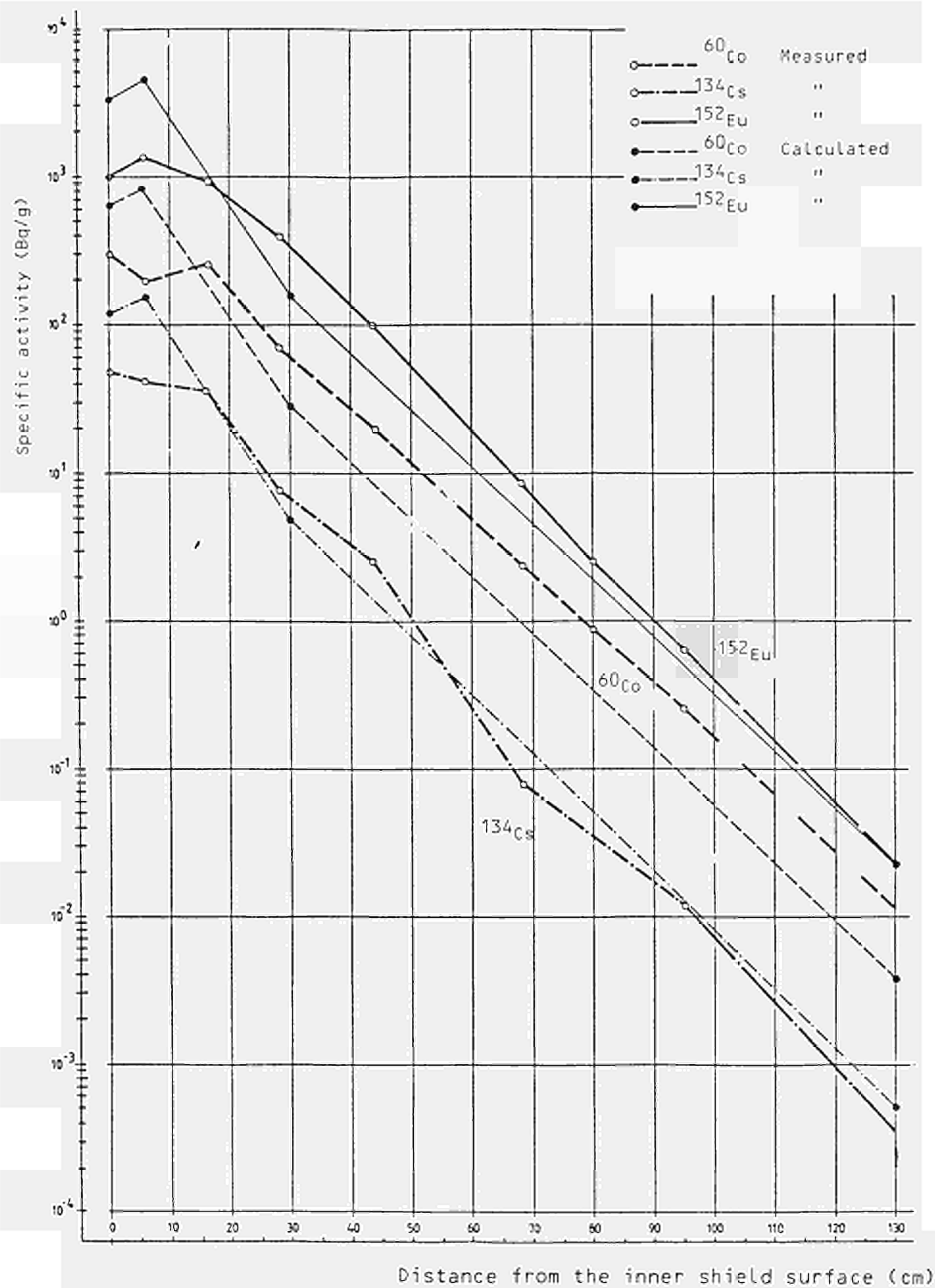

Figure 8. Gamma activity of the KWL biological concrete shield (at reactor core mid plane). Comparison of calculation and measurement.

To determine the region beyond which concrete activation is extremely low, it proved sufficient to calculate the activity at the mid plane of the reactor core. As far as the Kernkraftwerk Lingen is concerned, this region was found to end just before the outer concrete (primary concrete); consequently, only the inner concrete (secondary concrete), which is separated from the outer concrete by a styropor layer, is activated.

6.2. Activation Products in the Biological Shield of the KRB-A Reactor

Contractor: Kernkraftwerk Gundremmingen GmbH, Gundremmingen, Germany

Contract N°: DE-F-002-D Work Period: July 1981 - March 1983

This research work was linked to the work carried out by Kernkraftwerk Lingen GmbH (see Paragraph 6.1.) and had the same objective and work programme.

6.2.1. Progress and Results

The experimental work was completed in January 1983 and the final report has been published (Ref. 13). Table 13 shows the results of the activation analyses for trace elements not yet reported in the previous progress report (Ref. 3).

Table 13. Activation analysis for trace elements in concrete from the biological shield of KRB-A (Ca in %, from atomic absorption spectrometer; all others in ppm)

Element	Inner concrete		Outer concrete
	Sample 1	Sample 2	
Calcium	20.2	21.0	14.8
Cobalt	21.4	10.9	3.8
Samarium	1.88	1.86	2.4
Europium	0.37	0.39	0.35
Barium	140	102	30
Cesium	1.42	1.25	1.0
Nickel	44	230	11.4

There is good agreement between measurements and calculations. Figure 9 shows the curves of constant specific activity in a vertical section through the biological shield corresponding to the location of the samples taken. Differences between computations and measurements existing at low activation levels and in the lower parts are due to geometrical irregularities not taken into account in the calculations.

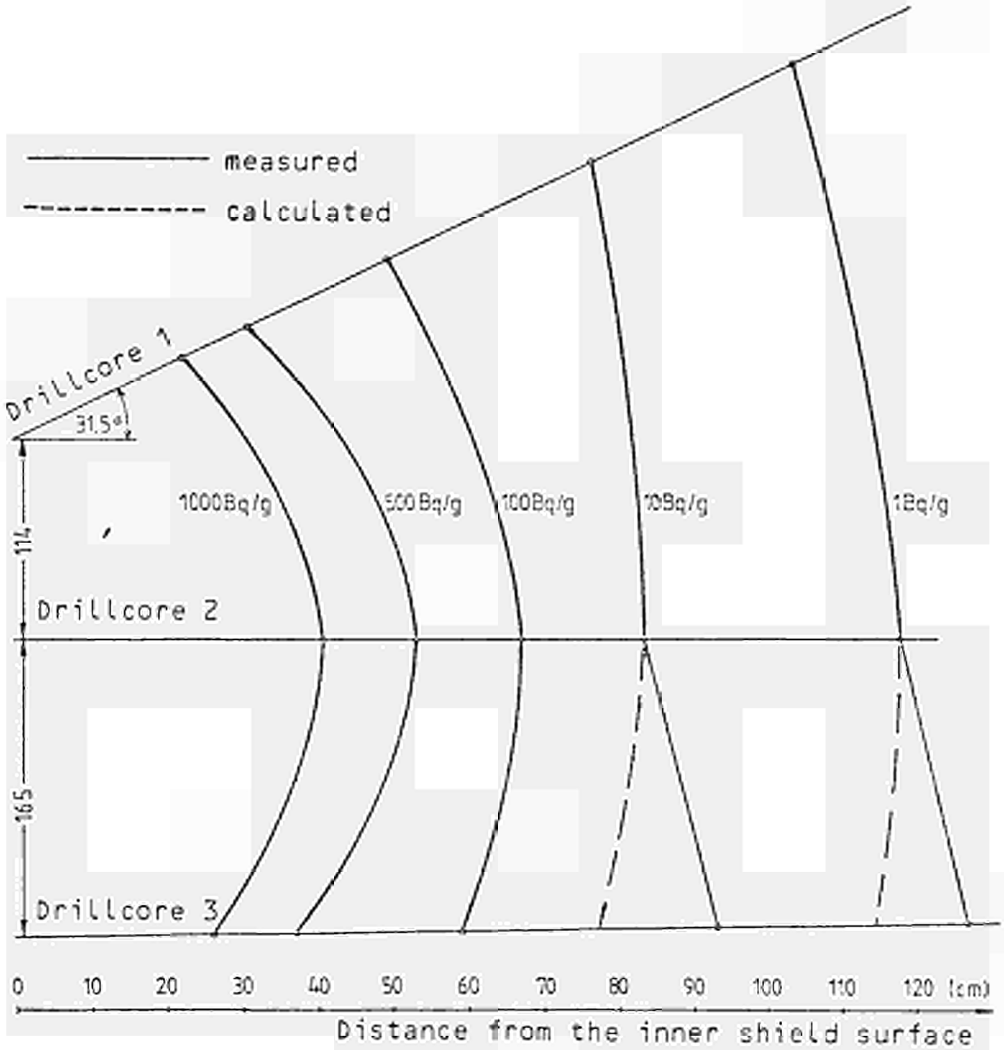

Figure 9. Curves of constant specific activity in a vertical section through the KRB-A biological shield.

6.2.2. Conclusions

The conclusions drawn for the KWL reactor (Paragraph 6.1.4.) are confirmed for the KRB-A reactor.

Good knowledge of the effective total of water content is essential for calculations in order to achieve good agreement with measurements.

6.3. Activation and Radiation at the Garigliano Reactor Pressure Vessel

Contractor: Nucleco SPA, Milano, Italy

Contract N°: DE-F-003-I Work Period: October 1981 - June 1983

6.3.1. Objective and Scope

The method of dismantling of the primary loop of a nuclear reactor will depend, even after a long decay time, on the gamma radiation of the structure. Consequently, it is important to know as well as possible the content in gamma-emitting isotopes of the reactor pressure vessel and its internals. The cobalt and niobium contents of the steel will be at the origin of the dominant gamma emitters at least during the first thirty years after the end of reactor operation.

The aims of this research were:

- to determine the content of cobalt and niobium in the reactor pressure vessel of the Garigliano plant;
- to evaluate the activation of the steel by calculation;
- to compare calculated and measured neutron flux data;
- to compare calculated values of activation with those measured on irradiated samples;
- to define the appropriate dismantling techniques as a function of decay time.

6.3.2. Work Programme. See Ref. 2, Paragraph 6.3.2.

6.3.3. Progress and Results

The research was completed in June 1983 and the final report has been published (Ref. 14).

The activation of the pressure vessel and the thermal shield of the Garigliano reactor were calculated. These calculations were checked by measurements on steel samples, which had been irradiated inside the reactor vessel. Very good agreement, within 10% of deviation, was found.

Starting from measured and calculated activation data, the dose rate in the center of the vessel was calculated as a function of decay time (Figure 10). A level of 15 mrem/h is reached after 200 years; this dose rate is mainly due to the thermal shield. For the vessel alone the same level would be reached after 90 years.

The techniques available for dismantling of the Garigliano reactor were examined, considering milling under water, explosive cutting, arc-saw cutting, plasma torch and oxyacetylene torch.

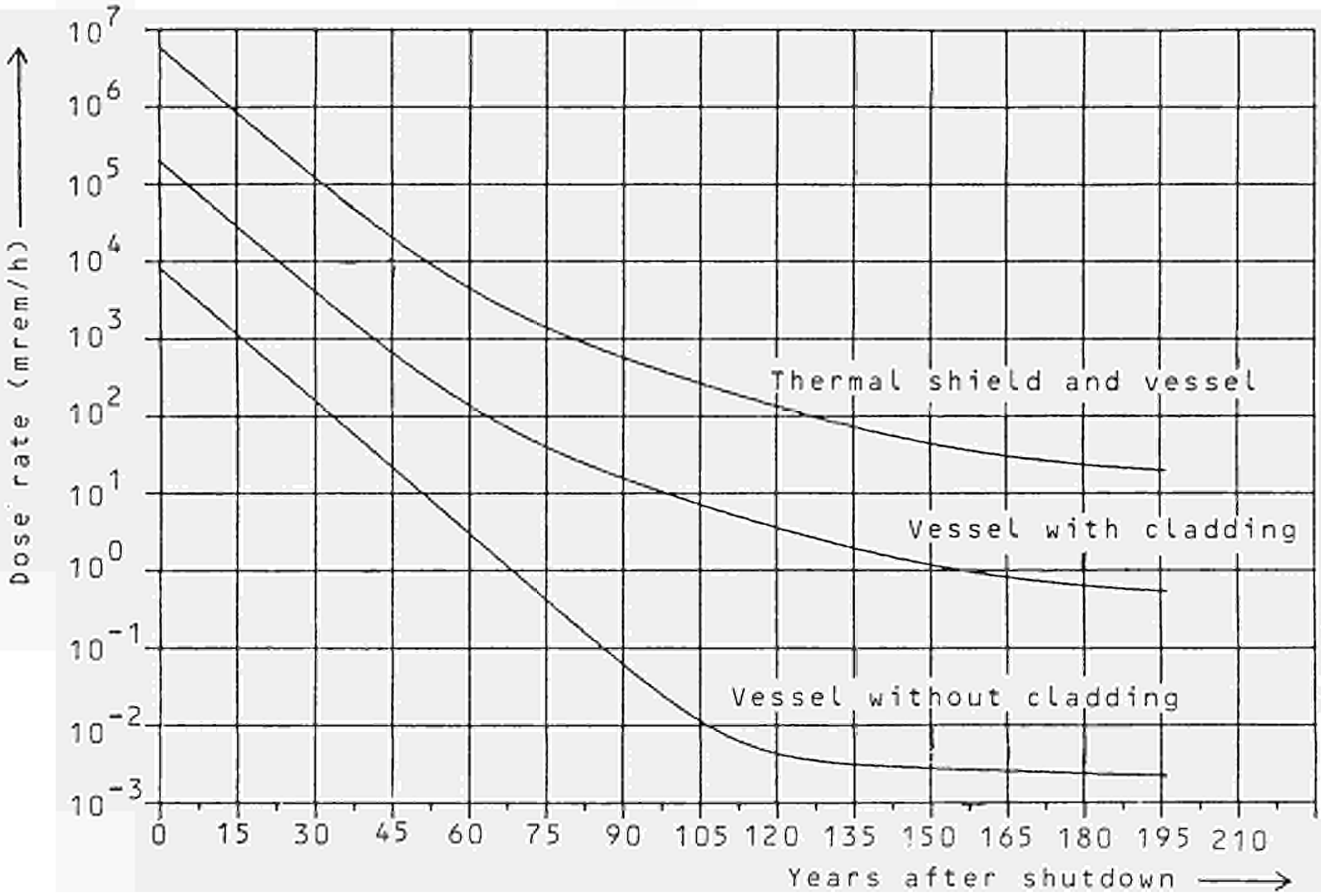

Figure 10. Evolution of the dose rate with decay time at the center of the Garigliano reactor vessel

6.3.4. Conclusions

The gamma activity of the vessel and the thermal shield of the Garigliano reactor at the core level were estimated by neutron-flux and activation calculations and by comparison with measurements. The contributions of the various gamma emitters to the activity at shutdown time are shown in Table 14. A variety of dismantling techniques have been shown to be applicable.

Table 14. Gamma activity (μCi/g) of the Garigliano reactor vessel and thermal shield at shutdown time

Isotope	Vessel base material	Vessel cladding	Thermal shield
^{60}Co	27	306	1280
^{94}Nb	9×10^{-4}	6.6×10^{-4}	2.1×10^{-3}
^{108m}Ag	4×10^{-3}	1.9×10^{-3}	7.5×10^{-3}
^{152}Eu	1.1×10^{-2}	4.1	16
^{154}Eu	1.3×10^{-3}	4.1×10^{-1}	1.5
^{155}Eu	2.0×10^{-5}	6.6×10^{-4}	9.7×10^{-3}

6.4. Trace Element Assessment of Low-alloy and Stainless Steels with Reference to Gamma Activity

Contractor: Imperial College of Science and Technology, London, United Kingdom

Contract N°: DE-F-004-UK Work Period: July 1981 - December 1983

6.4.1. Objective and Scope

The timing of dismantling of reactors after final shutdown has to take into account the radiation exposure to be expected during dismantling operations. Thus, knowledge of the radionuclides present in the vessels and internal components of reactors after the end of operation is needed.

The aim of the proposed research was to carry out experimental trace element assessment (for Mo, Ni, Co, Nb, Ag, Eu, Sm and Ho) of steel representative of those used in vessels and internals, with particular reference to potential gamma sources at long cooling times. In addition, using established data libraries, an assessment of long-lived activities was to be made for typical sample compositions in representative reactor fluxes.
A study of the variability of trace element concentrations in steels involves, inevitably, investigation of the detection limits of the various experimental techniques. For a number of elements this aspect dominated the investigation.

6.4.2. Work Programme: See Ref. 2, Paragraph 6.4.2.

6.4.3. Progress and Results

The research work has been completed and final report has been published (Ref. 15).

6.4.3.1. Analysis of samples

Reactor steel samples provided by a number of European organizations were analysed. They are listed in Table 15 : samples 1A-D are various CrNiNb 18/9 steels provided by Kraftwerk Union; sample 2 is vessel basis material from the Lingen pressure vessel; samples 3A and 3B correspond to parts of the Garigliano vessel; samples 4A and 4B and 5 are various AGR reactor steel types; sample 6 is a reinforcement steel; samples 7A and 7B are high niobium steels according to AISI 347; sample 8 is stock 18/8 stainless steel.

Table 15. Composition (weight %) of steel samples

N°	Steel sample	Mn	Cr	Mo	Ni	Nb	Co
1A	X10 CrNiNb 18/9	1.61	17.85	-	10.63	0.43	0.11
1B	" " "	1.53	18.03	-	10.53	0.62	0.18
1C	" " "	1.76	18.62	-	10.60	0.86	0.18
1D	" " "	1.53	18.15	0.11	10.10	0.33	0.05
2	KW Lingen P.V.	-	0.38	-	0.84	-	0.023
3A	Garigliano P.V.(L17)	-	-	-	-	-	-
3B	Garigliano P.V.(T22)	-	-	-	-	-	-
4A	AGR shield	0.70	0.05	0.02	0.10	-	0.011
4B	AGR fuel tie	<0.20	15-18	2.5-4.0	42-45	-	<0.25
5	AGR support strut	0.8-1.4	-	-	-	-	<0.03
6	Reinforcement	0.75	0.01	-	0.06	-	-
7A	AISI 347	1.41	19.28	-	9.28	0.61	-
7B	AISI 347	1.17	17.99	-	9.12	0.38	-
8	CrNiNb 18/8	1.60	17.58	0.36	10.55	0.92	-

The results of the investigations are summarized in the Table 16.

Molybdenum in steel yields 3500-year ^{93}Mo; this activity has, however, been shown in this work and elsewhere to be not a significant source of gamma activity. Activation analysis employing the 2.76-day ^{99}Mo activity was found to be satisfactory.

Levels of niobium in steel have received attention due to 20,300-year ^{94}Nb activity which decays with emission of 871 and 703 keV gamma rays, and due to past difficulties in establishing niobium concentrations. Selected results of the analysis are shown in Table 17. For the range of steels

Table 16. Analyses of steel samples

Sample N°	Mo %	Nb ppm	Co %	Ni %	Ag ppm	Sm ppb	Eu ppb	Ho ppb
1A	0.414 ± 0.017	3535	0.107 ± 0.007	10.32 ± 0.13	2.4 ± 0.1	< 33 ± 4	< 35 ± 11	21 ± 17
1B	0.382 ± 0.017	4968	0.184 ± 0.007	10.84 ± 0.14	1.7 ± 0.1	< 28 ± 4	< 39 ± 6	< 39 ± 13
1C	0.367 ± 0.017	7544	0.379 ± 0.009	11.08 ± 0.12	1.1 ± 0.1	-	-	-
1D	0.097 ± 0.008	2819	0.053 ± 0.008	9.38 ± 0.11	0.3 ± 0.1	-	-	-
2	0.652 ± 0.017	9	0.0213 ± 0.0012	0.762 ± 0.029	0.7 ± 0.1	< 41 ± 7	< 31 ± 6	< 18 ± 17
3A	0.577 ± 0.016	6	0.0187 ± 0.0007	0.752 ± 0.026	1.4 ± 0.1	15 ± 24	43 ± 42	< 24 ± 25
3B	0.697 ± 0.017	6	0.0178 ± 0.0007	0.732 ± 0.026	0.7 ± 0.1	-	-	-
4A	0.063 ± 0.009	6	0.0096 ± 0.0008	0.269 ± 0.009	1.6 ± 0.1	< 16 ± 1	< 7 ± 1	< 19 ± 17
4B	3.144 ± 0.034	41	0.205 ± 0.017	37.34 ± 0.34	0.7 ± 0.1	-	-	-
5	0.061 ± 0.008	6	0.0191 ± 0.0016	0.287 ± 0.008	0.9 ± 0.1	< 21 ± 2	< 11 ± 1	< 21 ± 12
6	0.080 ± 0.014	5	0.0253 ± 0.0042	0.047 ± 0.015	0.9 ± 0.1	-	-	-
7A	0.141 ± 0.007	5306	0.148 ± 0.007	8.92 ± 0.11	0.6 ± 0.1	-	-	-
7B	0.218 ± 0.008	3306	0.134 ± 0.006	8.85 ± 0.10	0.5 ± 0.1	-	-	-
8	0.787 ± 0.007	7710	0.159 ± 0.007	10.01 ± 0.13	4.5 ± 0.2	-	-	-

available, inductively coupled plasma (ICP) spectrometry was found to be satisfactory with a detection limit of about 5 ppm. The general level of niobium concentrations is seen to be lower than had been assumed in the past when setting nominal concentrations. Presence of significant niobium levels in samples 1A-D and samples 7A, 7B and 8 provided a valuable test of the methods for niobium.

No difficulties were encountered in cobalt analyses.

Studies by other workers have confirmed the potential significance of silver as a trace element due to 127-year ^{108m}Ag with gamma rays of 434, 614 and 723 keV. In the present work both activation analysis and flameless atomic absorption have been undertaken. Activation analysis employed the 24.6-second ^{110}Ag activity counted using a cyclic activation system on the Imperial College reactor. This technique was shown to have a detection limit of approximately 1 ppm and analyses were only possible for three of the available samples. However, flameless atomic absorption proved to have a detection limit of approximately 0.1 ppm. The general consistency of

Table 17. Niobium analyses (ppm)

Steel	Sample N°	ICP single dissolution	ICP 2-stage dissolution	Other methods
Low-alloy vessel steels	2	9	6	-
	3A	6	24	< 183^{+}
	3B	6	47	< 223^{+}
Miscellaneous	4A	6	< 5	
	4B	41	55	
	5	6	< 5	
	6	< 5	< 5	0.35+0.06°
Other stainless steels	316	36	41	
	321	38	82	
Niobium-bearing steels	1A	3530	4220	4300*
	1B	4970	5910	6200*
	1C	7540	8670	8600*
	1D	2820	3220	-
	7A	5310	6510	6100*
	7B	3300	3810	3800*
	8	7710	8580	9200*

+ : Neutron activation analysis (NAA)

° : Photon activation analysis (PAA)

* : Manufacturers specification (MS)

silver levels (lying between 0.3 and 4.5 ppm) is strong evidence that these levels may be anticipated in other steels of this general type.

Nickel was determined by activation analysis using fast-neutron activation yielding ^{58}Co with a 71-day half-life. Satisfactory accuracy was obtained for all samples.

Although normally present in very low concentrations in materials, the rare-earth elements merit careful attention due to high activation cross-sections and long half-lives. Activation of europium leads to 13.3-year ^{152}Eu and 8.8-year ^{154}Eu. Activation of samarium does not yield any significant samarium gamma emitters but does lead to the production of ^{154}Eu and, to a much lesser degree, ^{152}Eu and ^{155}Eu. Activation of holmium yields with a somewhat lower activation cross-section 1200-year ^{166m}Ho. Competition from other steel constitutents, combined with the low concentrations, prevented the detection of the rare earths with conventional activation analysis. In order to remove these competing activities, a solvent extraction procedure was established using duplicate ion-exchange columns and with the necessary use of standards and blank samples in order to ensure recovery of the rare earths and their proper counting after activation analysis. The general conclusion can be drawn that the upper limits for the samples studied were approximately 25 ppb, 30 ppb and 25ppb for samarium, europium and holmium, respectively.

6.4.3.2. Calculations of induced activity

In order to examine the significance in terms of relative potential gamma sources of the detection limits and the concentrations which have been observed in this work and to examine the influence of burn-up upon samarium and europium activities, the ORIGEN II computer code was used in conjunction with a light-water reactor core spectrum averaged cross-section data set. It should be noted that for fully realistic calculations of induced activities, and calculations of effective dose equivalent, proper account would have to be taken of the spatial distribution of both neutron spectrum and total flux, the detailed composition of various components and the self-shielding afforded by structural materials. In Figure 11 the potential gamma source (expressed in gamma Curie MeV per tonne) is shown as a function of time after shutdown following 30-year irradiation in a total flux of 10^{13} n cm^{-2} s^{-1}. Representative compositions for type 304 steel of 10% nickel, 500 ppm molybdenum and cobalt at concentrations of 1000 ppm and

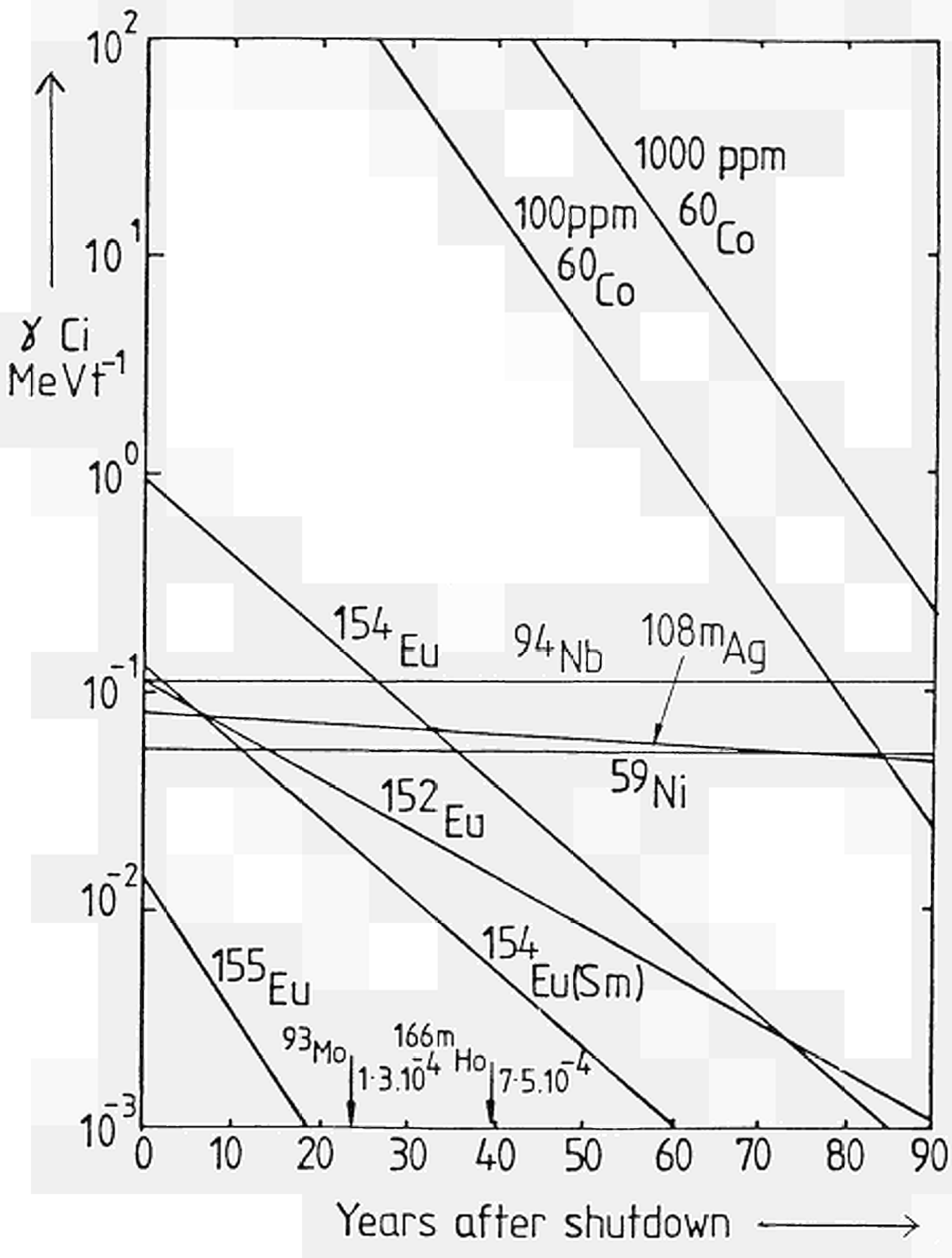

Figure 11. Potential gamma sources for type 304 steel following 30-year irradiation in PWR core spectrum at a flux of 10^{13} n cm^{-2} s^{-1}.

100 ppm were used. On the basis of the present survey, concentrations of 10 ppm niobium, 1 ppm silver, 25 ppb samarium, 30 ppb europium and 25 ppb holmium (the last three being upper limits) were employed. At cooling times where cobalt activity has declined sufficiently, ^{108m}Ag and ^{59}Ni dominate with ^{94}Nb at a somewhat lower level. In these comparisons the lower effective gamma-ray energies from ^{59}Ni should be borne in mind.

6.4.4. Conclusions

A range of steels, restricted in number but representative of a variety of reactor components, have been analysed for trace elements. It has been necessary to employ a variety of analytical techniques to determine all elements specified in the contract objective. For the steels studied, activation analysis has proved satisfactory for Mo, Ni and Co; ICP was the appropriate technique for Nb while flameless atomic absorption yielded analysis of Ag for all samples. An ion-exchange technique was used successfully to concentrate the rare-earth elements prior to activation analysis, but, for the full potential to be realized, would have to be used as a post-irradiation radiochemical technique prior to counting.

The silver content was remarkably consistent in all samples. For steels without a manufacturer's nominal specification, levels of niobium found tended to be lower than assumed in the past for stainless and low-alloy steels. Calculations of potential gamma-ray sources in a nominal type 304 steel with trace-element concentrations representative of those found in this study confirmed the likely significance of silver, together with nickel and niobium, once the ^{60}Co activity has ceased to be dominant. For samarium and europium concentrations less than about 30 ppb, their contributions to gamma activity should not be significant compared with other sources for this steel type.

6.5. Determination of Trace Elements in Concrete Samples from Various Nuclear Power Plants

Contractor: Commissariat à l'Energie Atomique, Laboratoire Pierre Süe, CEN Saclay, France

Contract N°: DE-F-005-F Work Period: January 1982 - December 1983

6.5.1. Objective and Scope

There is a large variation of the content of elements leading to long-lived radionuclides in the biological shield of nuclear reactors; its knowledge would allow to calculate more exactly the quantities of concrete to be handled and their decay time. The objective of this research has been to explore the range of trace-element contents in biological shield concretes, analysing samples from various reactors in the European Community.

6.5.2. Work Programme: See Ref. 3, Paragraph 6.5.2.

6.5.3. Progress and Results

The work has been completed and the final report has been published (Ref. 16).

21 concrete samples from the biologial shields of 15 reactors in four countries of European Community were analysed using neutron-activation analysis. The elements of special interest as potential sources of significant amounts of long-lived radionuclides, were Cl, Ca, Ni, Eu and Sm. Additionally, the following elements were determined: As, Au, Ba, Ce, Cr, Cs, Fe, Hf, Ir, La, Lu, Nd, Rb, Sb, Sc, Ta, Tb, Th, U, Yb and Zr. For the determination of Cl, Ni and Nb, post-irradiation radiochemical separation techniques had to be developed. The other elements were measured directly by gamma spectrometry. Table 18 shows the results of activation analyses for the above-mentioned elements of special interest and for Mo and Cs.

6.5.4. Conclusions

Activation analysis has proved to be an accurate method for the determination of trace elements in concrete.

The analyses carried out on samples from 15 reactors allow to give ranges of the contents of source elements for activation which are normally to be expected in biological shield concretes. For the most relevant source elements these ranges are as follows:

- calcium: 3-25 wt% (average: 13 wt%);
- chlorine: 16-500 ppm (average: 110 ppm);
  <120 ppm for all reactors but two;
- nickel: 6-26 ppm (average: 11 ppm);
- cobalt: 2-20 ppm (average: 5 ppm); <8 ppm for all reactors but one;
- niobium: <50 ppm; <25 ppm for all reactors but one;
- samarium: 1.3-6 ppm (average: 3.2 ppm);
- europium: 0.3-1.2 ppm (average: 0.6 ppm).

Table 18. Analyses of concrete from various reactors (accuracy $\pm$10% where not otherwise specified)

Reactor	Ca wt%	Cl ppm	Ni ppm	Co ppm	Nb ppm	Sm ppm	Eu ppm	Mo ppm	Cs ppm
KKP-2	8.8	57	14.9	5.1	8.8 $\pm$ 5.4	2.1	0.4	7.1	3.9
KKG(Normal)	9.4	97	17.8	6.4	4.9 $\pm$ 3.6	2.6	0.53	5.1	1.6
KKG(Hematite)	3.4	40	7.2	2.3	**	1.3	0.35	6.2*	0.4
KKI-2	11.3	87	8.5	3.8	9.0 $\pm$ 4.2	1.3	0.31	4.8	1.1
KWG	9.9	117	11.6	4.5	<6	2.5	0.49	5.4	2.1
KBR(Normal)	8.1	75	6.1	3.2	6.0 $\pm$ 4.4	3.4	0.82	6.2	1.0
KBR(Hematite)	4.6	42	11.4	4.7	**	1.4	0.4	5.3*	0.26
KWS	18.1	55	26.0	20.0	<5.8	5.6	1.18	10.8	4.3
KKU	5.8	74	10.0	3.7	<5.1	2.3	0.49	3.7	1.6
KWL(sample 1)	4.6	52	10.5	2.7	10.7 $\pm$ 3.5	4.6	0.75	3.1	0.9
KWL(sample 2)	4.8	40	8.9	3.0	9.9 $\pm$ 3.8	5.7	0.99	3.9	0.9
KWL(sample 3)	5.7	36	10.5	2.9	17.3 $\pm$ 5.2	6.4	1.04	3.8	0.9
KRB-A	14.8	32	11.4	3.8	12.2 $\pm$ 3.5	2.4	0.35	2.5	1.0
Garigliano(1)	22.2	317	10.0	7.1	24.9 $\pm$ 10.5	5.8	1.07	11.1	5.0
Garigliano(2)	22.3	326	10.2	7.8	17.3 $\pm$ 9.1	5.4	1.0	11.1	3.8
Garigliano(3)	21.8	493	8.2	5.4	22.7 $\pm$ 9.9	5.7	1.03	10.8	3.1
Oldbury	23.2	208	11.1	4.9	12.8 $\pm$ 5.3	2.3	0.50	5.5	0.66
EL2	24.8	27	6.1	3.2	<3.2	1.9	0.41	4.1	1.5
EL3	23.0	22	6.6	3.2	47 $\pm$ 11	2.3	0.49	3.9	1.3
TRITON	16.9	16	8.9	3.5	<7	1.8	0.33	4.5	1.9
G1	12.4	77	13.3	2.9	<3.5	1.6	0.35	5.3	2.3

*in soluble part ** Fe gives an interference

6.6. Methodology for Evaluating Radiological Consequences of the Management of Low-level Radioactive Waste from the Dismantling of Nuclear Plants

Contractor: National Radiological Protection Board, Chilton, United Kingdom

Contract N°: DE-F-006-UK Work Period: September 1982 - June 1984

and

Contractor: Commissariat à l'Energie Atomique, CEN Fontenay-aux-Roses, France

Contract N°: DE-F-007-F Work Period: September 1982 - June 1984

6.6.1. Objective and Scope

The aim of this joint study is to develop a methodology for the evaluation of individual and collective doses and of the dose commitment resulting from suitable management modes for very low-level solid radioactive waste arising from the dismantling of nuclear power plants.

6.6.2. Work Programme: See Ref. 3, Paragraph 6.6.2.

6.6.3. Progress and Results

6.6.3.1. Definition of reference waste characteristics

A study of the relation between the quantity and the activity level of contaminated wastes from various sources indicated that the waste quantity decreases by a factor of two if the specific activity is increased by a factor of ten. It was then assumed that this relation holds for the wastes arising from the dismantling of nuclear power plants.

Concerning steel, the very low-level waste considered in this study could come from auxiliary circuits, storage pools and racks, effluent and liquid waste systems, crane runways, stairs, etc.. The primary circuit of a Pressurized Water Reactor, even after decontamination, is not considered to belong to this category.

Table 19. Classification by the contamination level of steel and concrete waste from the dismantling of a 1000 MWe Pressurized Water Reactor (5 years after shutdown)

Contamination level (Bq/cm^2)	Steel Mass (t)	Steel Activity (Bq)	Concrete Mass (t)	Concrete Activity (Bq)
$>10^4$	250		250	
10^3-10^4	250	1×10^{12}	250	1×10^{10}
10^2-10^3	500	2×10^{11}	500	2×10^9
10-10^2	1000	4×10^{10}	1000	4×10^8
1-10	2000	8×10^9	2000	8×10^7
10^{-1}-1	4000	1.6×10^9	4000	1.6×10^7
10^{-2}-10^{-1}			8000	3.2×10^6

The average ratio of contaminated surface area to mass was assumed to be 100 m^2/t for steel and 1 m^2/t for concrete. Some heavy steel components would need special consideration.

Taking into account these assumptions, the waste arisings shown in Table 19 were defined as reference data for the purpose of this study.

On the basis of a review of relevant information and taking into account radionuclides found in reactor coolants and effluents, the reference radionuclide composition shown in Table 20 was defined.

Table 20. Radionuclide composition of surface contamination

Radionuclide	Half-life (years)	Activity for various decay times (arbitrary units)		
		5 years	25 years	100 years
^{55}Fe	2.86	1×10	7.8×10^{-2}	1×10^{-9}
^{60}Co	5.27	1×10^{2}	7.2	3.8×10^{-4}
^{59}Ni	7.5×10^{4}	1×10^{-2}	1×10^{-2}	1×10^{-2}
^{63}Ni	1×10^{2}	2	1.7	1
^{90}Sr	2.81×10	5	3.1	4.8×10^{-1}
^{93}Zr	1×10^{6}	1×10^{-4}	1×10^{-4}	1×10^{-4}
^{94}Nb	2×10^{4}	1×10^{-3}	1×10^{-3}	1×10^{-3}
^{99}Tc	2×10^{5}	1×10^{-3}	1×10^{-3}	1×10^{-3}
^{125}Sb	2.8	1×10	7×10^{-2}	6×10^{-10}
^{134}Cs	2	4	3.9×10^{-3}	2×10^{-14}
^{135}Cs	2.9×10^{6}	3×10^{-5}	3×10^{-5}	3×10^{-5}
^{137}Cs	3.01×10	1×10	6.3	1.1
^{147}Pm	2.6	1×10	5×10^{-2}	1×10^{-10}
^{151}Sm	9×10	1×10^{-2}	8.6×10^{-3}	4.8×10^{-3}
^{238}Pu	8.77×10	7.5×10^{-2}	6.2×10^{-2}	3.5×10^{-2}
^{239}Pu	2.4×10^{4}	9×10^{-3}	9×10^{-3}	9×10^{-3}
^{240}Pu	6.6×10^{4}	1.2×10^{-2}	1.2×10^{-2}	1.2×10^{-3}
^{241}Pu	1.44×10	2.1	7.9×10^{-1}	2.1×10^{-2}
^{241}Am	4.32×10^{2}	2.5×10^{-2}	6.6×10^{-2}	7.6×10^{-1}
^{244}Cm	1.8×10	3.5×10^{-2}	1.5×10^{-2}	8.4×10^{-4}
Total		1.5×10^{2}	1.9×10	2.8

The activation of the biological shield concrete was calculated. Five years after shutdown, only the inner 1m layer of the shield is above an activity level of 10 Bq/g. The remainder of the concrete contains a total activity of 10^9 Bq, composed mainly of ^{152}Eu, ^{154}Eu, ^{60}Co, ^{36}Cl and ^{41}Ca.

6.6.3.2. Definition and radiological evaluation of disposal systems

Definition of the disposal systems under consideration was completed, and model parameters relevant to the calculation of radiological impact were determined. Calculations have been made of the radiological impact of disposal of selected radionuclides at a municipal landfill site and on the seabed in coastal waters. The radionuclides considered so far were chosen on the basis of those identified in concrete and steel arising from the decommissioning of a Magnox reactor, and are listed in Table 21. The doses arising from the disposal of 10^{12} Bq of each were evaluated. Among the radionuclides considered, only ^{14}C is likely to become globally dispersed following landfill or seabed disposal. Therefore, doses to the EC population arising from global circulation of ^{14}C were also calculated.

Table 21. Radionuclides considered in radiological evaluations

Radionuclide	Half-life (years)	Occurrence
^{14}C	5.69×10^3	steel and concrete
^{55}Fe	2.70	steel and concrete
^{60}Co	5.27	steel and concrete
^{59}Ni	7.50×10^4	steel
^{63}Ni	1.00×10^2	steel
^{94}Nb	2.03×10^4	steel
^{108m}Ag	1.27×10^2	steel
^{41}Ca	1.03×10^5	concrete
^{151}Sm	9.00×10^1	concrete
^{152}Eu	1.33×10^1	concrete
^{154}Eu	8.60	concrete

Municipal landfill disposal

Wastes are assumed to be placed directly into a simple trench sited in glacial sand overlying a bedrock of sandstone. The water table is assumed to lie about 10 m below the base of the landfill. The landfill is therefore surrounded by permeable material and is well drained. Any infiltrating rainwater will tend to move through the wastes and then down through the underlying unsaturated zone. Once water reaches the water table, it tends to move downslope towards the stream.

Two mechanisms for release of radionuclides have been considered: contact by infiltrating rain-water or groundwater, and human intrusion for building purposes. Transport of activity in groundwater leads to contamination of farm produce and the stream adjacent to the site. Individual and collective doses to members of the public were calculated for various foodstuff and drinking water pathways. Human intrusion for building purposes leads to exposure during excavation of the site via external gamma irradiation and inhalation of contaminated dust. In this case only individual doses were calculated.

For ingestion pathways, and on the basis of disposal of unit activity of each radionuclide, ^{14}C gives rise to much the largest individual dose. Several of the radionuclides decay to very low levels before release from the geosphere to the stream or soil zones occurs. For doses following intrusion, the external gamma dose is more important than inhalation of dust. However, for ^{60}Co, ^{152}Eu and ^{154}Eu these doses are much reduced within 100 years.

^{14}C is also most important in collective dose commitments. However, there are considerable uncertainties associated with parameters for ^{14}C.

Coastal seabed disposal

The model used for the dispersion of radionuclides following seabed disposal was modified to take account of return of activity from bottom sediments. However, the parameters adopted in the sediment model should be regarded as tentative and are subject to revision. Maximum individual doses depend strongly on the volume of, and annual exchange rate from, the local compartment. The parameters adopted to date correspond to a sheltered coastal site. Collective doses were calculated on the basis of EC fish catches, etc..

Calculations were made for disposal into local water compartments in the southern Irish Sea and the northern North Sea. Results were not substantially different for the two disposal sites. For disposal of unit activity the most significant individual doses arise from the gammaemitters via the external gamma at beach pathway during the year or so after disposal. However, there may be no coastline within the local box, in which case the corresponding dose at the coastline of the southern Irish Sea would be about a factor of 30 lower. For seafood ingestion pathways ^{60}Co, ^{108m}Ag and ^{14}C are most important. Inhalation of suspended sediment and marine aerosols give rise only to very small doses. Return of radioactivity from bottom sediment has no effect on maximum individual doses. ^{14}C also gives rise to a much higher collective dose commitment than the other radionuclides, most of this arising from global circulation. However, if commitment is truncated at 100 years, ^{108m}Ag, ^{94}Nb, ^{60}Co and ^{152}Eu are also important.

Discussion

For both seabed and land disposal options ^{14}C and ^{60}C are the most significant radionuclides on the basis of disposal of unit activity. Since they are also likely to be among the most abundant, they will be major contributors to the total doses. ^{41}Ca, ^{152}Eu and ^{151}Eu are also likely to be of interest.

6.7. Review of Techniques for Measuring Very Low-level Radioactivity in Relation to Decommissioning

Contractor: GKSS-Forschungszentrum Geesthacht GmbH, Germany

Contract N°: DE-F-008-D Work Period: October 1982 - May 1984

and

Contractor: Commissariat à l'Energie Atomique, CEN Fontenay-aux-Roses, France

Contract N°: DE-F-009-F Work Period: September 1982 - December 1983

6.7.1. Objective and Scope

The objective of this joint study is to review the existing techniques for measuring and controlling radioactivity at a level near the limits for unrestricted release, considering both the activated and the contaminated materials arising in the dismantling of nuclear power plants.

6.7.2. Work Programme: See Ref. 3, Paragraph 6.7.2.

6.7.3. Progress and Results

The requirements on measuring instruments and methods are defined by the licensing authorities, case by case, on the basis of regulations. To define appropriate measurement strategies, the condition of the plant or parts thereof is to be taken into account. A good knowledge of the operation history is, therefore, necessary.

It would be desirable to norm decommissioning procedures and limits of radioactivity and to acknowledge certain measurement techniques and strategies.

6.7.3.1 Available experience

The most important radionuclides found in activated or contaminated materials were listed in order to determine the characteristics required for the counting devices. Some of these radionuclides are not directly detectable but occur in known ratios with others which are easier to measure. In some cases, radiochemical and spectrometric analyses in laboratory are needed.

The choice of the best counting device depends on

- its physical characteristics (weight, autonomy, easy handling, solidity);
- its electronical characteristics (energy threshold, background substraction, data processing);
- the cost of the measurement including staff and material costs.

Difficulties encountered by staff in charge of the measurements show the limits of existing devices and suggest new methods or apparatus. Table 22 shows an example of surface contamination measurement using a low-energy beta counter.

The experience gained during decommissioning of the nuclear plant of NS Otto Hahn has been drawn up. A direct measurement of activity was to be supplemented by an indirect measurement method. An indirect measurement strategy, based on statistical sampling, was proposed and accepted by the authorities. The quality of the method had to be proved by setting up a balance of initial state activity, activity transfer and final state activity. This involved the need to measure scrapping samples (initial state), acid solvent or sludge samples (activity transfer) and material samples (final state).

6.7.3.2. Existing hardware

A survey of instruments for measuring alpha-beta contamination, beta-gamma contamination and of contamination monitors available on the

French and German markets was made. For each of the about 20 instruments the most relevant information was tabled and fitness for on-site low-level measurements was assessed.

Table 22. Example of surface contamination measurement using a low-energy beta counter

Radionuclide	Emission	Efficiency (imp/s per Bq/cm^2)	Contamination level (Bq/cm^2)	Measurement (imp/s)
^{55}Fe	K	0	1 x 10	0
^{60}Co	β - γ	12	1 x 10^2	1.2 x 10^3
^{59}Ni	K	0	1 x 10^{-2}	0
^{63}Ni	β low E	0	2	0
^{90}Sr	β	18.6	5	9.3 x 10
^{90}Y	β	18.6	5	9.3 x 10
^{93}Zr	β low E	0	1 x 10^{-4}	0
^{94}Nb	β - γ	16.5	1 x 10^{-3}	1.6 x 10^{-2}
^{99}Tc	β	12	1 x 10^{-3}	1.2 x 10^{-2}
^{125}Sb	β	13	1 x 10	1.3 x 10^2
^{134}Cs	β - γ	14	4	5.6 x 10
^{135}Cs	β	7	3 x 10^{-5}	2.1 x 10^{-4}
^{137}Cs	β - γ	15	1 x 10	1.5 x 10^2
^{147}Pm	β	8.7	1 x 10	8.7 x 10
^{151}Sm	β low E	0	1 x 10^{-2}	0
^{241}Pu	β low E, γ	0	2.1	0
Total			1.53 x 10^2	1.8 x 10^3

7. PROJECT N° 7: INFLUENCE OF NUCLEAR POWER PLANT DESIGN FEATURES ON DECOMMISSIONING

The object of this project is to identify, develop and assess reasonable improvements in plant design with a view to facilitating decommissioning. Such improvements should be evaluated with due consideration to their effects on nuclear safety and radiation protection during the operational life of the plants.

The following six items had been identified as research subjects of interest for this project:

a) catalogue of design features facilitating decommissioning;
b) prolongation of the useful plant life;
c) reduction of the cobalt content of reactor materials;
d) concepts minimizing the activation of concrete;
e) protection of concrete against contamination;
f) improved documentation system.

7.1. Catalogue of Design Features Facilitating Decommissioning of AGRs

Contractor: National Nuclear Corporation Ltd, London, United Kingdom

Contract N°: DE-G-001-UK   Work Period: September 1981 - December 1983

7.1.1. Objective and Scope

In the process of designing the current Advanced Gas-cooled Reactors (AGRs) it has been recognized that if consideration is given to the decommissioning requirements during the design stage, it will be possible, without adversely affecting construction times and costs or operational aspects, produce a design which will minimize decommissioning time, minimize the production of radioactive waste, and minimize radiation dose to the decommissioning staff and members of the public.

To assist in the current design and future design work, a catalogue of design features facilitating decommissioning has been produced. The features are specific to the AGR but the principles could be applied to other designs.

Design considerations to facilitate decommissioning may sometimes conflict with conventional operational requirements and, therefore, some design provisions may not be accommodated in view of the many constraints involved in commercial design and manufacturing work.

7.1.2. Work Programme: See Ref. 2, Paragraph 7.1.2.

7.1.3. Progress and Results

The Heysham II/Torness AGRs, which are under construction, were chosen as a reference for this study. The indications are that the design features specifically installed to facilitate decommissioning and the optimized inherent design aspects will make the task of decommissioning this design of AGR viable. The preliminary estimates together with the anticipated manpower requirements (based on limited use of remotely operated equipment and robotics) give an operator dose for decommissioning of only a few hundred man-rems. The main design features of the Heysham II/Torness AGRs, which are utilized to facilitate decommissioning, are as follows.

The station layout is designed to facilitate radiation and contamination control. The main principles of this control are to minimize radiation dose by appropriate segregation and shielding of radiation areas and to minimize the spread of contamination by appropriate containment, zoning, ventilation and a system of changerooms and sub-changerooms. Decommissioning is therefore facilitated by both the layout (which can be utilized as a basis for a similar control system during decommissioning) and the segregation and limited spread of contamination produced during reactor operation. Also, the AGR has several other inherent features which assist in the minimizing of the amount of plant which becomes significantly contaminated during reactor operation, e.g., relatively low contamination of coolant circuit and a non-active turbine.

Shielding is provided around the reactor core inside the Pre-stressed Concrete Pressure Vessel (PCPV). These internal shields are primarily included to reduce the radiation levels inside the PCPV during reactor shutdown periods to allow personnel access into the reactor vessel via permanent man-access penetrations for inspection (as a back-up/supplement to the remotely operated in-service inspection equipment). However, this shielding has the dual advantage for decommissioning of reducing induced radiation levels outside this shielding to very low levels (mrem/h) during decommissioning (thus enabling extensive personnel access to be made for manual dismantling or setting up remotely operated dismantling equipment) and reducing the neutron-induced activation of the structures outside the shielding, which greatly reduces the radiological problems associated with dismantling and disposal of these structures. The activation of the internal shields themselves (based on the use of graphite blocks) does not present a major radiological problem for decommissioning. These blocks, which do not activate significantly, are simply keyed together and supplemented by steel inserts and plates, all of which can be removed and transported without difficulty.

There is tight control on the materials used inside the PCPV, with particular attention paid to the levels of impurities which could produce significant amounts of long-lived neutron-induced radioactivity. It is not feasible to reduce the levels of relevant impurities to insignificant values, since some trace elements are only present in very low concentrations and avoiding them would be excessively expensive if possible at all. As a result of a cost benefit analysis, the maximum levels of cobalt specified in all stainless and mild steels used inside the PCPV have been restricted to 1500 ppm and 300 ppm respectively. The average actual cobalt levels in steels used have been shown by sample analysis to be significantly lower than these. This material control, in a similar way to the provision of internal shields, facilitates decommissioning by both reducing radiation levels for access inside the PCPV and reducing the activation of the components themselves.

In high radiation areas, where direct personnel access may not be possible during reactor operation, consideration is given in the initial design to the provision of such features as will facilitate future operations (e.g., mounting features for remote handling equipment and boxed-up openings in the cast concrete structures filled with more easily removable material such as concrete blockwork). One specific example of this is the provision in the High Activity Debris Vault (HADV) of support beams for use during decommissioning in the removal of the high-activity waste. These support beams extend part way through the concrete shield wall of the vault, where they are accessible by removing concrete blockwork during decommissioning of the vault.

Where there are changerooms, access ways, etc., which are required for decommissioning only, floors and walls should be installed which permit easy modification, e.g. precast concrete slabs rather than 'in situ' cast structures. This has been applied adjacent to the HADV openings.

In all potential contamination zones the surfaces are designed to facilitate decontamination. Although the surface finishes (chlorinated rubber paint, epoxy paint etc.) will deteriorate during station life (and will need maintenance/replacement), they will have served their purpose in avoiding a significant build-up of contamination on surfaces and the basic surfaces will present a reasonable base for decontamination during the early years of decommissioning. It is not considered necessary to improve on standed surfaces specifically to aid decommissioning, since life-time performance is subject to many variables and equipment is already available for cleaning seriously contaminated surfaces (particularly when there are no onerous requirements on the standard of the post-decontamination surface finish during decommissioning).

Plant access routes are provided in the layout and construction of the station either as permanent operational routes or temporary construction routes. In most cases the decommissioning plan is essentially a reversal of the construction plan and use can be made of construction routes (sealed for reactor operation) for the removal of large mechanical items prior to the dismantling/breaking-up of the main building structure (which for contamination areas can serve as a useful containment for contamination control, although apart from the PCPV and HADV basic structural integrity is not relied upon for long-term decommissioning plants).

This work has been completed and the final report is being published (Ref. 17).

7.2. Design Features of Civil Works of Nuclear Installations Facilitating their Eventual Refurbishing, Renewal, Dismantling or Demolition

Contractor: Taylor Woodrow Construction Ltd, Southall, United Kingdom

Contract N°: DE-G-002-UK Work Period: January 1982 - January 1984

7.2.1. Objective and Scope

The primary aim of this research is to identify features which may be incorporated at the design stage of future gas-cooled nuclear power plants to facilitate their eventual decommissioning and dismantling. A supplementary aim is to extend this main study to incorporate light water nuclear power plants.

7.2.2. Work Programme: See Ref. 2, Paragraph 7.2.2.

7.2.3. Progress and Results

7.2.3.1. Advanced Gas-cooled Reactors

In the 1982 Report (Ref. 3, Paragraph 7.2.3.), it was reported that the literature review and study had been completed and that work was well advanced on the generation, development and assessment of ideas and features that would overcome the problems identified in the literature. This work has now been completed and the results and conclusions of it may be summarized as follows.

Planes of weakness: A substantial amount of analytical work has been carried out on this idea relating to the zone of activated concrete in a prestressed concrete reactor vessel (PCRV), and this has indicated its likely feasibility. Various methods for forming the planes have been considered and a number of possible options are believed to be possible. Figure 12(a) shows one possible construction arrangement.

Forceful break-up: It is believed that traditional use of explosives may offer an effective way of breaking up massive concrete structures (Paragraph 3.5.). It is also recognized, however, that chemical agents and mechanical devices may be useful when it is necessary to restrict the noise, vibration, dust and flying debris associated with explosives. One feature that would significantly aid the application of these three techniques is the provision of preformed holes in the active concrete. Whilst this is considered feasible, some measures may be necessary to maintain the shielding properties of the concrete and anchorage of the liner.

Selective use and location of materials: It is believed that reinforcement and tendons could be kept out of the activated zone, if desired. Whether or not this would be done in future designs is a matter to be considered at the design stage, taking into account possible effects on design, plant layout, construction and overall cost. The development and use of steels with low content of trace elements such as cobalt, silver and niobium would have known benefits in reducing the radiation problem (Ref. 18).

The provision of a more efficient and comprehensive shield, ideally to prevent neutron escape from the core, would again have obvious benefits and it is suggested that work be done by reactor engineers to assess whether such a shield is feasible. Improvement in the shielding property of concrete by increasing its hydrogen content or reduction in its activity by introducing boron are considered possible ways by which the volume of activated material may be reduced.

Removal of liner and penetrations: Discussions with liner design engineers have indicated that, without a change in fundamental concept, they see no future alternative to a mild steel liner firmly anchored to the concrete, as in present PCRV designs. Some simplification is believed possible to assist disconnection of penetrations from the liner, as shown in Figure 12(b), and review of certain technical and layout requirements could eliminate the need for heavy steel fabricated anchors to be found on liners of some existing PCRVs.

Some work has also been directed towards conceptual changes in liner design to make them more easily removable. The basic objective has been to make the liner immune from the effects of prestress application and thereby to reduce the degree of anchorage it needs. Possible ways in which this might be achieved are to have a free-standing liner or a grouted liner, as shown in Figure 12(c). It is believed that these might be feasible but further work would be necessary to investigate this. It is also felt that some development of liner insulation systems might lead to less intricate arrangements.

Removal of the standpipes has been considered as a separate exercise and work has been undertaken to produce an outline method for removal of them within an overall scheme for providing access to and removing the reactor internals.

7.2.3.2. Pressurized Water Reactors

Time has necessarily been spent gaining an understanding of the layout, design, construction and operation of the light-water reactor system, specifically the Standardized Nuclear Power Plant System (SNUPPS) design developed in the USA during the 1970s. This has indicated that, as far as civil engineering aspects are concerned, the only significantly activated region is an inner 1.0 m to 1.5 m thick zone of the primary shield wall that surrounds the reactor vessel. The decommissioning problems of this are similar to those of the activated zone of a PCRV and it therefore seems likely that similar solutions may be adopted.

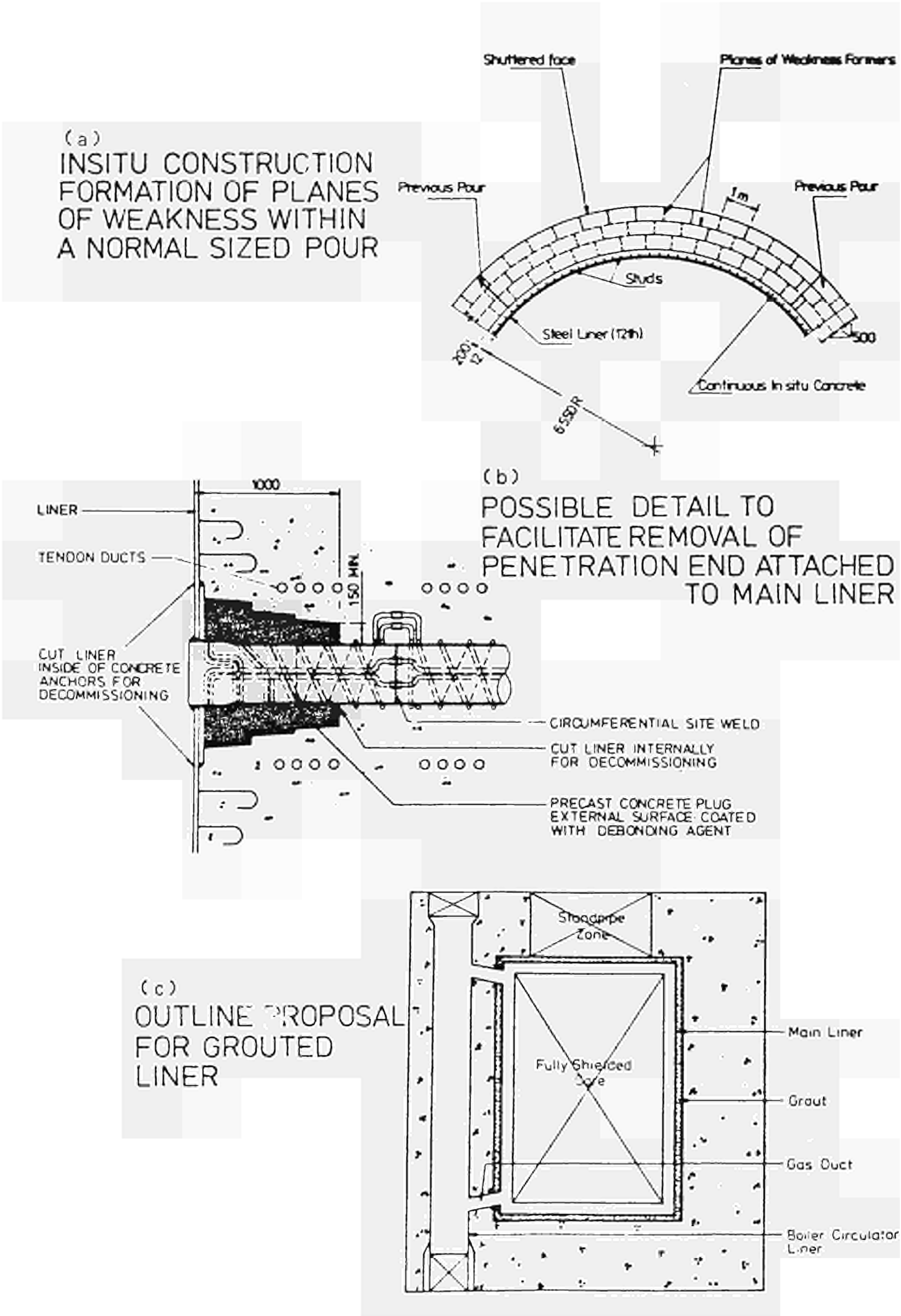

Figure 12. Features for facilitation of decommissioning AGRs

7.3. Erosion-Corrosion Testing of Cobalt-free Materials to Substitute Cobalt Alloys

Contractor: Kraftwerk Union AG, Erlangen, Germany

Contract N°: DE-G-003-D Work Period: January 1983 - October 1984

7.3.1. Objective and Scope

Cobalt is the main parent element for the formation of the strong gamma emitter cobalt-60. Material from cobalt-based alloys, used for instance in valve seatings, is removed by corrosion and abrasion and carried along with the primary coolant, then neutron activated in the reactor and, eventually, deposited on the inner surfaces of the coolant system. Consequently, the substitution of cobalt alloys by other materials would reduce the overall radiation level of primary systems of nuclear power plants to be decommissioned.

Previous research had shown that special alloys considered as candidates for substituting cobalt alloys have comparable tribological behaviour and corrosion resistance. It remained, however, to be verified, whether the erosion-corrosion resistance of these substitutes is adequate.

As the initial part of the research work, the resistance of seven cobalt-free alloys against erosion-corrosion was investigated (Ref. 3, Paragraph 7.3.3.). As a follow-up to this work, additional tests were initiated using water with suspended magnetite particles such as present in reactor coolant.

7.3.2. Work programme for the follow-up research

The additional tests will be carried out in the test device used for the first part of the research. In order to avoid leakage problems, which had appeared, the maximum flow velocity will be reduced from 65 m/s to 44 m/s. The water temperature (150°C) will not be changed.

The same materials will be tested. The water medium will be slightly changed according to the following specification: O_2: <0.01 ppm; B: 1000 ppm; Li: 1 ppm; Cl: 0.03-0.06 ppm; pH: 6.0.

Three different test conditions are planned to be used, i.e.:

- a test (600 h) with water containing Fe_3O_4 particles (the particle size and the concentration will be selected in pre-tests);
- if the weight loss of the specimens in the first test is too small, this test will be repeated with modified specimen supports producing turbulences in order to enhance the weight loss;

- a third test will be performed without Fe_3O_4 particles in order to show a possible influence of the slight change of the water specification.

7.3.3. Progress and Results

The fabrication of the specimens was started.

7.4. The Control of Cobalt and Niobium Content of Reactor-grade Steels

Contractor: United Kingdom Atomic Energy Authority, AERE Harwell, United Kingdom

Contract N°: DE-G-004-UK Work Period: July 1981 - May 1983

7.4.1. Objective and Scope

Cobalt is the principal parent element for the production of cobalt-60 in neutron irradiated steels. For several decades after reactor shutdown this will be the dominant source of gamma radiation. Niobium-94 is also an important source of gamma radiation which becomes dominant in the long term.

A study has been made of the possibility of reducing the cobalt and niobium content of reactor grade steels. The study included the following items:

- a review of the literature, supported by additional calculations in order to identify all the potential sources of long-lived gamma radiation in reactor steel structural components;
- sophisticated chemical analysis to establish typical levels of the most relevant elements in typical casts of steel;
- an evaluation of the possibility of reducing the levels of relevant elements by special steel-making treatments;
- an evaluation of the possibility of reducing the levels of relevant elements by using more suitable ores and other raw materials;
- an estimate of the likely additional steel-making costs.

7.4.2. Work Programme: See Ref. 2, Paragraph 7.4.2.

7.4.3. Progress and Results

The work was completed by April 1983, and the final report has been issued (Ref. 18).

7.4.4. Conclusions

A re-assessment of sources of long-lived gamma radiation in reactor grade steels indicates that besides cobalt, other potentially hazardous residual elements are niobium, silver and, of less importance, europium. Although the decay of ^{60}Co dominates the gamma activity for several decades after reactor shutdown, the longer-lived ^{94}Nb and ^{108m}Ag isotopes eventually become higher contributors to the radiation dose rate. Analyses of four heats of the type 304 stainless steel (reactor grade, with cobalt contents in the range 140-350 ppm) revealed niobium and silver contents of up to 85 ppm and 0.8 ppm respectively; the europium content of the steels was insignificant at $<4\times10^{-4}$ ppm. The main sources of cobalt, niobium and silver in steels are scrap materials and ferro-alloys used in steel-making. Since these elements are not easily oxidized, they are retained in the steel. However, the residual levels of these potential gamma radiation sources can be decreased by using higher purity raw materials in steel-making. The use of sponge iron and high purity nickel and chromium should enable residual levels of <50 ppm Co, about 5 ppm Nb and <0.05 ppm Ag to be achieved in type 304 steel.

7.5. Removable Coatings for the Protection of Concrete against Contamination

Contractor: Nucleco S.p.A., Milano, Italy

Contract No.: DE-G-005-I Work Period: October 1981 - December 1983

7.5.1. Objective and Scope

The quantity of radioactive waste arising from the dismantling of contaminated concrete structures of nuclear power plants could be greatly reduced, if the concrete would be efficiently protected by surface coatings against contamination during plant operation, and if the coatings could be removed before the dismantling of the structures.

The aims of this research are:

- to evaluate the feasibility and technical-economical data of protective coatings for concrete surfaces, which are easy to apply and to remove;
- to recommend coatings for concrete surfaces in new nuclear plants.

7.5.2. Work Programme: See Ref. 2, Paragraph 7.5.2.

7.5.3. Progress and Results

Six stripping paints (Table 23) were selected from the national market for investigation.

Table 23. Stripping paints tested

Paint	Manufacturer	Nature	Solvent
Pelableau 2000	Peinture Meyer Levallois (Paris)	Acrylic resins	Water
Duco 9.92	Maxmeyer Duco Fombio (Milano)	Vinyl copolymers + phthalic plasticizer	Organic
Pelante Blu G.265	Paramatti Vernici Torino	Plasticized vinyl	Organic
Magistripp	Magiplast Bologna	Vinyl copolymers + phthalic plasticizer (Flexol 810)	Organic
Enveloppe 47053	I.S.V.A. Torino	Vinyl copolymers + phthalic plasticizer (DBTF)	Organic
Pelabile Transparente 664.005	Colorificio Sammarinese S. Marino	Chlorinated rubber vinyl	Organic

These paints were tested for their conventional properties according to Unichim specifications for protective coatings in laboratories and nuclear plants, for water and chemical agents resistance, abrasion and impact.

Tests for stripping capacity and tensile strength were carried out on a 500 kg Amsler machine, equipped with a 100N resistor load cell, which allows to record load-time and load-elongation diagrams. The decontaminability of paints was measured according to the Unichim regulations at the ENEA Laboratories in Rome.

The protective power of the coating against contamination was checked by dipping painted concrete samples in radioactive solutions and by recording the surface activity before and after stripping the paint film.

The activity having permeated the coating was found to be very low (<1% of the deposited activity).

The results of the tests lead to the following conclusions:

- it is preferable to apply removable coatings on concrete surfaces already previously painted, for instance with a normal epoxy cycle (multi-layer technique), in order to get always good stripping conditions;
- the stripping paints should be applied in a thick layer, not less than 150-160 µm;
- the use of paints without inflammable solvent is recommended for safety reasons.

A technical evaluation for the possible utilization of the removable coatings in the Caorso Nuclear Power Station, has been carried out with limited results.

7.6. Characterization and Improvement of Coatings Protecting Concrete against Contamination

Contractor : Commissariat à l'Energie Atomique, CEN Saclay, France

Contract No. : DE-G-006-F   Work Period : January 1981 - December 1983

7.6.1. Objective and Scope

There exist various protective coatings for concrete walls and floors, which enable easy decontamiantion by washing or by removing the coating. Normally, these coatings do not assure efficient long-term protection against the penetration of contamination into the base material. An ideal coating would be :

- easy to apply and inexpensive;
- tight, adherent, resistant to chemical attack and to aging;
- easy to decontamination in normal operation;
- removable for dismantling of the structure.

Research of the best compromise between these qualities and development of a method for acceptance tests are the aims of this study.

7.6.2. Work Programme: See Ref. 2, Paragraph 7.6.2.

7.6.3. Progress and Results

A number of paint-coating systems have been definitely selected;

they are decontaminable, resistant to chemical corrosion, adherent to the concrete surface and also strippable. All are of the composite type, built up of the usual systems of decontaminable paint with one or two layers of strippable coating added. This type of system guarantees a high efficiency against contamination of the underlying concrete.

7.6.3.1. Products selected for the sandwich-type coatings

As most of the available strippable coatings are not suited for decontamination of fission products, excellent results have been obtained by enclosing a strippable layer between two decontaminable layers. The strippability is much better than if the strippable layer is applied directly on the concrete surface.

System HEMPEL: From the systems proposed by the company, a very homogeneous, totally water-based system has been preferred for its technical qualities, ease and safety of application, as well as for its relatively low cost. It is decontaminable for plutonium (86%) and fission products (93%) and consists of a layer of expoxypoliamid resins (HEMPADUR ENDUIT F 551 + WB F 522) decontaminable for fission products (89%), a layer of vulcanized, water-soluble and strippable rubber (B 205), which is not decontaminable, and a final layer of chlorated resin paint (HEMPACRYL TOP 5803) of very high ductility, decontaminable for fission products (95%).

System MEYER: This well-performing system, which is comparable with the HEMPEL system, but has the disadvantage of not being decontaminable for plutonium, consists of a strippable acrylic emulsion paint (IONIFIXE-AU), an extremely ductile strippable polyurethane layer (SOUPLISOX) and a base layer of decontaminable polyurethane paint (ISOX 1817); the whole system holds a decontaminability of 90% for fission products even after artificial ageing.

System GEHOLIT: This system with water as solvent is again a strippable paint (GEHOTEX AC) on the base of copolymers, as a layer between a decontaminable epoxypolyamid paint (GEHOPAN EX) and an upper layer of copolymer paint (GEHOTEX ACC). The decontaminability is excellent with 99%, decreasing to 92% after ageing.

System REVETANCH REVALPA: The system is decontaminable for fission products (95%), but not for plutonium. After a first layer (REVENDUIT) a polyurethane paint (ULTRAPRIM B) is applied; the following layer is a strippable, non-decontaminable acrylic paint (PELREV), covered by two layers of polyurethane elastomer paint (REVETANCHE), providing tightness and decontaminability. This system is very ductile and perfectly strippable even after ageing.

7.6.3.2. Strippable and decontaminable paint

Only the society GEHOLIT succeeded in producing a copolymer paint GEHOTEX MP which is decontaminable and strippable as upper layer. It is advantageously applied on a traditional coating. The paint is decontaminable to 94% and sufficiently removable after ageing.

7.6.3.3. Product trapping contamination

One product, a strippable acrylic emulsion paint (PELABLEAU 2000) from MEYER is an excellent trap for contamination. Such a product is useful to protect, temporarily, surfaces against contamination and for decontamination by stripping.

7.7. Evaluation of Design Features Facilitating the Decommissioning of PWRs

Contractor : NIRA and ENEA, Genoa, Italy

Contract N°: DE-G-007-I Work Period: April 1982 - October 1983

7.7.1. Objective and Scope

The aim of this study is to evaluate the design features that facilitate the decommissioning of Pressurized Water Reactor plants, in order to make it possible to take account of these features in the drafting of the project for the Italian reference nuclear power plant.

7.7.2. Work Programme : See Ref. 3, Paragraph 7.7.2.

7.7.3. Progress and Results

The method developed, which consists of a break-down of decommissioning operations into three logic schemes (operational graph, lay-out and safety graph) has been tested (Figure 13 for a flow-sheet) and applied to review the preliminary design of the fuel building of the reference power plant for the Italian nuclear plan. Graphs and dismantling sequences for each component are modified at each iteration of the method and this

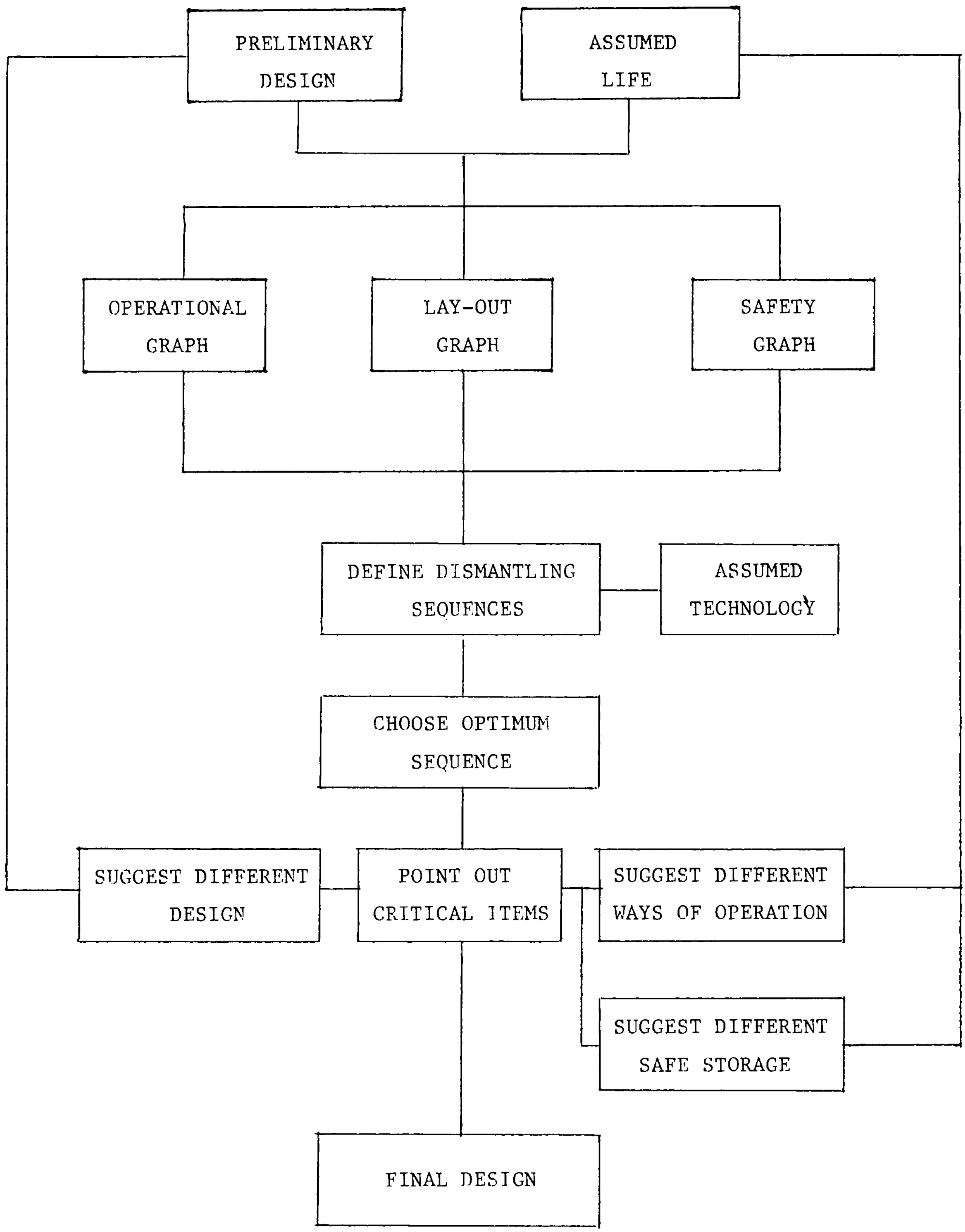

Figure 13. Flow-sheet for evaluation of plant design features

produces a great amount of data to be treated step by step. Analytically, a graph can be represented by a matrix and sequences can be treated in the matrix; this feature will allow an easy computerization of the method, which would be the next step in development.

The work, which covered the analytical development and an application, was completed in 1983 and the final report has been published (Ref. 19).

7.7.4. Conclusions

A method has been set up which can help to define the best decommissioning strategy and dismantling sequence for a nuclear power plant. This method has been developed in order to perform the design taking into account the problems that will arise during the decommissioning phase. Applying this method, starting from a preliminary design, engineers can detect the critical items of the decommissioning phase and introduce into the design the appropriate modifications.

The method, which is to be executed by computer, starts from the following elements :

- a preliminary design of the power station;
- the assumed life of the plant, to evaluate activation level of each sub-system;
- an assumption on the technology available at the time of the decommissioning of the power plant.

Analysing the system, one can derive three graphs (logic relations network), i.e.:

- operational graph, related to the systems to be maintained in operation during the phases of safe storage and dismantling;
- lay-out graph, related to the physical inter-connections between the various components and/or sub-systems, the building and the supporting structures;
- safety graph, related to the radiological risk linked to the activity of the various components and/or sub-systems.

The joint elaboration of these graphs allows to define several possible "dismantling trees" and to associate to them a "cost" defined as the sum of the economic cost and the radiological risk of exposure related to each operation. Then, the optimum sequence is evaluated and the critical items are pointed out.

The analysis of critical items can induce a feed-back into the design (modifications of the lay-out), into the way of operating the plant (modifications of the operating specifications) or the suggestion of appropriate strategies of safe storage.

7.8. Concepts Minimizing the Activation of the Biological Shield

Contractor: Electrowatt Engineering Services Ltd, Horsham, United Kingdom

Contract N°: DE-G-008-UK Work Period: April 1982 - March 1983

7.8.1. Objective and Scope

The aim of this research was to select and analyse realistic design concepts to minimize the activation of the biological shield in a nuclear power plant. The goals to be achieved by such concepts would be a reduction in the personnel exposure during dismantling operations and a reduction in the quantities of radioactive waste arisings.

7.8.2. Work Programme: See Ref. 3, Paragraph 7.8.2.

7.8.3. Progress and Results

The work was completed in March 1982 and the final report has been published (Ref. 20).

The study represents an initial investigation into concepts to minimize activation in the concrete of the biological shield by reducing the neutron flux. The approach was to insert a neutron-attenuating material between the reactor and the concrete biological shield. Two LWR designs were identified which have biological shields comprising of inner and outer walls. For these designs the concept involves using a suitable neutron-attenuating materials other than concrete for the construction of the inner wall.

Thirteen neutron-attenuation materials were identified as candidate materials for inner wall construction. The attenuation performance of each of these materials was assessed and compared. A reduced number of five materials, including the reference material concrete, were selected for activation analysis. Activation analysis was then carried out so that the induced activity of the candidate materials could be compared to that of concrete.

Results for both the BWR and PWR designs indicated that all four substitute materials compare favourably with concrete and that the R/X material 'Ricorad' has by far the lowest induced radioactivity. Many of the undesirable reactions in the substitute materials are greatly reduced by the presence of boron-10 with its high thermal neutron capture cross-section. However, it is likely that the activity levels calculated for the substitute materials are non-conservative due to lack of detailed information on impurities. A thorough chemical analysis of samples is required before a more rigorous assessment of the materials can be made.

7.8.4. Conclusions

The investigations have looked at the nuclear aspects of concepts minimizing the activation of the biological shield of LWRs but have not examined other aspects, such as the engineering problems, which have to be considered to establish the overall feasibility of the concept. However, from the information available on the shielding materials and the analysis carried out, it can be concluded that significant reductions in neutron flux activation levels can be achieved in both the inner and outer shield walls and that the entire concrete outer shield wall could be classified as non-radioactive at the start of the decommissioning phase.

These initial investigations could now be followed by an investigation of the engineering problems both for construction and dismantling and of the likely costs to assess the economic viability of the concept. The acceptability of the concept should also be addressed since a change in design practice will involve extensive justification to establish the safety case with appropriate regulatory bodies.

7.9. Biological Shield Design with Dose-reducing Effect in Decommissioning

Contractor: Dyckerhoff & Widmann, München, Germany

Contract N°: DE-G-009-D Work Period: July 1982 - December 1983

7.9.1. Objective and Scope

The objective of this study is to identify and evaluate alternatives in biological shield design that pose less radiological problems in decommissioning than existing designs. The possibilities of achieving this aim are in selection of the materials (trace elements in steel and concrete) and a design facilitating the dismantling operations without endangering structural integrity. A large Pressurized Water Reactor will be taken as a reference for this study.

7.9.2. <u>Work Programme</u>: See Ref. 3, Paragraph 7.9.2.

7.9.3. <u>Progress and Results</u>

The neutron-flux calculations were completed. The following materials were considered for the free-standing biological shield part (55 cm thick), to calculate fluxes for the whole biological and support shield area:

- 55 cm normal concrete;
- 10 cm B_4C and 45 cm normal concrete;
- 45 cm normal concrete and 10 cm B_4C at the outer diameter;
- 45 cm normal concrete and 10 cm Fe;
- 45 cm normal concrete and 10 cm ceramics;
- 55 cm heavy concrete (hematite).

The comparison of obtained values shows that hydrogen in concrete and boron in boron-carbide essentially influence the flux distribution.

The activation was calculated, by means of the ORIGEN 2 computer programme, for full-power operating times from 1 to 30 years and decay times from 1 to 100 years.

The introduction of a B_4C layer between the reactor core and the concrete leads to a lower thermal neutron flux and, consequently, to a much lower activation of the concrete (Figure 14). The activation of the B_4C layer is also lower than the activation of a corresponding concrete annulus.

In order to reduce the induced activity, the constituents of the concrete could be selected accordingly; to avoid a large contribution from activated steel, the reinforcement bars would have to be arranged at the outer diameter, which leads to the need for pre-stressing of bars or a liner.

The arisings of activated material were calculated for the different designs assuming limits for unrestricted release of 37, 3.7 and 0.37 Bq/g. The exposure of workers (mrem/h) during dismantling operations and the heat generation (W/g) also were computed for the different versions.

In addition to the choice of materials, alternative designs facilitating the dismantling of the biological shield were considered. A design assembled from pre-fabricated concrete blocks and enforced by pre-stress-

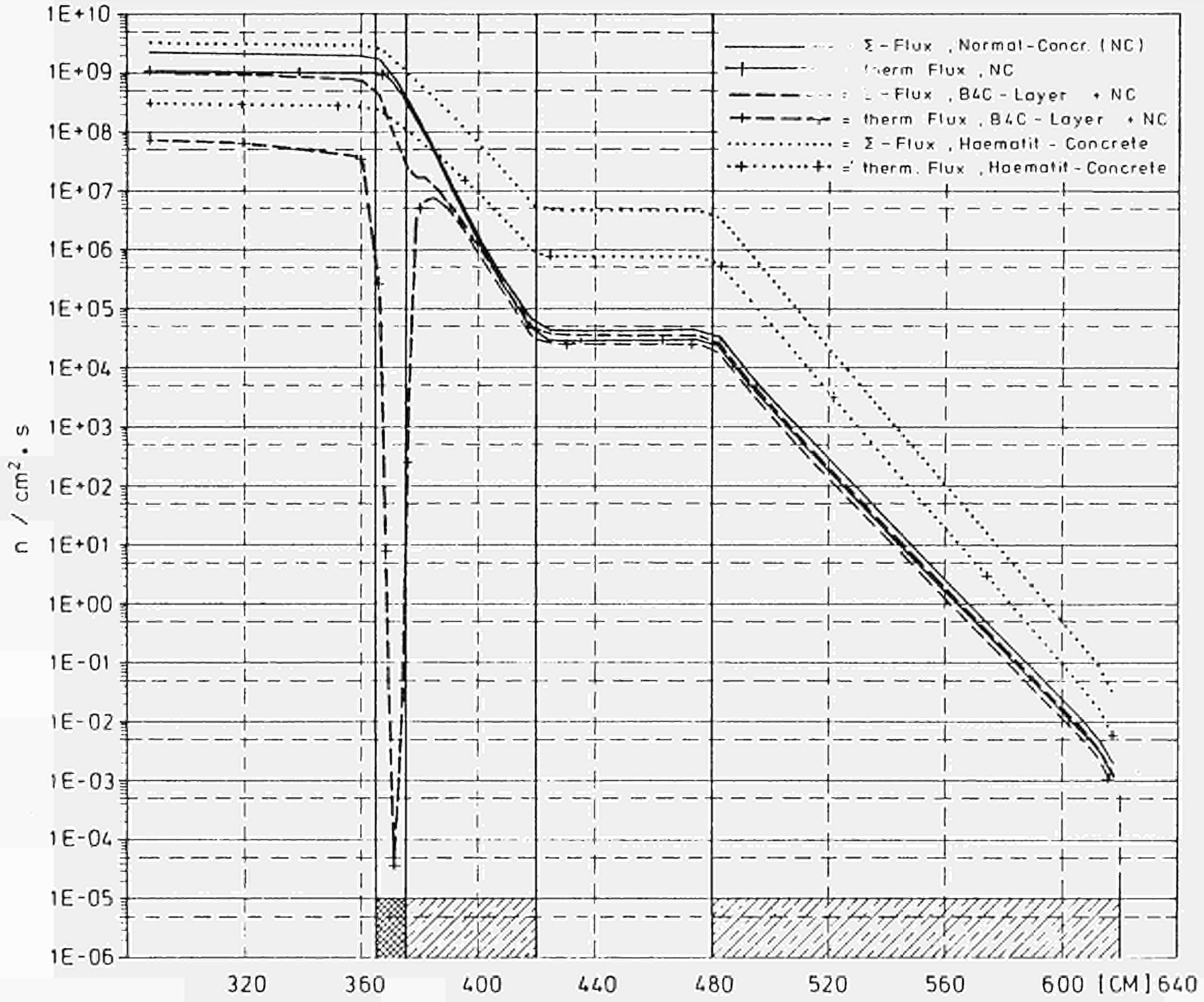

Figure 14. Neutron fluxes in the biological shield of a PWR at core mid plane

ing at the outer diameter, has been worked out in more detail. To fabricate the concrete elements, a patented process involving 3 mm steel plates for the obtainment of contact surfaces of the required finish during the pouring of the concrete, which are removed after hardening, is proposed.

7.10. Documentation System for Decommissioning of Nuclear Power Plants

Contractor: Technische Hochschule Darmstadt, Germany

Contract N°: DE-G-010-D   Work Period: September 1982 - March 1984

7.10.1. Objective and Scope

A good documentation of the real state of a nuclear power plant is essential for the decommissioning of the plant, especially if the dismantling is delayed to take advantage of the radioactivity decay.

The aim of this research is to develop a specific documentation language and to specify software and hardware components that are capable of working autonomously together with a microfilm storage and retrieval system.

7.10.2. Work Programme: See Ref. 3, Paragraph 7.10.2.

7.10.3. Progress and Results

The following results were achieved:

- A further definition with full details and modification of the documentation language due to new requirements concerning:
  - volume and distinct marking of the vocabulary;
  - enhancement of the record description of the documentation unit;
  - extensions to the application of software;
  - storage of the original paper records.
- Specification of the software system, especially :
  - specification of utility programs appropriate for the required data base management processes of the system;
  - indication of suitable and available DBMS-software;
  - description of enhancements of application software.
- Specification of the hardware system, especially :
  - requirements of the hardware and operating system's software which is able to support the application programs;
  - requirements of the peripheral storage and back-up system;

. description of the peripheral devices for the user interface.

- Proposals for a suitable and powerful microfilm system with some hints on state-of-the-art analog storage and retrieval systems.
- First proposals for the organization of the documentation system's operation.

For nuclear power plants built between 1960 and 1980, the whole stock of data which is essential for the process of decommissioning comprises 300,000 to 400,000 sheets of paper including about 30,000 to 50,000 engineering drawings. Nuclear power plants which are running since 1980 or are near to completion, will produce a total of 2.5 to 3 million document pages including 150,000 to 200,000 engineering drawings.

Discussions with licensees of nuclear power plants being decommissioned lead to a documentation system which has been decomposed into a model of layers separating infological from technological layers (Figure 15).

The documentation language has been extended, as compared to the existing system INFODOK, and a first proposal for a ground-level thesaurus with the following categories for documentation and retrieval is presented:

M01: (SUB-) SEGMENTS OF DATA STOCK
M02: TYPE OF DOCUMENT
M03: COOPERATING ORGANIZATIONS
M04: CONSTRUCTION SYSTEM
M05: REACTOR SYSTEM
M06: ENGINE / MACHINERY / POWER SYSTEM
M07: PUBLIC UTILITIES AND DECOMMISSIONING SYSTEM
M08: CONTROL SYSTEM
M09: FUNCTION AND PROCESS
M10: INVENTORY OF NUC ACTIVATION / LEVEL OF DECONTAMINATION / ACTIVITY
M11: RELATION OF DOCUMENT'S CONTENTS TO RUNNING PHASE OF THE NPP
M12: ACTIVITIES
M13: HARD-/SOFTWARE SYSTEM OF THE DOCUMENTATION SYSTEM

The description of the data record of a documentation unit has been adapted to the requirements and the number of data fields was incremented from 26 to 36. Two video display forms are proposed to support the documentation process in practical application.

Requirements to the software system are functionally defined within a model of layers (Figure 16) and the interfaces between the system's main

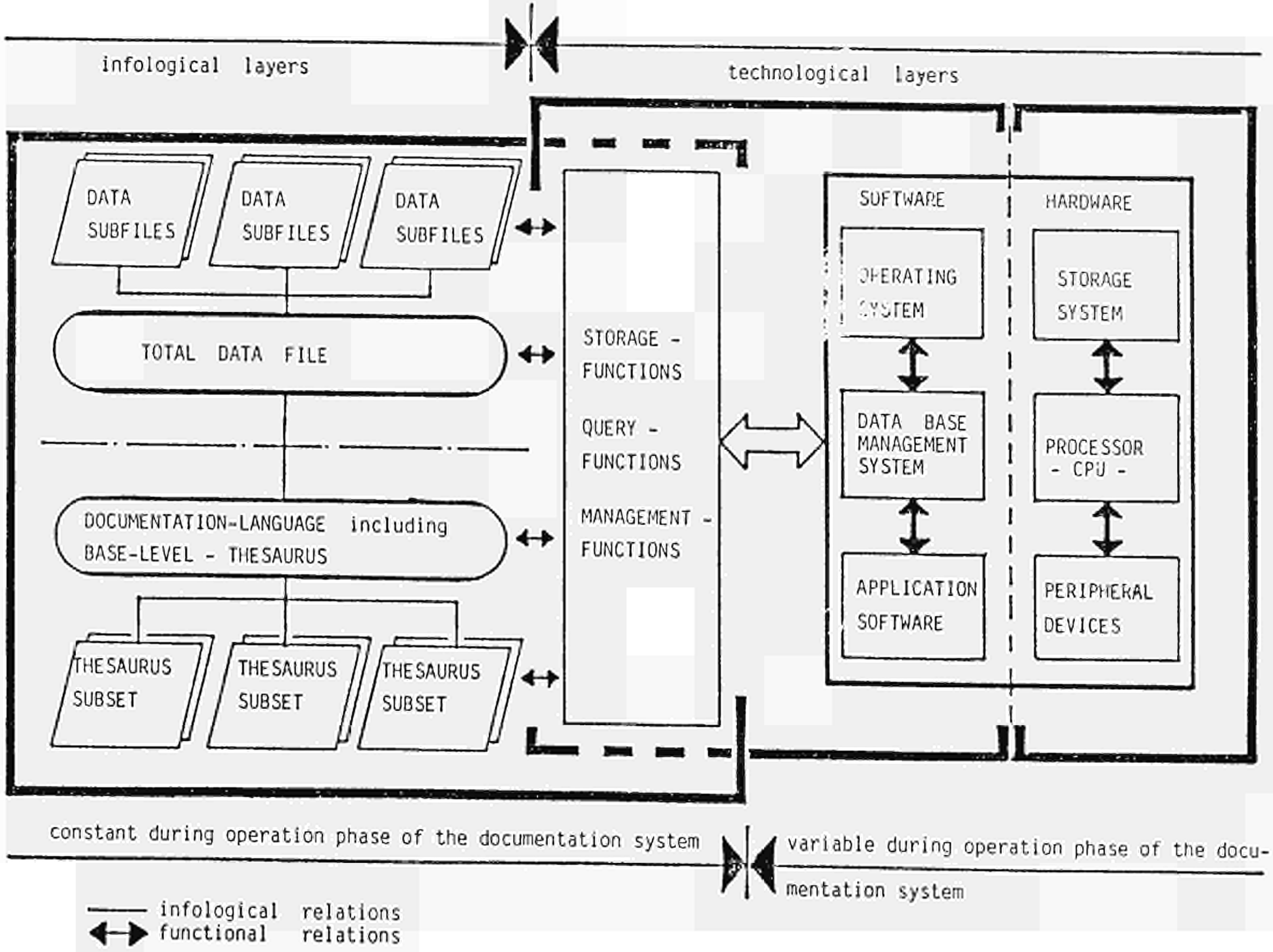

Figure 15. Structure of layers of the documentation system

components the application programmes of the system INFODOK, the data base management system, and the operating system as well as the interfaces between the units of function in the interior of a layer are described. Requirements to the hardware system are functionally and quantitatively evaluated.

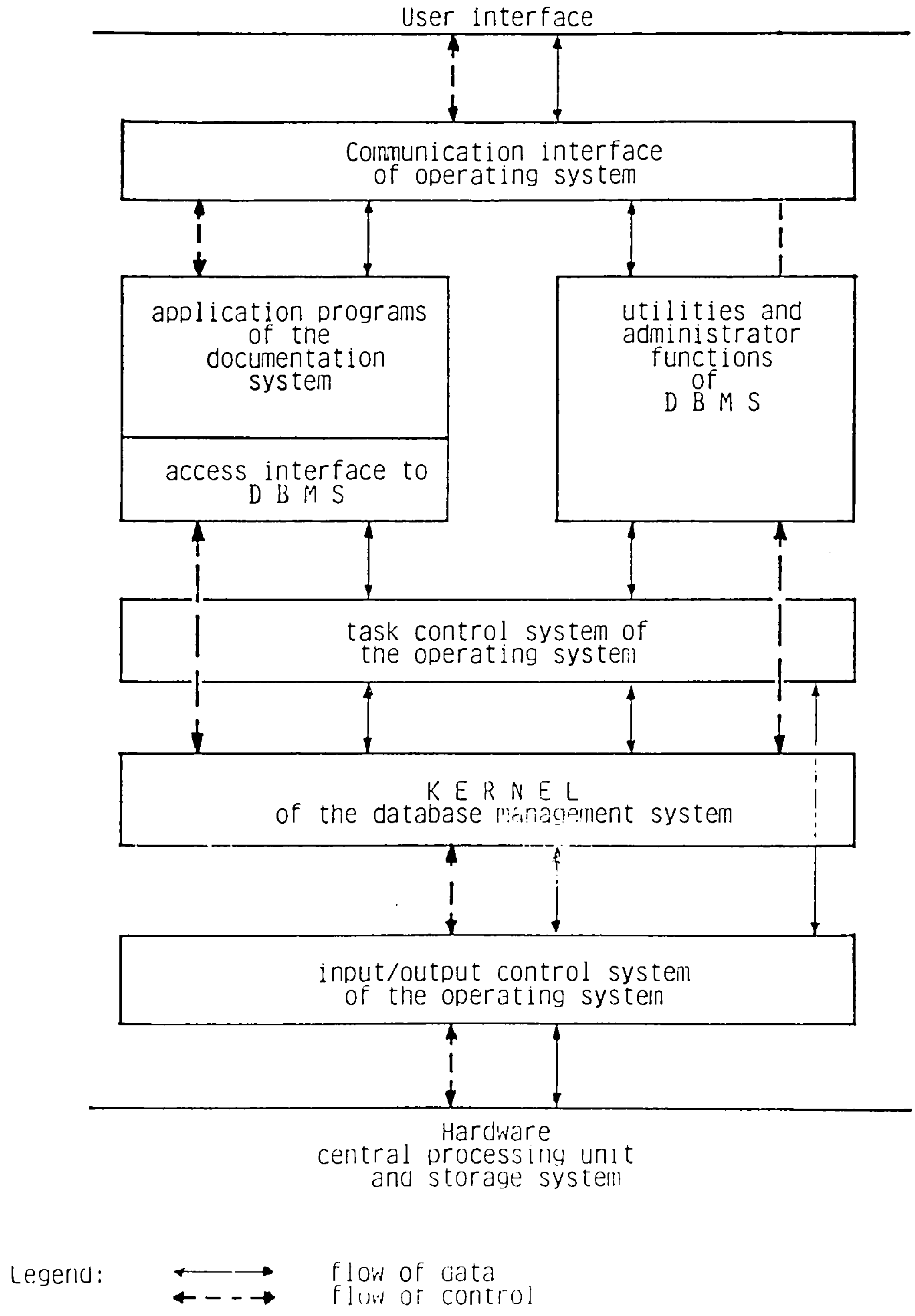

Figure 16. Structure of layers of the software system

The requirements to the application programmes are enhanced by specific sub-functions with respect to the following operations:

- to separate significant sections of the stock of data for each individual running phase of a nuclear power plant (specification, design, construction and running process) from the entire stock of data. This process will automatically be supported by standardized queries;
- to derive the vocabulary and ground-level thesaurus of documentation languages specific to each running phase for the different sections of the stock of data.

An automatic microfilm storage and retrieval system integrated into the whole system and coupled via an interface to the computer is specified. To organize the computer aided documentation process, a basic sequence of operations is proposed (Figure 17).

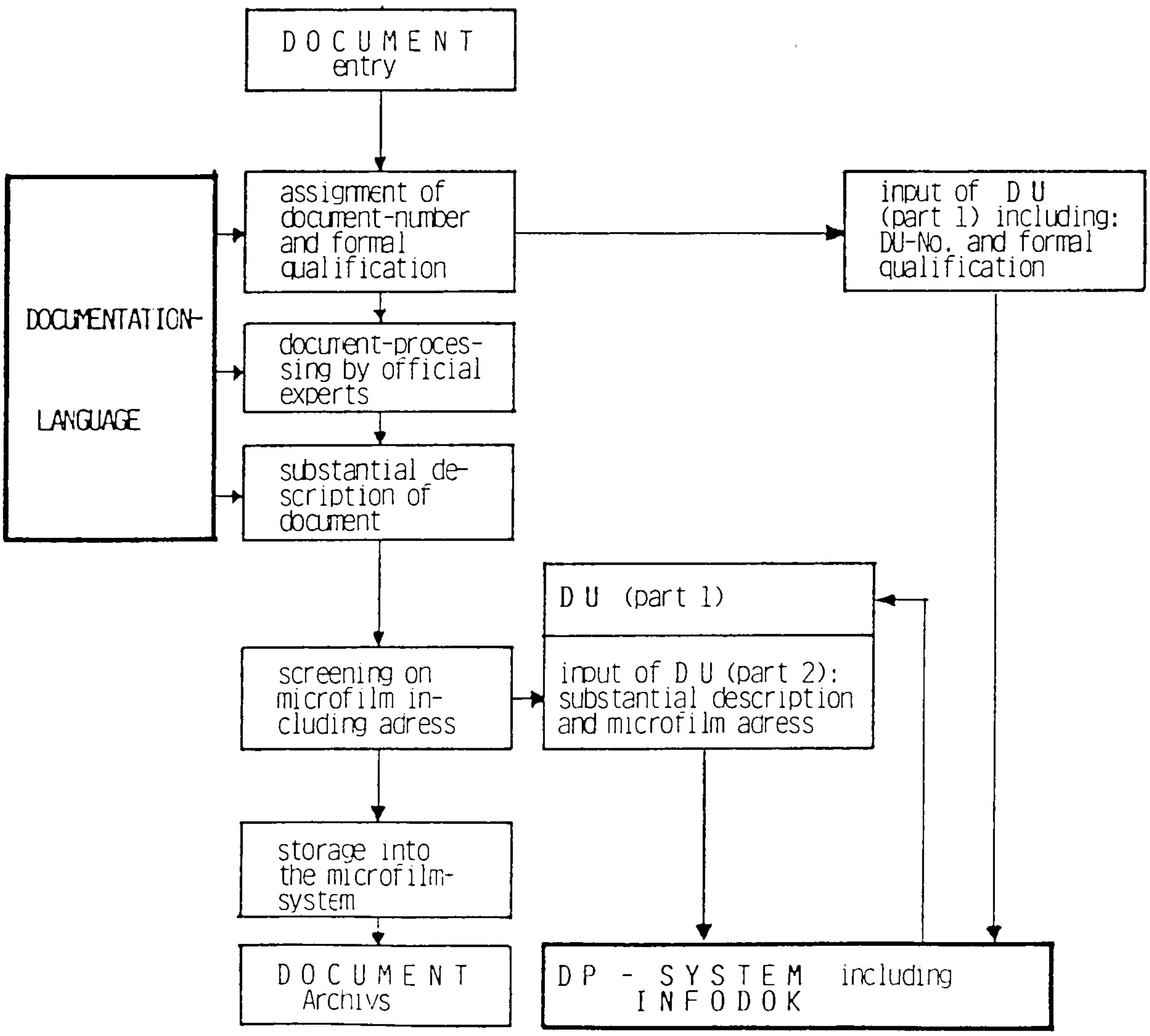

Figure 17. Basic procedure of the documentation process

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The present report describes the further progress of research and contains a large amount of results. For a majority of the 51 research contracts composing the 1979-83 programme, work was completed by the end of 1983; the conclusions drawn from this work are in this report.

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