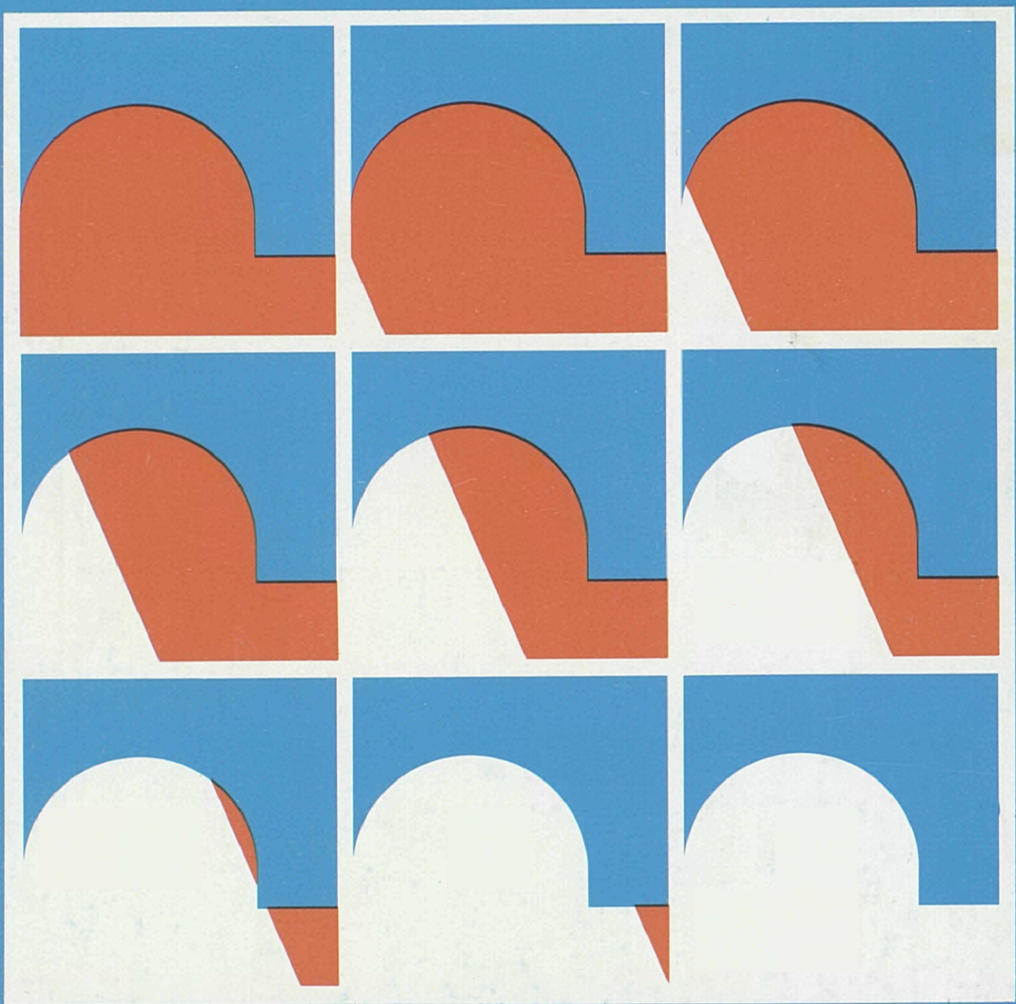
Commission of the European Communities

nuclear science and technology

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

Third annual progress report (year 1982)



Report

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Directorate-General
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FOREWORD

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

The Council of the European Communities has adopted the programme in March 1979 (Ref. 3), considering:

> "Certain parts of 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 seeks 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 is 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 planned for this five-year programme amounts to 4.7 million ECU.

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

The 1980 and 1981 Reports described the work programmes of most research contracts and intial results of the research relating to Projects N° 1 to N° 5.

The present report describes the further progress and results of research. Since 1982 has been a very active year of work under the programme, this report contains a large amount of results. Besides, the work programmes of some additional contracts awarded through 1982, most of them relating to Projects N° 6 and N° 7, are given. The Commission staff in charge of the programme during 1982 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

** Part-time

CONTENTS

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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

1.1.3.1. Visits to and inspections of Magnox nuclear power stations

Visual inspections of two nuclear power stations were made, both the stations were to be used for sampling and testing of building materials.

The purpose of the visits was to make a preliminary assessment of the station buildings' condition at Oldbury power station on the one hand, and to make a detailed inspection of the reactor building at Trawsfynydd power station, with the specialist testing contractor, in order to identify specific locations for sampling and testing.

1.1.3.2. Analysis of surveys on the condition of power station buildings

The study and analysis of existing surveys made independently of this contract was completed. These covered six coal-fired stations and four nuclear stations with an age range of 15-28 years.

The conclusions were that two materials had, almost universally, caused problems due to degradation. These were bituminous roofing felt and steel wall cladding, whether galvanized or with other coatings. While the effective life of felt was expected to be limited, the early corrosion of such cladding sheets was not anticipated.

In general, other materials and systems reported on in the surveys were giving good service apart from some problems which were due to lack of adequate maintenance, or design shortcomings, rather than premature material failure.

1.1.3.3. Sampling and testing of materials from nuclear power stations

Following the inspection of Trawsfynydd nuclear power station reactor buildings, detailed proposals for the locations and types of samples and the nature of in-situ non-destructive tests were made. The sampling and in-situ testing is to take place in January 1983.

Arrangements are in hand for a visit to Oldbury nuclear power station by the specialist testing contractor to select locations for sampling and testing at that station.

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 - December 1983

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. Inventory of plant components

This part of the study was completed in 1982 by a catalogue of primary circuit components in the reference PWR station. This catalogue lists, for each component, the type of information which will be useful during decommissioning at Stages 1 and 2, such as whether the equipment is to be kept in service or in working order, materials, constituents, type of radioactivity, weights, volumes, etc.

1.2.3.2. Aging of plants

This work was completed by a study of the effects of aging on the strength of concrete, in particular the conditions in which fissures are formed and can be examined and repaired. Fissuration is the form of deterioration that is most often met and whose consequences may jeopardize the integrity of structures. A procedure has also been proposed for examining containments after final shutdown.

1.2.3.3. Functions which are to be kept operational after final shutdown

A study was performed of the functions of equipment or systems that:

- contribute to the confinement of radioactive substances;
- play a part in maintaining the barriers against radiation;
- contribute to technical functions.

As regards the primary circuit, aims of this study were, amongst others:

- to identify areas prone to corrosion;
- to ascertain the consequences of operations to decontaminate or flush the circuits;
- to determine whether or not lagging needs to be left in position;
- to assess whether or not it is advantageous to remove perishable components such as seals and pump packing.

As far as the essential facilities remaining in service are concerned, the study also made it possible:

- to draw up instructions for supervision, maintenance and operation;
- to assess the costs of maintaining plant in working order, purchasing spare parts and replacing equipment.

These various lines of enquiry were pursued for Stages 1 and 2 of decommissioning.

1.2.3.4. Residual radioactivity

Analysis of the results of recent experimental work permitted to throw fresh light on the contamination and, in particular, the dose rate due to caesium isotopes and the effect of each nuclide on the dose rate due to corrosion products. Graphs plotted on the basis of this work show the evolution of the dose rates over a period of hundred years after final shutdown.

1.2.3.5. Relationships between postponement times and their consequences

An initial approach to studying these relationships was made on the basis of the comprehensive results obtained for the facilities remaining in service. The results of this initial research are presented by curves of cost versus postponement time; they are very tentative and concern only the facilities remaining in service.

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

2.1.3.1. Tests on inactive specimens

Further removal tests were carried out on inactive coated and uncoated specimen slabs, using the flat-section jet torch (width: 100 mm). The concrete used had the same density (2.39 g/cm^3) as in the previous tests, but was finer grained. By modification of the remotely-controlled feeding device, the velocity of the torch could be increased to 1 m/min.

The depths of removal and the dust concentration in the exhaust air of the working cabin were measured. The results (Table 1) show that it is possible to reach depths of removal which are sufficient for the decontamination of concrete surfaces. During repeated flame-scarfing treatments, decreased dust concentrations were measured. In comparison with the previous tests, the dust concentration during burning off was considerably higher, probably due to the finer-grained concrete. For mechanical cleaning of the burnt-off surfaces, the dust concentration in the cabin exhaust air could be reduced by about a factor of ten by installation of a local suction removal.

Table 1. Dust development and removal depth during flame-scarfing

Test No.	Concrete finish	Surface treatment	Number of treatment	Dust concentration (mg/m^3)	Removal depth (mm)	Feed rate (m/min)
1	bare	flame-scarfing	1. removal	140	1.5-4	1.0
2	bare	brushing, grinding	-	15.6	-	-
3	bare	flame-scarfing	2. removal	88.8	4.5-7	1.0
4	bare	brushing, grinding	-	16.1	-	-
5	bare	flame-scarfing	1. removal	110.0	2-4	0.5
6	bare	flame-scarfing	2. removal	96.4	3-5	0.5
7	coated	flame-scarfing	1. removal	70.0	2-5	0.5
8	coated	flame-scarfing	2. removal	41.6	4-9	0.5
9	coated	flame-scarfing	1. removal	92.8	2-5	0.5

2.1.3.2. Tests on contaminated concrete surfaces

Flame-scarfing tests on contaminated surfaces were carried out in the controlled area of the shut-down nuclear power plant Gundremmingen

KRB-A. Suitable areas for these tests were selected in the store for low-level radioactive solid operating wastes, considering their history and the expected level of contamination. The areas were coated with PVC varnish and the underlying concrete had finish quality.

In order to determine the contamination having penetrated, boring samples were taken from the surface and examined with a gamma spectrometer. The surface contamination was determined by means of a contamination monitor.

The tests were carried out manually, using a flat-section torch (width: 250 mm) at a feed rate of about 1 m/min. During the tests, a large part of the inner building surface of the solid waste store was decontaminated. With the help of a mobile aerosol collector, air samples were taken during jet burning in order to determine the aerosol activity. Moreover, the air in the room was analysed for CO and HCl.

The following results were obtained:

- initial contamination: 6×10^{-10} - 1.5×10^{-8} Ci/cm^2;
- depth of penetration of contamination: 4 - 5 mm;
- contamination composition: ^{137}Cs (90%) and ^{60}Co;
- removal capacity: 1 - 1.5 mm for each treatment (after four treatments, the decontamination was achieved);
- activity in the room atmosphere: 6×10^{-11} Ci/m^3.

The dust-loaded filter was examined using a gamma spectrometer. Only traces of ^{137}Cs and ^{60}Co were found in the filter. The bulk of the activity was concentrated in the slag, which was removed after every jet burning, using a floor grinding machine with local suction removal.

Results of the air analyses are given in Table 2. They show that for coatings containing PVC, mechanical removal of the coating is necessary before the jet burning, in order to avoid the release of toxic gases into the atmosphere.

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 - June 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

Table 2. HCl and CO contents (mg/m³) in the room atmosphere during jet burning

	Coated surface	Surface with coating rests	Bare surface	MAK*
HCl	56 - 91	2.8 - 5.6	-	7
CO	77 - 110	55	-	33

*MAK = Maximale Arbeitsplatzkonzentration (maximum allowable work place concentration) according to "Technische Regeln für gefährliche Arbeitsstoffe TRgA 900"

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

A new type of cavitation element, based on a circular geometry, will be designed and tested. The water will be supplied along the central axis, perpendicularly to the target surface and then be diverted to flow radially outward, producing an annular zone of erosion.

2.2.3. Progress and Results

A new circular cavitating nozzle has been designed by optimization of the geometric characteristics of the nozzle used during the previous contract (Figure 1).

A first series of tests was performed with various specimen materials, exposure times and downstream pressures (Table 3). Efficiency, defined as the ratio of mass-loss versus energy consumed, was found to be 16 times higher than for the cavitation element with one-directional flow.

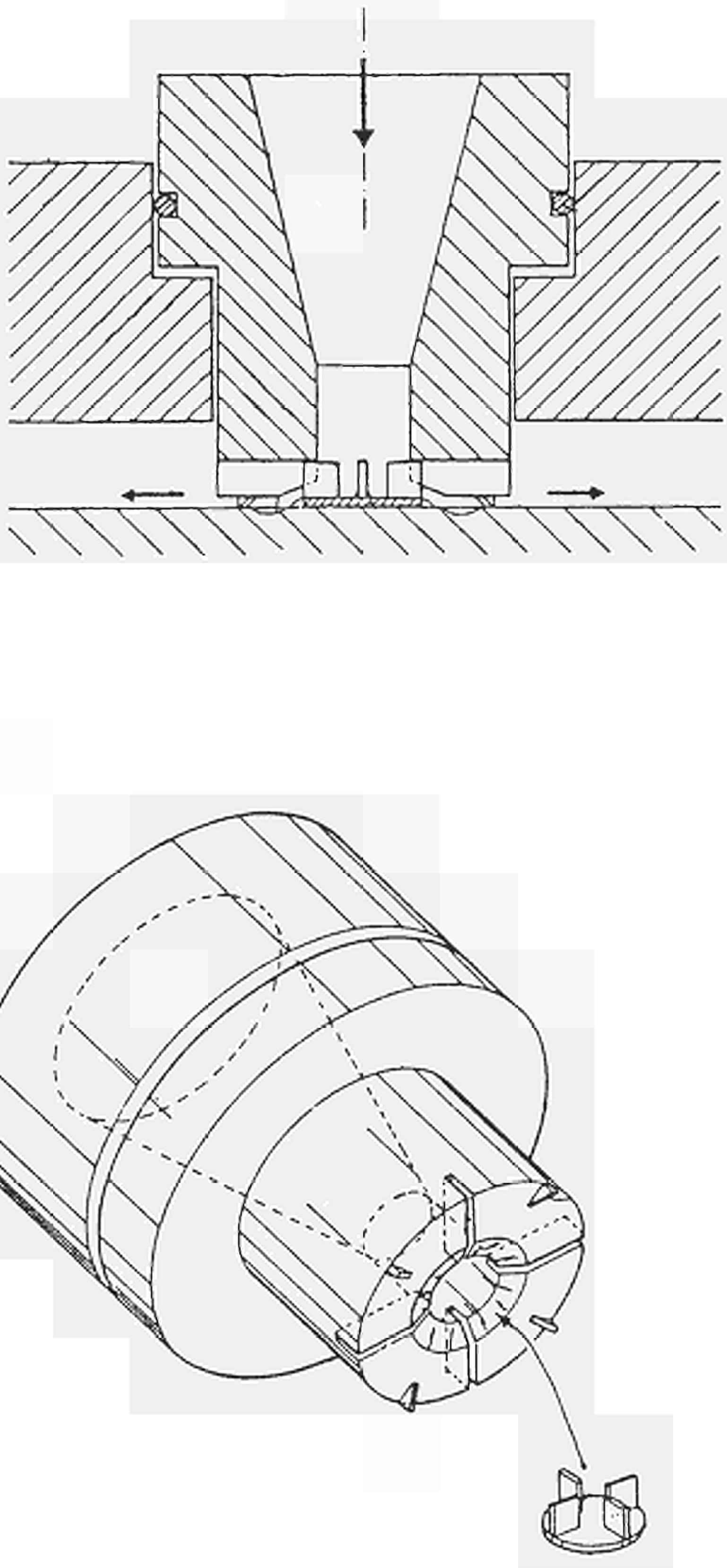

Figure 1. Circular cavitating nozzle

Table 3. Erosion tests using the circular cavitating nozzle

Test No.	Specimen material	Downstream pressure (bar)	Exposure time (min)	Volume loss (mm^3)
1	Aluminium (99% pure)	14	60	220
2	Aluminium	14	0.5	4
3	Stainless steel	14	600	0.35
4	Stainless steel	14	60	<0.1
5	Aluminium	14	3	7
6	Plexiglass	14	60	24
7	Duraluminium (4% Cu)	14	10	0.24
8	Aluminium	25	10	13
9	Aluminium	14	10	47
10	Aluminium	5	10	12
11	Aluminium	10	10	45
12	Glycérophtalic paint	14	3	
13	Mild steel (A 37-1)	14	600	0.30

The brass of the device was not eroded after twenty hours of service, which produced strong erosion on hard metals like stainless steel.

Another cavitation element, using a vortex flow, is now being designed and will be tested during the next months.

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

2.3.3.1. Characteristics of contamination layers

Photographic and raster electron microscope examinations of the inner faces of untreated steam and condensate pipe samples were performed. On the steam pipe sample, intercrystalline corrosion attack with an intensity varying over the pipe circumference was found. On the condensate pipe sample, intercrystalline corrosion was less and uniform.

Measured characteristics of the oxide layers on pipe samples from the primary water, steam and condensate ducts are summarized in Table 4.

Table 4. Characteristics of oxide layers

Characteristic	Unit	Primary water pipe	Steam pipe	Condensate pipe
Oxide quantity	g/m^2	45.8	7.96	10.4
Fe_2O_3 content	wt%	76.4	76.6	74.6
Cr_2O_3 content	wt%	15.7	16.2	18.4
NiO content	wt%	7.4	6.5	6.7
CoO content	wt%	0.5	0.7	0.3
^{60}Co activity	Bq/g	9.78×10^8	3×10^6	1.8×10^7

Activity depth profiles have been determined on the pipe samples from the primary water and the condensate ducts. After removal of the oxide layer, base material was electrolytically removed in layers of 1 µm thickness. Subsequently, the activity and metal contents were determined in the electrolytic solutions.

Figure 2 shows the ^{60}Co activity profile in the primary water pipe samples. After removal of 26 µm, the ^{60}Co content equals that of the outer pipe wall. A quite different profile was found in the condensate pipe samples, where a ^{60}Co content equal to that of the outer pipe wall is only reached after removal of about 80 µm and 90 µm, respectively (Figure 3). This is all the more astonishing as the corrosion attack and the activity deposit were clearly lower than on the primary water pipe samples. These results seem to indicate that the presence of ^{60}Co in the deeper layers is not due to penetration of ^{60}Co but to activation, for instance by neutron-emitting fission products.

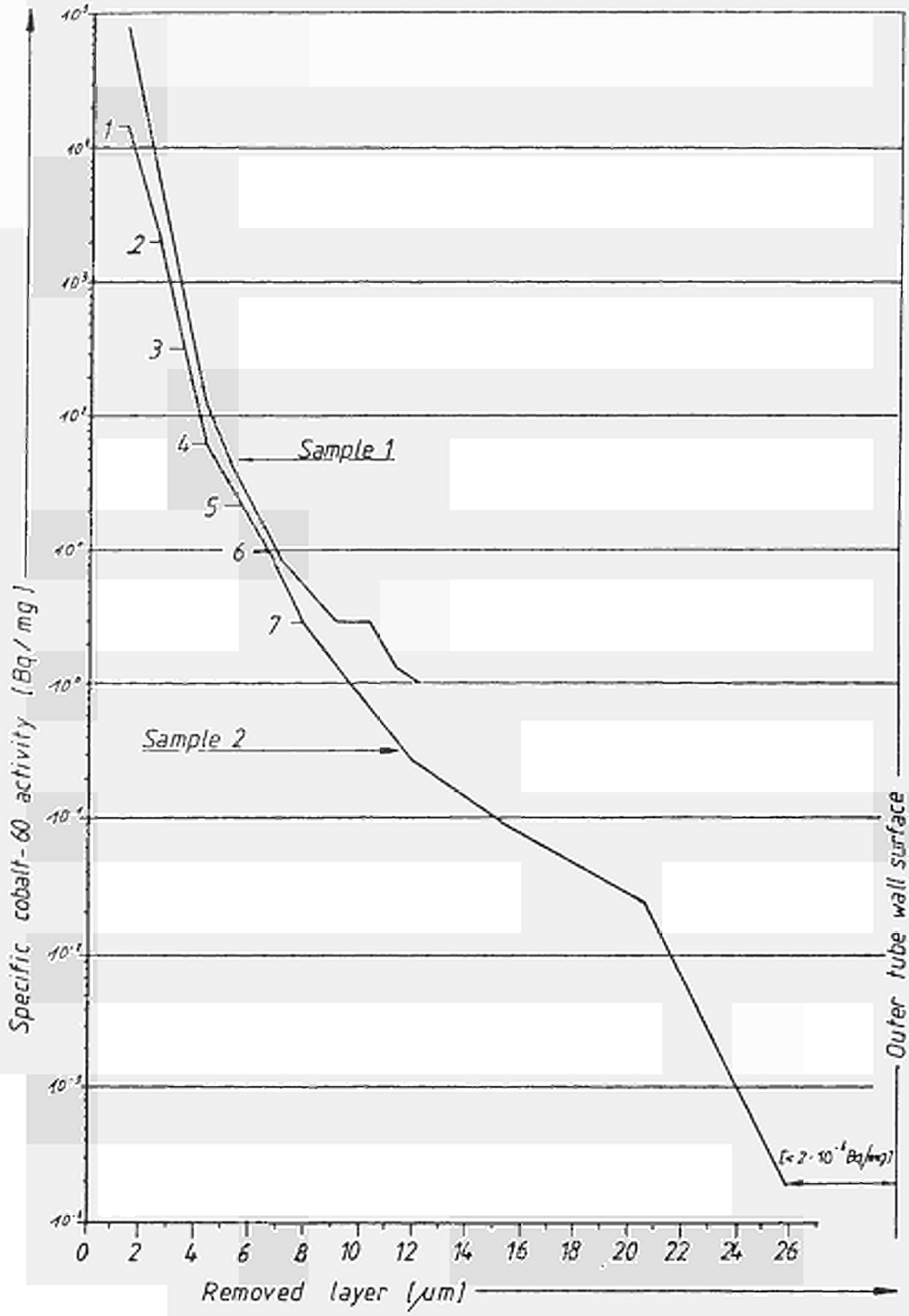

Figure 2. ^{60}Co activity profile in primary water pipe samples

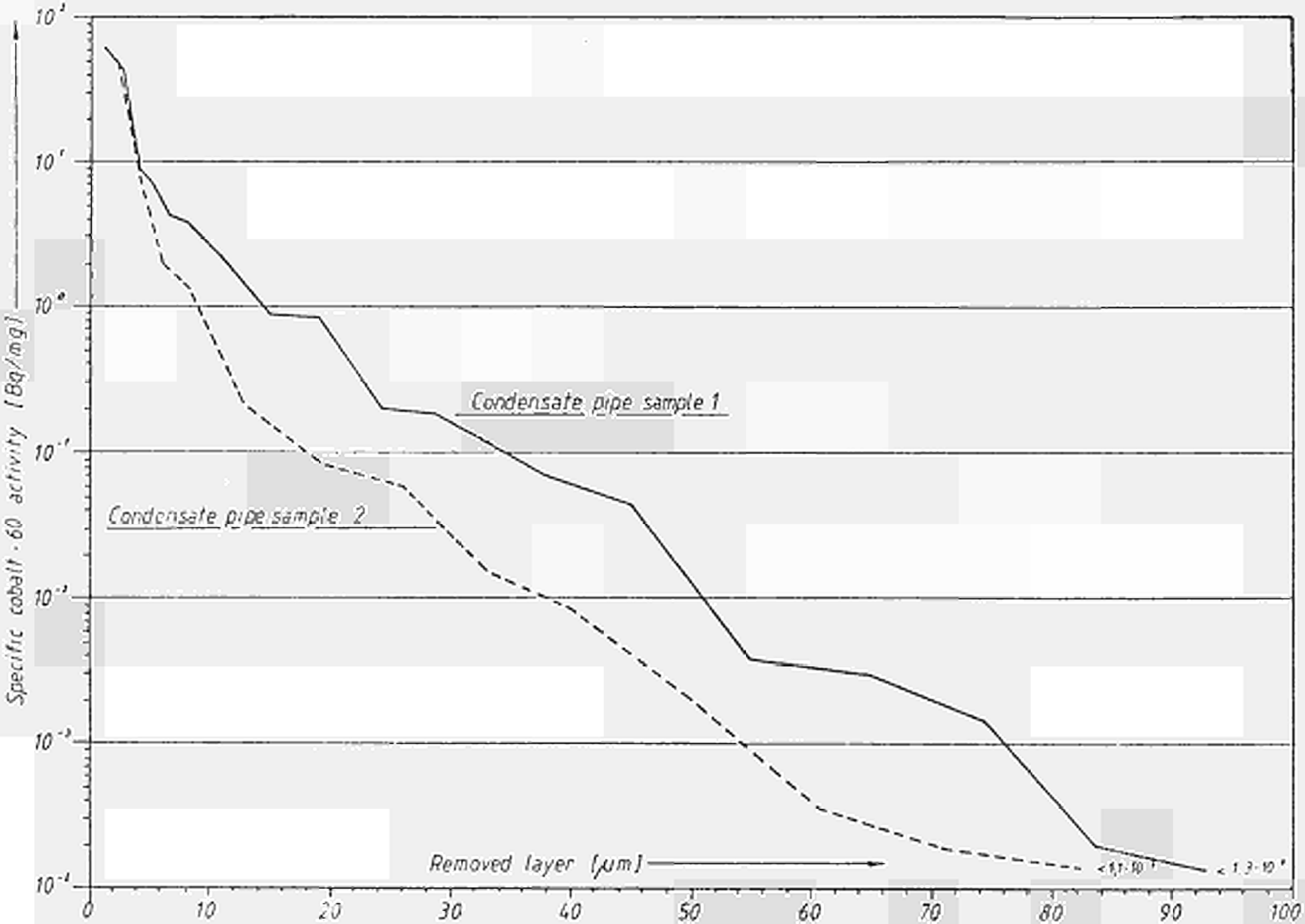

Figure 3. ^{60}Co activity profile in condensate pipe samples

2.3.3.2. Vigorous decontamination tests

The preliminary tests on material removal by acid treatment of titanium-stabilized AISI 304 stainless steel were finished. The aim of these tests was to achieve uniform removal without pitting, using acid concentrations as low as possible. With an acid content of 1% (HNO_3/HCl = 1:1), about 25 µm were removed in 16 h at 60°C.

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*

2.4.3.1. Sample characterization

Most contaminated samples originate from the Garigliano plant. Additionally, two carbon steel samples from the Cirene Assembly Reactor Test loop (CART loop) of the Essor reactor (JRC Ispra) were examined.

Gamma spectrometry with a Ge(Li) detector, on Garigliano samples, showed only the presence of ^{60}Co; traces of ^{137}Cs (about 1/1000 of ^{60}Co) were detected by coincidence analyses. On prepared oxide specimens subjected to radiochemistry treatment (for alpha spectrometry), traces of

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* Information on this research is also published in Ref. 4.

^{125}Sb and ^{144}Ce were also found.

The maximum contamination level (1-2 $\mu Ci/cm^2$) was measured on samples from the baffle plate of the Garigliano Secondary Steam Generator A (SSG/A). Preliminary oxide alpha spectrometry showed a total activity of 0.006 $\mu Ci/cm^2$ with the following composition: ^{238}Pu: 38%; ^{241}Am: 38%; $^{239+240}Pu$: 23%; ^{244}Cm: traces. The separation effectiveness was about 30%.

On samples from the Garigliano 24" safe-end pipe, an oxide layer of about 10 μm thickness was found by optical microscopy. First observations on secondary electron image by scanning electron microscopy showed an oxide composed by many crystals uniformly spread on a very thin and compact sublayer; semi-quantitative analyses showed that the crystals are mainly nickel-ferrite spinels ($NiOFe_2O_3$) with traces of chromium. Traces of copper and manganese were also found in some samples.

2.4.3.2. Static decontamination tests

Three batch decontamination test series were performed so far.

In the first series, the influence of ultrasound was investigated. In the device used, the activity normally reached a constant level after about 2 min of ultrasonic treatment (Figure 4). Ultrasound enhanced decontamination very effectively when combined with chemical action.

For the second series, activity versus test time is shown in Figure 5. In this series, the influence of reducing the acid concentrations was studied; generally, it causes slowing of the decontamination process. Among the tested solutions, the mixture of hydrofluoric and nitric acid (1.5 vol.% HF + 5 vol.% HNO_3) appears to be the best. Hydrochloric acid, particularly at concentrations of 1-2 vol.%, seems also very effective.

Three tests were performed with organic acids. Two of these tests, using 3 wt% oxalic + 3 wt% citric acid buffered with ammonia at pH=3.5 (one test also with hydrazine as reductant), showed very low decontamination factors (1.3-1.5). The third test, with a typical two-step procedure, showed total removal of the oxide and a very high decontamination factor; however, this result is still unexplained.

In the third test series, performed on samples from the baffle plate of SSG/A, the influence of a commercial inhibitor on 1.5 vol.% HF + 5 vol.% HNO_3 and 1 vol.% HCl solutions was investigated; the preliminary results show that the inhibitor does not change the etching morphology but delays the removal process; this effect is very large with HCl.

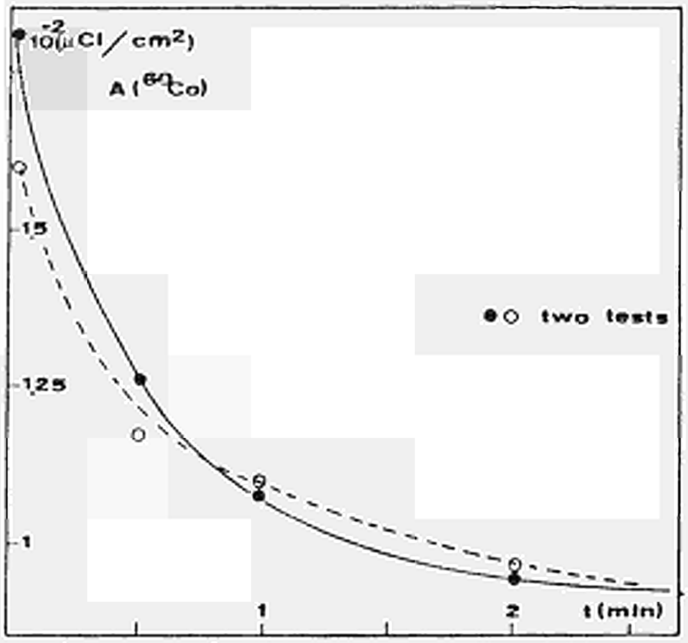

Figure 4. Influence of ultrasound in a static decontamination test

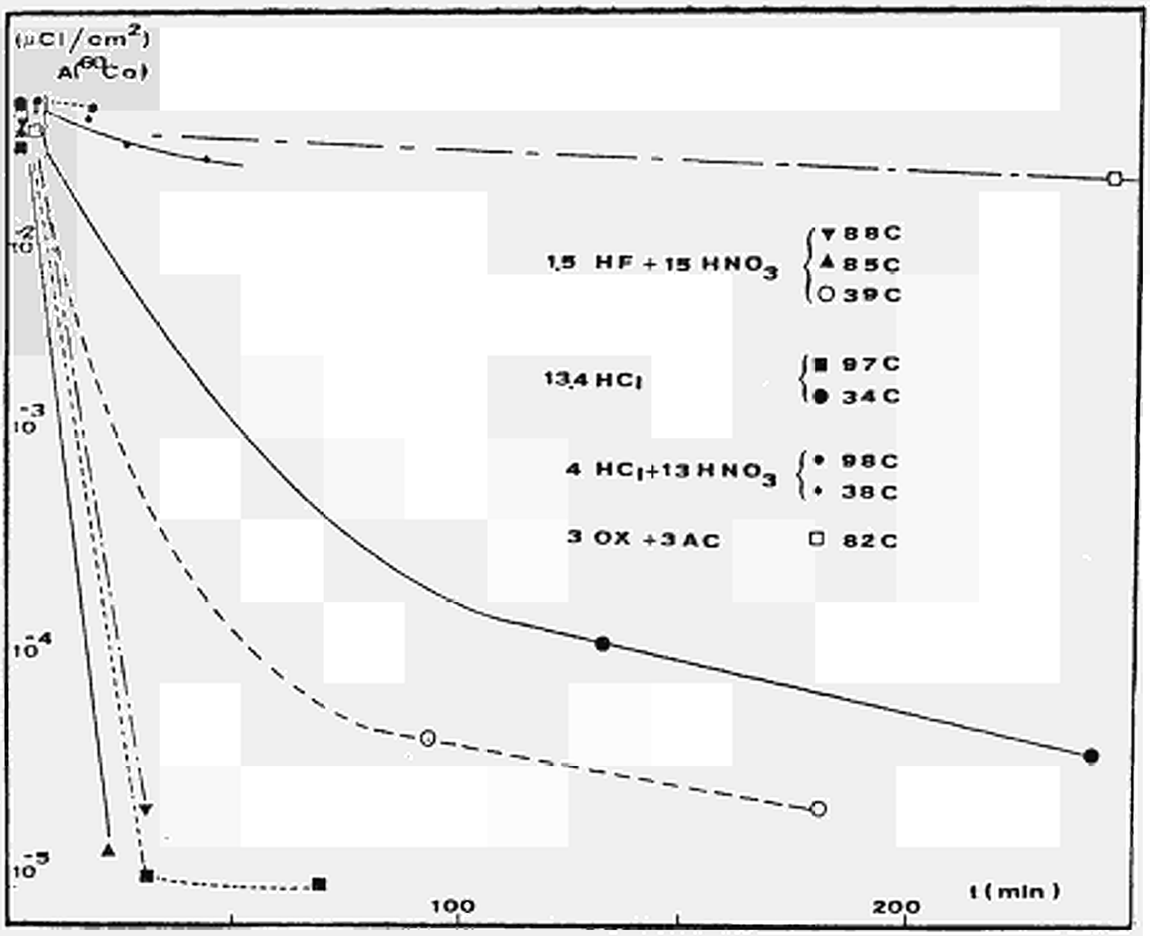

Figure 5. Static decontamination tests with various chemical agents

When the oxide layer is totally removed, the decontamination factor is very high (>1000). Residual contamination levels are similar to background and below the limits indicated for unrestricted release of material in some countries.

No cracks were found after decontamination. Spent decontaminants from strong mineral acids are treated by particulate removal, followed by metal precipitation with hydrates.

2.4.3.3. Dynamic decontamination tests in the DECO loop

Construction of the dynamic test loop was completed in March and the commissioning tests were performed in June and July 1982.

Five decontamination tests were carried out in 1982. They are summarized in Table 5. The evolution of decontamination with time, in the tests No. 1 and No. 2, is shown in Figures 6 and 7, respectively.

Table 5. Conditions and results of dynamic decontamination tests

Test No.	1	2	3	4	5
Sample material	Carbon steel	AISI 304	AISI 304	AISI 304	AISI 304
Sample origin	CART loop	Garigliano	Garigliano	Garigliano	Garigliano
Decontaminant (vol.%)	organic acids (1)	1.5 HF + 5 HNO_3	organic acids (2)	1 HCl	0.8 HF + 2.5 HNO_3
Temperature (°C)	60-70	80	80-90	85	85
Flow rate (m/s)	1-1.3	1	1	1	1
Test time (h)	6.5	0.5	23 (3)	0.82	0.47
Decontamination factor	16	>1000	1.7	3.5	>1000

(1) 35 g/l of oxalic acid + 35 g/l of citric acid + 12 g/l of hydrazine + 10 g/l of inhibitor at pH=3.5 with ammonia.

(2) 1st step: 30 g/l $KMnO_4$ + 100 g/l NaOH; 2nd step: 30 g/l of oxalic acid + 30 g/l of citric acid at pH=3.5 with ammonia.

(3) 1st step 0.75 h and 2nd step 15.75 h followed by 1st step 0.75 h and 2nd step 5.75 h.

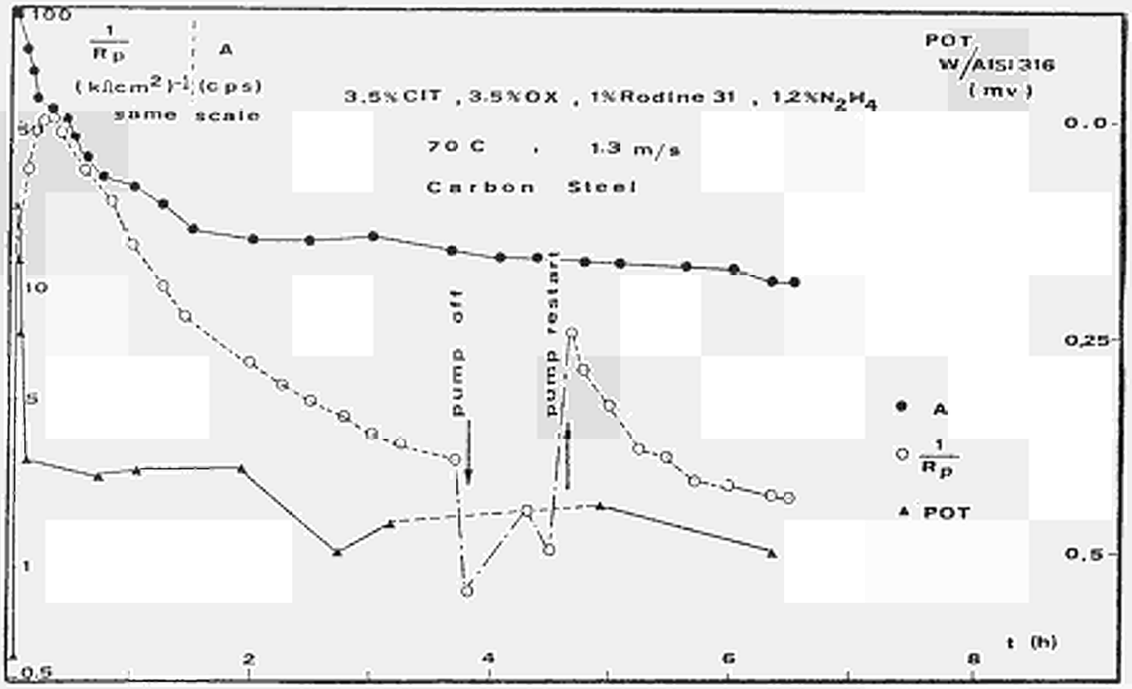

Figure 6. Decontamination of carbon steel in a loop test

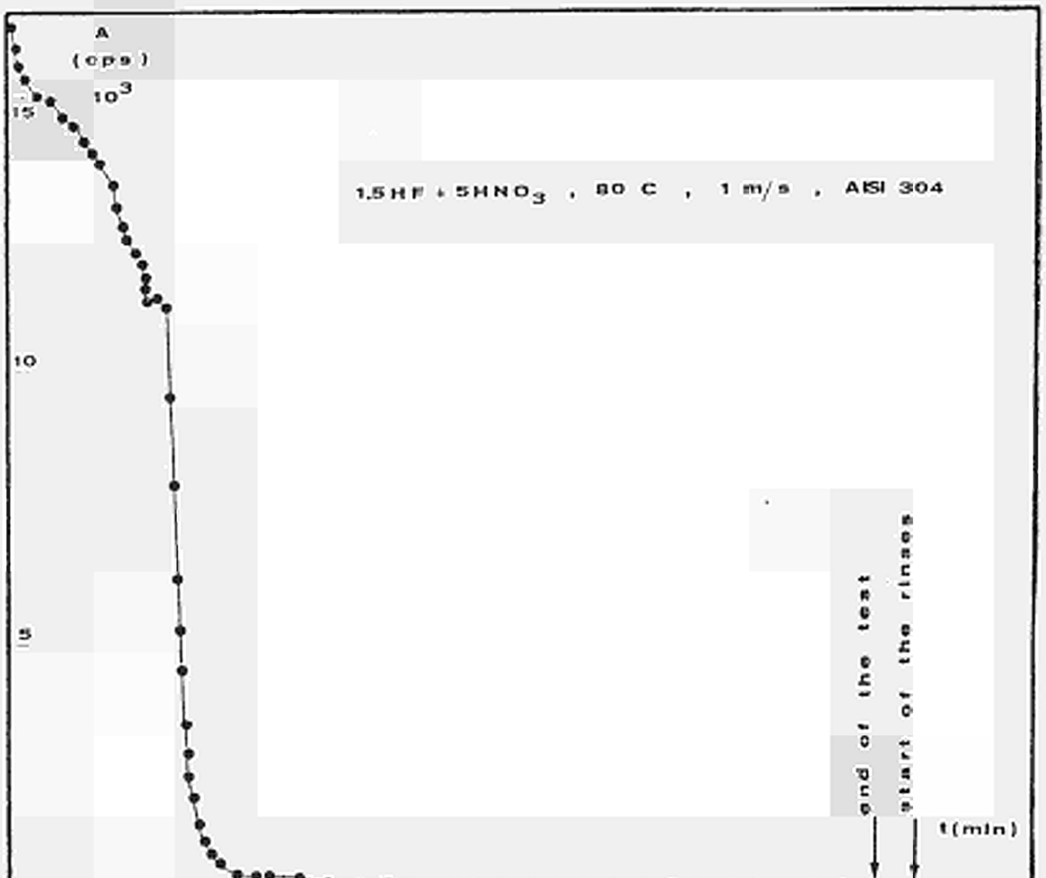

Figure 7. Decontamination of stainless steel in a loop test

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 shut down 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

2.5.3.1. Penetration of contamination into concrete structures

The penetration of contamination into the concrete walls and floors was measured on boring samples of 55 mm diameter and 30 mm length, taken at 17 selected locations of the controlled area. Cesium showed a much higher penetration capability than cobalt.

Almost all concrete surfaces of the plant, with the exception of 200 m^2 (0.4%), are covered with a protective layer. Figure 8 shows a typical activity penetration profile for a coated wall. More than 90% of the activity is in the coating. Even in special areas of very high contamination and frequent damages of the protective paint during plant operation, the penetration does not exceed 7 mm. Assuming a decay time of 30 years between shutdown and dismantling, and a limit for unrestricted release of 3.7 Bq/g, it was estimated that the necessary treatment of 50,000 m^2 of surface will result in 150 m^3 of contaminated concrete and 20 m^3 of coating.

2.5.3.2. Decontamination tests

Previous tests had shown, that the usual two-step chemical decontamination procedures and attack by aggressive acids would require at least three consecutive treatments with fresh solutions, to reduce contamination on primary loop pipes below the limit for unrestricted release. Thereupon, electrochemical decontamination, which seems more promising as an economically suitable procedure, has been investigated.

70% phosphoric acid was chosen as electrolyte and the current density was varied between 0.31 and 3.12 A/cm^2. The decontamination velocity is roughly proportional to the density and high densities are necessary for short treatment times (see Figure 9).

This reference method did not in all cases achieve complete oxide removal. Pre-treatment with commercial detergents proved to be very effective on ferritic steel samples from the condensate duct, removing 80-90% of cobalt-60 and 30-40% of cesium. A subsequent electrolytic treatment with 23.4 A/dm^2 removed almost all remaining contamination.

Stirring the solution vigorously did not significantly improve the decontamination. Reducing the phosphoric acid concentration from 70% to 40% resulted in preferred attack of surfaces where oxide layers were thin, and in higher residual contamination. A mixture of concentrated sulphuric acid and 70% phosphoric acid brought no improvement, even when vigorously stirred.

Tests with and without detergent pre-treatment were repeated on samples from the primary steam duct. Electropolishing is very efficient on this material, which has a higher chromium content. On these samples, the total decontamination factor was slightly increased by the pre-treatment.

Analyses of the electrolyte showed a steady increase of the metal content (Fe, Cu, Ni, Cr, Mn, Co). No reduction of efficiency was observed with iron contents up to 23 g/l.

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

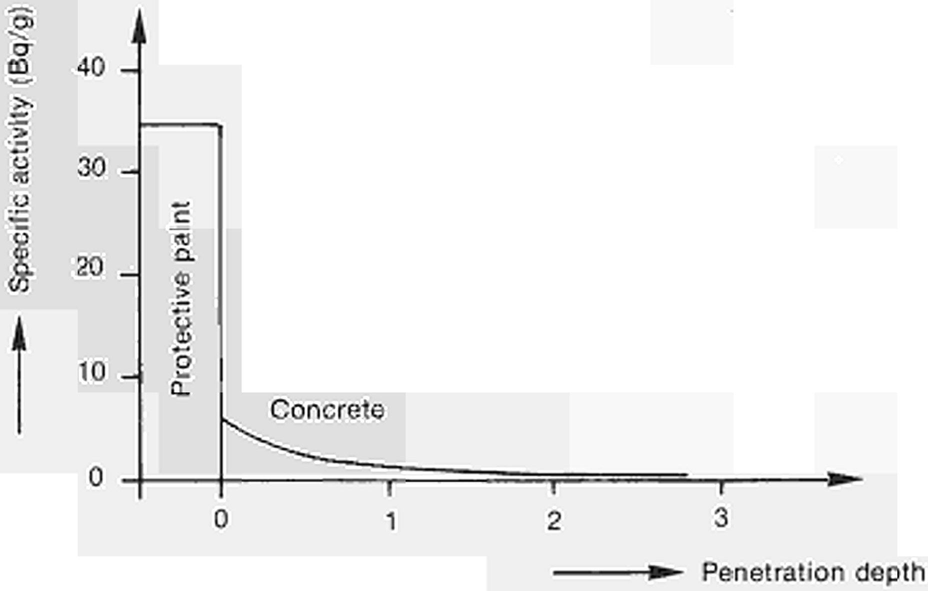

Figure 8. Typical activity penetration profile for a coated concrete wall

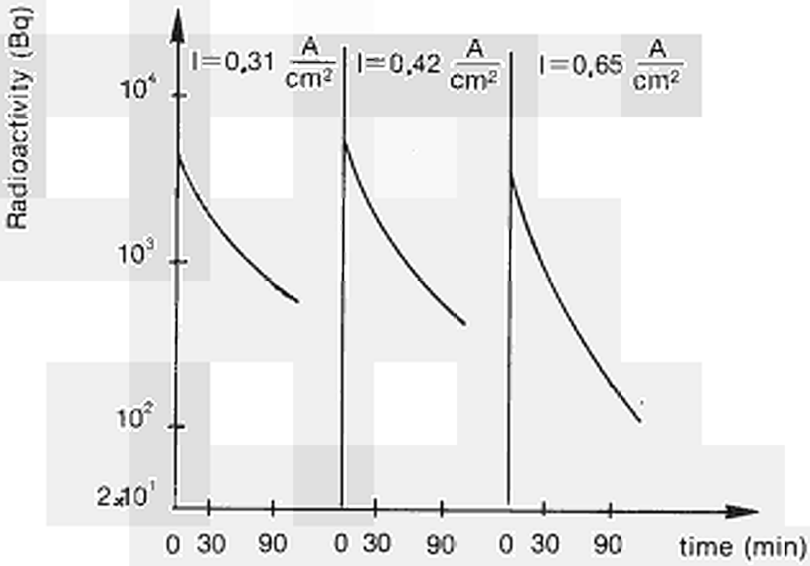

Figure 9. Electrochemical decontamination at various current densities

radioactive effluent. Less decontaminant is required, if the decontaminant is applied by means of a gelatineous carrier substance.

The objective of this research is 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

2.6.3.1. Decontamination of artificially contaminated samples

Samples of stainless steel, mild steel, an aluminium magnesium alloy, copper and plexiglass were contaminated by applying a solution containing either fission products (^{137}Cs, ^{90}Sr, ^{90}Y) or ^{239}Pu. After drying for 24 h and activity measurement, the samples were rinsed in water and measured for fixed activity. The samples were then decontaminated by applying gels containing decontaminants and, for comparison, by immersion in an aqueous solution of decontaminants. Finally, the residual activity was measured.

The efficiency of decontamination is valued by the decontamination factor (DF) defined as the ratio of fixed activity to residual activity.

The following mixtures of gels and decontaminants were studied:

- glycerophtalic gel as carrier of sulphamic, oxalic, citric, tartaric and sulphuric acid;
- glycerophosphoric gel alone and as carrier of phosphoric acid or of a commercial phosphoric detergent;
- silica gel as carrier of sulphuric or nitric acid;
- diopside gel as carrier of sodium hydroxide and potassium permanganate.

On stainless steel (AISI 304L), compounds of glycerophosphoric gel gave decontamination factors between 20 and 200, values obtained likewise with silica gel as carrier of nitric or sulphuric acid. Decontamination for plutonium on stainless steel is effective with glycerophtalic gel as carrier of sulphuric acid (DF=165) or with diopside gel as carrier of sodium hydroxide and potassium permanganate (DF=740).

On mild steel, decontamination factors higher than 100 were obtained with glycerophtalic gel as carrier of sulphamic acid and a tensio-active commercial product (nonylphenol-polyglycol-ether), or with silica gel as carrier of sulphuric acid.

Aluminium is well decontaminated by diopside gel as carrier of sodium hydroxide and potassium permanganate for plutonium and by silica gel as carrier of sulphuric acid for fission products (DF=810).

On copper, sulphuric acid carried by glycerophtalic gel or silica gel gives good decontamination (DF>80).

Plexiglass is well decontaminated by glycerophosphoric gel as carrier of phosphoric detergent or by diopside gel as carrier of sodium hydroxide and potassium permanganate.

2.6.3.2. Treatment of secondary waste

The efficiency of decontamination by evaporation of aqueous radioactive wastes is not changed if 10% of contaminated gels are added to these wastes.

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 is to develop highly effective methods for the decontamination of steel for decommissioning purposes.

Chemical decontamination methods will be 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 will be 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.

The methods thus optimized will be evaluated with a view to their industrial application, taking into account, in particular, their effectiveness, limitations, costs and radiological consequences.

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

2.7.3. Progress and Results

Decontamination tests were performed on samples from an electromagnetic filter used for 20 months in the primary circuit of a PWR (Advanced Prototype Nuclear Steam Supply System at Cadarache). For these stainless steel parts, two-step chemical treatments give good results, if strong acids are used in the second step. The main part of the research was aimed at finding the best pre-treatment allowing easy decontamination and secondary waste treatment.

With pre-treatment by soda permanganate (1 h at 85°C), followed by treatment at room temperature, 0.5 mol/l HF + 2 mol/l HNO_3 gave the best results. The efficiency was enhanced by treatment at 85°C. Sulphuric acid in place of nitric acid also produced good decontamination.

Of the other solutions tested, only ozone pre-treatment and electrolytic oxidation of the surfaces with a contaminated oxide layer appear to be promising.

Ozone sparging tests were conducted under alkaline and acid conditions and in demineralized water. Only the results obtained with demineralized water were encouraging. The mechanical action of sparging was found important.

The effectiveness of the pre-treatment can be evaluated by the amount of chromium removed from the oxide layer. This amount was approximately the same with the following pre-treatments:

a) soda permanganate solution (1 h);

b) anodizing in soda medium (100 mA/cm^2, 0.5 h);

c) sparging with 2.5% ozone in oxygen (7 h).

Treatment with fluoric-nitric baths (0.05 mol/l HF + 2 mol/l HNO_3; 8 h at 25°C) gave decontamination factors of 2000, 170 and 400 with the above-mentioned pre-treatments a, b and c, respectively. In all experiments, the results with ozone pre-treatment were found to be 2 to 5 times lower and the results with electrolytic oxidation about 10 times lower than those with soda permanganate.

The additional application of ultrasound during the decontamination treatment regularly produced 5 to 10 times higher decontamination factors.

For samples pre-treated by ozone sparging in demineralized water, various methods of chemical and anodic oxidation were tested (Table 6).

Table 6. Decontamination tests on samples pre-treated by ozone sparging

Type of bath, concentration (mol/l)	Nitric acid: 2 Hydrofluoric acid: 0.5			Nitric acid: 2 Hydrofluoric acid: 0.05			Anodic oxidation I_a = 100 mA/cm² Oxalic acid: 0.16	
Temperature (°C)	25		85	25			25	
Time (h)	2	4	1	2	4	8	0.156	0.5
Decontamination factor	630	860	2000	140	240	400	400	2000

Regarding the treatment of spent decontaminants, it was shown that oxalic acid can be oxidized by electrolysis on a glassy carbon anode and thereby removed up to 99% from spent baths before coprecipitation of residual oxalates and of radiochemical activity by lime.

2.8. Economic Assessment of Decontamination for Unrestricted Release

Contractor: Nuklear-Ingenieur-Service GmbH, Hanau, Germany

Contract N°: DE-B-009-D Work Period: February 1981 - March 1982

2.8.1. Objective and Scope

Information on the economic feasibility of decontaminating nuclear power plant components so that they can subsequently be released without restriction is needed as a basis for the further development and optimization of concepts for decommissioning. The objective of this study was to prepare such information, with due regard to existing regulations on limit values for the unrestricted release of waste arising from decommissioning operations and taking as a basis existing decontamination techniques.

2.8.2. Work Programme: See Ref. 1, Paragraph 2.9.2.

2.8.3. Progress and Results

Authorizations for unrestricted release of material from the controlled area of nuclear facilities are mostly granted on a case-by-case basis. Agreed procedures have been set up for a number of facilities in co-operation with the licensing authorities. Considering the Euratom Basic Safety Standards and the Recommendations of the International Commission on Radiological Protection (ICRP-26), the licences have been discussed in order to evolve ideas and tendencies about possible future criteria for unrestricted release.

Available relevant information on contamination and on decontamination techniques was reviewed.

As the decontamination techniques for unrestricted release purposes are in an early stage of development, at least for stainless steel, no reliable cost data for this kind of decontamination are readily available.

The costs for the management of steel waste without extensive decontamination were estimated, considering three disposal modes, i.e., sea dumping, disposal in a salt mine and shallow land burial.

This estimate was based on average labour costs of 90 DM/man-hour and on the following waste package characteristics:

- drum volume: 200 l
- drum mass: 40 kg
- steel mass: 312 kg
- concrete mass: 268 kg (filling mass)
- filling factor: 0.2 (steel volume to available volume)

The following costs (DM per Mg net mass) were estimated for the individual operations:

Cutting	
- 10 man-hours:	900
- material:	50
	950

Transportation	
- 100 km by truck:	92
- 400 km by railway:	278
- surveillance:	30
	400

Conditioning		Disposal (three options)	
- 4 man-hours:	360	- embarking, sea dumping:	945
- material:	310	- land burial:	1400
- 3 drums:	945	- mine disposal (Asse):	3200
	1615		

The total costs (DM per Mg net mass) of the above-mentioned operations are as follows:

- total cost with sea dumping: 3910
- total cost with land burial: 4365
- total cost with mine disposal (Asse): 6165

The final report on this study is being published (Ref. 5).

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. Thermal and Mechanical Dismantling Techniques

Contractor: Transnuklear GmbH, Hanau, Germany

Contract N°: DE-C-001-D Work Period: March 1980 - June 1982

3.1.1. Objective and Scope

The purpose of this research was to try out various dismantling techniques on non-radioactive test specimens representative of components and structures of light water reactors and to evaluate these techniques as to performance, radiation exposure of personnel and environmental impact.

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

3.1.3. Progress and Results

The research was completed in the first half of 1982 by the works described in the following.

3.1.3.1. Complementary tests with the powder cutting technique

Tests with the powder cutting technique were carried out in order to determine the dust content and particle size distribution in the off-gas. A heavy concrete plate (thickness: 60 cm; density: 4 g/cm^3) and a carbon steel I-girder were used as test specimens.

The measured dust contents were 3.0 g/m^3 and 1.2 g/m^3 for cutting concrete and steel, respectively. The results obtained with the oxygen lance had been 0.9-1.5 g/m^3 and 2.3 g/m^3 for concrete and steel, respectively. However, in comparing these data, the larger cutting speed of powder cutting has to be taken into account.

The largest fraction of particles occurred at a size of 0.3 μm. Compared with the oxygen lance, powder cutting produced a smaller proportion of fine particles, especially when steel was cut.

3.1.3.2. Investigation of sawing techniques

Using a bow saw, cutting tests on a carbon steel I-girder were carried out. The results are compared in Table 7 with those obtained with thermal techniques.

Table 7. Cutting of an I-girder using various techniques

Cutting technique	Cutting time (min)	Waste (kg)*
Bow saw	15	0.56
Oxyacetylene torch	23	0.50
Oxygen lance 3/8"	2.1	4.1
Powder cutting	1.2	1.3

* Chips (sawing) or slag (thermal techniques)

Moreover, the following sawing techniques were tested:

- a high-speed, dry-operation circular saw (diameter: 520 mm) for cutting pipes;
- low-speed circular saws (diameters: 560 mm, 500 mm) for cutting round steel;
- a high-speed, corundum-coated disk (diameter: 905 mm) for cutting concrete.

3.1.4. Conclusions

The following conclusions were drawn from the work performed under this contract.

Of the three oxygen lances (1/4", 3/8" and 1/2") tested, the 1/4" lance is best for cutting heavy concrete; in particular, the slag deposit is very small. For steel, however, the 3/8" lance is superior.

The powder cutting technique performed in comparative tests on concrete and steel is better than the other thermal techniques with respect to cutting rate, suitability for remote control and, to some extent, slag deposition. The cutting depth on concrete achievable with the available powder cutting equipment was limited to about one metre.

The oxyacetylene torch, whose application is restricted to ferritic steels, is characterized by little dust and slag formation, low gas consumption and negligible tendency to malfunction.

The plasma torch produced good results in cutting austenitic and ferritic steel plates; it is, however, prone to malfunction, especially on workpieces with irregular geometry.

Thermal cutting techniques produce a large amount of fine aerosols. With an exhaust flow of 2200 m^3/h, the dust content in the off-gas ranged from 1 to 3 g/m^3 as a function of the cutting technique and the material. The largest fraction of particles occurs at a size of 0.3 µm. Thermal cutting of radioactive material, therefore, requires a major off-gas filtering effort. In particular, filters must be fit for cleaning in order to attain adequate service time. Two such filter systems were tested, i.e. a cartridge-type filter, which showed only moderate efficiency (81-85%), and a HEPA filter, which achieved an efficiency of 99.96%.

Sawing techniques have advantages in producing almost no airborne contamination and little secondary waste. Also, they are suitable for remote operation. However, due to the limitation of cutting depth, normal sawing equipment cannot be used for the initial dismantling of large components.

The final report on this research is being published (Ref. 6).

3.2. Plasma Techniques for Cutting Mineral and Metal Materials

Contractor: Ansaldo Meccanica Nucleare, Genoa, Italy

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

3.2.1. Objective and Scope

This research relates to plasma-arc cutting and its basic aim is to provide more information on the process, 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 plasma-arc cutting technique will be studied independently of the type of power station in which its use is envisaged; however, where specific examples have to be considered, the contractor will, as a licensee for Boiling Water Reactor stations, refer to this type of plant.

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

3.2.3. Progress and Results

During 1982, two techniques have continued to be developed, i.e. the plasma torch for cutting steel components and the oxygen lance for cutting concrete structures.

3.2.3.1. Plasma cutting

AMN, in co-operation with Messer Griesheim, continued the optimization of the cutting parameters.

Two types of torch were used, i.e. the plane-shaped electrode torch and the pointed electrode torch. The cutting tests were performed in air and under water with steel AISI 304 samples of 40 mm thickness. These tests showed that:

- the utilization of both torch types presents no difficulties;
- cutting rates are reproducible;
- the nozzle life is satisfactory, provided the distance between the nozzle and the work-piece is correct.

A second series of tests was performed to investigate the possibility of cutting tubes close to a wall. This problem relates to the severing of a recirculation inlet tube in a BWR. The analyses of these tests are in progress.

At the beginning of 1983, another series of tests will be started, in order to determine the maximum thickness that can be cut and to characterize the dust and aerosols produced in cutting different steel plates.

3.2.3.2. Oxygen lance

AMN, in co-operation with Fondibeton-ATMC, performed a series of tests cutting a block of reinforced concrete which represents typical reinforced-concrete walls existing in a BWR plant (Caorso Nuclear Power Plant). The time needed to pierce a hole through one metre of reinforced concrete ranged from 400 to 600 seconds. The average lance consumption was about 1 m/min. The presence of many reinforcing iron bars had some effect in slowing down the perforation process and caused sometimes large deviations from the straight path. The best slope angle in the cut is about 25°, a horizontal cut is not recommended. Since the large amounts of lapilli which are ejected from the cut could damage components that are behind the structure being dismantled, it is recommended to remove or to shield the components that are to be preserved.

3.2.3.3. Analysis of radiological protection and safety aspects

The experiments relating to radiological protection and safety aspects, which are to be carried out at the ENEA Casaccia Centre, have been delayed due to problems of commissioning the equipment and instrumentation. The tests are now planned to be carried out in 1983.

3.3. 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

3.3.1. 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.

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

3.3.3. Progress and Results

In 1982, the development of large diamond-tipped circular saws was continued satisfactorily. Two blades were manufactured, i.e.:

- a blade of 0.6 m diameter, to cut the relatively thin surface layer of concrete covering the reinforcement bars and the bars themselves;
- a blade of 2.5 m diameter, to cut up to 1 m deep into high quality

concrete.

Construction of the power pack and the saw unit and delivery to the test site at Portishead Power Station, implementation of work and preliminary testing were successfully completed.

3.3.3.1. Implementation of work

For the implementation of the saw unit, the following work had to be carried out.

The power pack and saw frame hydraulic systems were connected and operated at full pressure of about 200 bar to demonstrate their serviceability.

The tracking of the saw frame was checked and adjusted so that it was as exact as possible. A misalignment of only 7 minutes of arc produces a deviation of +2.5 mm at the leading edge and -2.5 mm at the trailing edge of the 2.5 m diameter blade when cutting 1 m deep. This results in hysteresis in the blade, more power being needed to cut a given length of kerf, and a kerf wider at the top than the bottom.

The response characteristics of the controller were adjusted so that the power required to feed or plunge the blade into the concrete, or to traverse the blade through the concrete did not exceed the power available and thus cause a stall. In particular, the saw has to decelerate when moving out of the relatively free cutting concrete and into the more resistant reinforcing steel. This has now been achieved.

A method of moving the saw frame from one test site to another without affecting the positional accuracy of the sawing was developed. This is now a matter of routine.

Factors affecting the tracking, such as external forces applied by the hydraulic hoses, were reduced to a minimum.

3.3.3.2. Preliminary tests

The preliminary tests with the 0.6 m blade unit were carried out and produced the following results. The blade can be plunge cut, i.e., saw straight into concrete with only minor forward movement, to a depth of 150 mm in less than 40 s. When cutting a kerf 150 mm deep in granolithic concrete, the saw will traverse at a rate of 275 to 300 mm/min and will continue through 20 mm reinforcing steel without overloading at optimum rate as set by the controller.

3.4. Plasma-oxygen Cutting of Steel Pressure Vessels

Contractor: Salzgitter AG, Salzgitter, Germany

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

3.4.1. 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.

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

3.4.3. Progress and Results

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

3.4.3.1. Cutting tests on thick plates of ferritic steel clad with stainless steel

The systematic test series with the combined plasma-oxyacetylene torch have been extended to include the thickness range of clad plates of 400 to 600 mm in various cutting positions, i.e.:

- flat position (vertical torch moving horizontally);
- horizontal position (horizontal torch moving horizontally);
- vertical position (horizontal torch moving downwards).

The base material used was ferritic steel grade St 37-2 (thickness: 150, 200, 250, 300, 400 and 600 mm) and the cladding consisted of austenitic steel grade X10CrNiTi 18 8 (thickness: 10 mm). These tests confirmed that the cutting of clad ferritic steel with this technique is possible up to a thickness of 600 mm. However, the 600 mm specimens required maximum cutting speed of the oxyacetylene torch in order to avoid break-down of the plasma arc; as a consequence, a lower quality of cut must be accepted.

The vertical cutting position proved to be the most favourable one, for the following reasons: the slag falls onto the hot end of the kerf; the heating of the oxyacetylene torch can be reduced by setting at appropriate distance to the workpiece; the hot process gas rises within the kerf and contributes to a better quality of cut. In the horizontal cutting position, the slag falls onto the flank of the kerf, which leads to an unclean cut. In the flat cutting position, the oxyacetylene torch is strongly heated up by rising hot gas.

Table 8 shows the optimized cutting parameters for the flat and vertical cutting positions.

Table 8. Optimized cutting parameters

	Unit	Flat position						Vertical position	
Base material thickness	mm	150	200	250	300	400	600	400	600
Nozzle Ø of plasma torch	mm	4.5							
Angle of plasma torch	degree	45				30			
Current	A	200							
Voltage	V	150				130			
Nozzle of oxyacetylene torch	-	type 200-300 mm				type 300-600 mm			
Distance oxyacetylene torch - workpiece	mm	8			12	37		35	
Angle of oxyacetylene torch	degree	5				0			
Pressure of fuel gas	bar	0.22				1.1			
Pressure of fuel oxygen	bar	3.5				7			
Pressure of cutting oxygen	bar	14	18	20		17			
Cutting speed	mm/min	140	120	90	60				
Cutting depth	mm	160	210	260	310	410	610	410	610
Width of kerf	mm	12-20				20			
Distance between torches	mm	30				40		39	

3.4.3.2. Characterization of aerosols and testing of filter systems

For these investigations, aerosols were produced by plasma cutting of ferritic and of austenitic steel specimens and by oxyacetylene cutting of ferritic steel specimens. The specimen thickness was 20 and 40 mm.

The largest fraction of aerosol particles were found in sizes of 0.2

and 0.12 μm for the austenitic and the ferritic steel specimens, respectively. On cutting of ferritic steel both with the oxyacetylene torch and with the plasma torch, the largest fraction of particles occurred at the same size. When cutting with the oxyacetylene torch, the range of particle sizes was narrower and the proportion of fine particles was lower.

Three filter systems connected in series were tested. The following average efficiencies were obtained for each of the systems:

- bulk material type filter: 87.5%
- box type filter: 91%
- barrel type HEPA filter: 84%

Therewith, the overall efficiency of the three systems is 99.8%. However, the bulk material type filter, used as pre-filter, did not attain adequate service life and should therefore be replaced by a different filter.

3.5. Dismantling of Concrete Structures and Metal Components Using Laser

Contractor: FIAT TTG SpA, Torino

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

3.5.1. 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.

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

3.5.3. Progress and Results

The development of the cutting technique using the 15 kW carbon dioxide laser has been continued in 1982. Due to laser malfunction, the tests scheduled for the second half of 1982 had to be postponed to 1983.

3.5.3.1. Development of equipment

Nozzles of various diameters for one or several assistant gas jets were manufactured. In parallel, a nozzle design was developed which permits to supply the gas co-axially to the laser beam.

3.5.3.2. Cutting and drilling of concrete

Preliminary experiments on concrete produced encouraging results. Oxygen at a pressure of 7 bar as assistant gas and a needle-valve nozzle of 1.5 mm diameter proved to be appropriate. Cuts of 90 mm depth were performed with 10 kW power and 0.02 m/min velocity. The kerfs were regular and only 2 mm wide. Furthermore, holes of 8 mm diameter and 65 mm depth were drilled, using an impulse of 5 s at a power of 10 kW.

3.5.3.3. Cutting of steel

Cutting tests on steel were done with 10 kW power and oxygen as assistant gas. Two types of specimens were used, 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.

The position of the beam focus was optimized with respect to cutting velocity. For the stainless steel specimens, moving the focus from the specimen surface to the optimum position, which was found to be 15 mm beneath the surface, allowed to rise the cutting rate from 0.21 m/min to 0.27 m/min. For the mild steel specimens, a cutting rate of 0.41 m/min was achieved. In order to achieve these maximum cutting rates, the oxygen flow must exceed a critical value, which was found to be 6 m^3/h and 7 m^3/h for the stainless steel specimens and the mild steel specimens, respectively.

3.5.3.4. Analysis of by-products

Off-gas samples taken during the cutting tests described in the two preceding paragraphs were analysed. The following dust concentrations, corresponding to an exhaust flow rate of 1350 m^3/h, were determined for the various specimen materials (rounded average values):

- stainless steel: 40 mg/m^3;
- mild steel: 20 mg/m^3;
- concrete: 370 mg/m^3.

The largest fraction of particles was found in the size range of 0.3-0.5 μm. However, the probably important fraction of particles that are smaller than 0.3 μm could not be determined with the apparatus used for these first measurements.

Preliminary analyses for trace element concentrations in the dust were carried out.

3.6. Explosive Demolition Techniques for Concrete Structures

Contractor: Taylor Woodrow Construction Ltd, Southall, United Kingdom

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

3.6.1. 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.

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

3.6.3. Progress and Results

3.6.3.1 Cratering Tests

Preliminary tests: These tests were carried out on an existing concrete block to help decide target sizes to be used in subsequent work. 4 gram charges at 200 mm depth of burial were fired near the four corners of the concrete block. The distances from the edges were varied between tests. From these preliminary studies a target size of 0.5 m dia x 0.5 m thick was considered to be suitable for the small-scale tests.

Depth of burial and charge weight effects - untamped (7 tests): Depth of burial effects on cratering were checked using untamped 4 gram charges in models with depths of burial of 50, 75, 100 and 125 mm. The charge weight effects on cratering were investigated using untamped 6, 8 and 10 gram charges at 125 mm depth of burial. Only one true crater was formed, i.e. with 50 mm depth of burial. In all other tests, a camouflet or partial camouflet was formed. These tests indicated that considerable energy was lost in "blow back" because the charge holes were untamped.

Depth of burial effects - tamped (4 tests): Repeat tests with improved tamping were carried out, for the four cases in which depth of burial effects were investigated earlier. Damage in all cases was more severe than with the untamped tests and with crater diameter increasing with depth of burial up to a critical depth of 100 mm. However, other damage to the models indicated that the diameter of the models may have been too small so that boundary proximity produced effects giving unrepresentative crater diameters.

Concrete strength effects - tamped (2 tests): A weak and a strong model were tested to complement a mean strength test carried out earlier. The results indicated that variations in concrete strength had very little effect on cratering. Size of the models and boundary proximity effects may, however, have masked any differences.

Reinforcement effects - tamped (3 tests): Three models were tested in which the front faces were reinforced. The indications were that the presence of reinforcement distributed the explosive forces over a wider area resulting in models with larger scabbed areas. It would appear that a slight reduction in crater diameters was also obtained.

Scaling effects - tamped (3 tests): The validity of model techniques and scaling effects have been assessed by testing three structures in which linear dimensions were increased by a factor of four over previously tested small-scale models. The results indicated there was very close agreement in damage characteristics between the models of both sizes with crater parameters scaling up in proportion.

Shape effects - tamped (3 tests): Earlier tests had indicated that the size of the models being used might be too small with edge effects influencing the test results. To investigate this, three large diameter models, but with characteristics similar to previous models, were tested. These initial tests indicated that the results from the smaller models may have been exaggerated by boundary effects. Further tests are to be carried out to investigate this in detail.

3.6.3.2. Boring tests with shaped charges

Five tests were carried out in which single shaped charges were used to bore holes into concrete targets. Each charge was mounted with its axis vertical and with the jet directed vertically downwards onto the concrete targets. Stand-off distances and type of shaped charges were varied between tests. The results have given promising possibilities of using shaped charges to provide charge holes for concrete cratering work.

3.6.3.3. Analytical studies

Consideration has been given to the analytical methods which may be used for assessing charge cratering effects in concrete structures. Two methods are being studied, i.e.:

- empirical method, in which cratering information related to wartime experience of bombs in various soils and rocks is considered to assess whether correlations exist which might enable extension to small charges in concrete structures;
- analytical method, in which a finite difference computer programme is studied to determine whether by the introduction of a point source of energy the model response can be seen to simulate cratering (including camouflet information) effects.

3.7. 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

3.7.1. 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.

3.7.2. Work Programme

3.7.2.1. Experimental investigations

Bench tests with the fissuration cutting method will be carried out on sheets of various materials including austenitic stainless steel, ferritic stainless steel and mild steel. Starting from 10 mm, the sheet thickness will be increased in order to identify the maximum thickness that can be cut in the case of each steel grade and of each geometric configuration.

The following configurations will be studied:

- edge cutting of flat sheets in various positions, i.e. floor, ceiling, cornice;
- circular cuts (window opening) in flat sheet;
- straight cuts proceeding from window openings in flat sheet;
- cutting of tubes, drums and tanks.

The following measurements will be carried out:

- dimensional measurements using conventional and optical methods;
- infra-red camera temperature measurements;
- strain-gage measurements.

3.7.2.2. Methodological studies

On the basis of the experimental results obtained, a manual of procedure will be drawn up which will give the technical instructions required for the application of the fissuration cutting method. These instructions will take into account the problems posed by the cutting of nuclear power plant components having special geometric configurations.

3.7.3. Progress and Results

Three cutting techniques have been used, i.e.:

- a cutting head incorporating a Tungsten Inert Gas (TIG) torch and supply of the addition alloy as a wire; in this method, which is most currently used, the work-piece is locally melted over part of its wall-thickness;
- heating with a Metal Inert Gas (MIG) torch; this method has the advantage that the addition material can be incorporated into the electrode, but it lacks the flexibility required at the early stage of development;

- high-frequency induction heating, whereby the heat is generated within the work-piece; this method makes it possible to avoid melting of the work-piece material.

The existing test benches enable close control of the cutting parameters, such as movement of the cutting head relative to the work-piece, feed rate of the addition alloy and length of the electric arc. Most cutting tests were performed on austenitic steel (AISI 316), but ferritic steel, refractory steel and nickel alloys (Inconel type) were cut too. After preliminary screening tests, four addition alloys were selected for further investigations (Cu-Si-Sn-Fe; Cu-Sn-P; Cu-Si-Mn; Cu-Sn).

The maximum cutting thickness achieved so far is 32 mm; this required to use the TIG method and, due to the relatively large melt, to position the work-piece horizontally. Horizontal cuts on vertically positioned sheets ("cornice") were performed on thinner sheets (5 mm).

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, following amendment of the contract in 1982, disposal by shallow land burial.

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

4.1.3. Progress and Results

4.1.3.1. Arisings and radionuclide inventories

By far the greatest contribution to the total inventories of radionuclides in Magnox and AGR graphites comes from activation of the graphite and its stable impurities, estimates of which were reported previously.

Contributions from other sources were now estimated. Natural uranium may be present in graphite as an impurity at levels of about 0.1 ppm; preliminary calculations on the fission of this impurity indicate that the levels of long-lived alpha emitters would be in the order of 10 MBq/t at the end of reactor life, and the levels of fission products would be in the order of 100 MBq/t.

Compared with this, the contribution of uranium present on the outsides of fuel elements would be negligible. Few data are available on the levels of surface contamination to be expected at the end of reactor life, though this is not expected to be a major contributor to either the total radionuclide inventory or the radiological impact of disposal.

4.1.3.2. Technological assessments

Graphite has many of the properties desired of solid waste forms, so packaging is the only prerequisite of most of the disposal options considered. Technological assessments of three methods of packaging graphite for disposal and the flowsheeting of a conceptual incinerator which would discharge the carbon-14 and tritium content of the irradiated graphite to atmosphere, leaving a smaller volume of solid waste for disposal, were completed.

4.1.3.3. Radiological assessments

Figure 10 gives an outline of the disposal options and associated environmental pathways considered in the study. For each option, doses to individual members of critical groups and collective doses to the population have been calculated as functions of time, and the important pathways and radionuclides have been identified. Detailed results have been presented for the reference Magnox reactor, with summaries of results for the Magnox decommissioning programme and for the reference AGR. For some options, ranges of results have been presented, corresponding to ranges of initial assumptions.

4.1.4. Conclusions

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

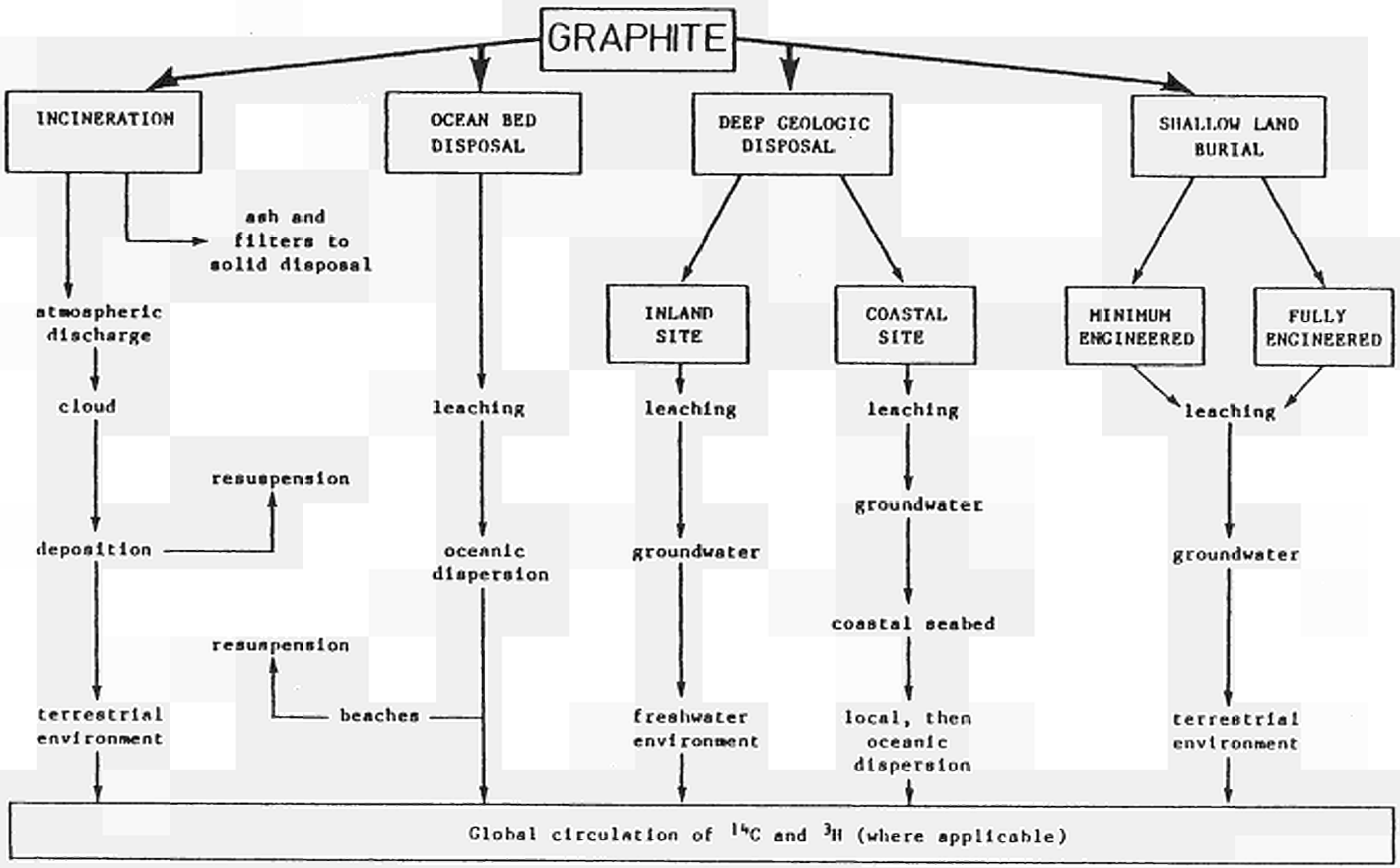

Figure 10. Simplified environmental pathways for graphite management options

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 as 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, but the effects of the different leaching conditions were found to be small.

Radiological assessments of the following four options for disposal of packaged graphite were made: on the bed of the deep ocean, in a deep geologic repository either inland or near the coast, and in a shallow land burial facility. Collective doses to the exposed population were calculated, and doses to individual members of critical groups would generally be acceptably low fractions of the dose limit.

In the case of shallow land burial it would be necessary to restrict the use of the site to prevent accidental intrusion during the lifetime of the ^{60}Co: it would be feasible to guarantee controls for the necessary period. There are uncertainties about the radiological impact of shallow land burial in the longer term, so this option merits more detailed and preferably site-specific assessment.

Incineration of graphite is possible, but presents logistical difficulties of either building several facilities at different sites or transporting the graphite to fewer centralized incinerators; the corresponding costs could vary widely. The radiological impact of incineration arises 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 to be a desirable alternative to direct disposal of solid graphite, unless the reduced volume of ash for transport and disposal is judged to override all other considerations.

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 1983

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 purposes of volume reduction and radioactivity immobilisation. 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

4.2.3.1. Characterization of materials for the melt programme

As a result of visits to nuclear power stations and to reactor research sites of the CEGB and the UKAEA, materials were supplied or reserved for the melt programme. Materials received are from the CEGB Magnox station at Trawsfynydd and from the Steam Generating Heavy Water

Reactor at the Winfrith site of the UKAEA. There are materials awaiting further consideration after initial inspection also at Winfrith and at the Berkeley and Hinkley Point Magnox and AGR stations of the CEGB.

The materials received from the Magnox station were, on the one hand, routine maintenance replacement items (shield closure gear, thermocouple mechanism) and, on the other hand, components only recoverable in reactor shutdown time (heat exchanger boiler tube, fuel chute materials).

The shield closure gear showed differential activation, mainly ^{60}Co, with one face of the gear at five times the level of the rest of the component. That this was not contamination, was shown by machining 0.1 mm from the high-level face, with no reduction in radiation level.

For the thermocouple mechanism, a flange facing into the reactor vessel had evidently received deposits from the coolant gas, as it showed beta contamination at various energies up to 0.7 MeV with a total contamination level of about 2×10^{-4} $\mu Ci/cm^2$. This was fixed contamination as determined by tests with strong decontaminants. A general activation of the component at about 2×10^{-5} $\mu Ci/g$ was identified to be chiefly ^{60}Co.

For the boiler tube material, the monitoring indicated no activation but strongly showed deposition of contamination between the finning of the tube. Gamma spectrometry showed this contamination to have high ^{60}Co and ^{137}Cs contents, but due to inaccessibility, no accurate contamination levels could be measured.

Of fuel chute material, there was a substantial quantity (about 1 t) as opposed to 1 or 2 kg only of the other materials. This whole amount was monitored and divided into three lots, i.e.: lowest activation; mixed activation and contamination; highest activation. Activation had occurred variously where the chute passed through the shield and into proximity with the reactor vessel. Contamination may have arisen through transfer from a leaking fuel element.

4.2.3.2. Melt tests at laboratory small scale

Three melts were made at laboratory small scale, i.e. melt weight 50 kg, and involved the melting of shield closure gear (1 kg), thermocouple mechanism (2 kg), boiler tube (2 kg). The melt weight was made up with non-radioactive scrap.

For the melts of shield closure gear and thermocouple mechanism, no measurable levels of activity were observed in the emission/extraction furnace fume nor in the furnace slag or steel melt product.

For the melt of boiler tube, no measurable level of activity was observed in the emission/extraction furnace fume, but the furnace slag and the steel melt product had activity concentrations of about 1 pCi/g and 2 pCi/g, respectively. Gamma spectrometry showed the slag activity to be largely ^{137}Cs with some ^{60}Co. For the metal, only ^{60}Co was identified.

4.2.3.3. Melt tests at laboratory large scale

Three melts were made at laboratory large scale, i.e. melt weight 500 kg. These were using 10 kg amounts from each of the lots of fuel chute material and 490 kg of non-radioactive scrap.

For the lowest activation level material, no measurable levels of radioactivity were found in fume, slag or steel melt products.

For the material both activated and with some contamination, the melt products were found to have activity concentrations of 2 pCi/g for the metal and 1 pCi/g for the slag. ^{60}Co was identified in the metal product. The slag activity was not identified.

For the highest activation level material, a full sampling programme was applied. The results are shown in Table 9. The derived activity concentration in the fuel chute material is thus 3.5×10^{-4} µCi/g.

Table 9. Radioactivity of products from melting fuel chute material
(Fuel chute material: surface dose rate up to 0.6 mR/h, surface contamination not measurable; dilution factor: 50)

	Activity concentration	Total activity
Metal product	7 pCi/g	3.5 µCi
Slag product	1.2 pCi/g	3.6×10^{-3} µCi
Hopper dust	1.0 pCi/g	2.0×10^{-4} µCi
Furnace fume	0.11 pCi/l	-

The subject of the melt did not generate a 'radiation area' and was not a 'radioactive substance', as these are defined in UK regulations. The metal would, however, have been 'waste' and subject to the need for disposal, except for the procedure of this trial work where a concentration <10 pCi/g was obtained. The furnace off-gas activity did not approach the permissible work place air concentration. The slag activity was within the ordinary disposal or use level.

4.2.3.4. Overall comments

The surface monitoring of furnace charge material can be misleading in respect to the actual activities found in the melt products and perhaps shows that occasional high readings are over-estimated in relation to the uniform distributions and dilutions of the melt. This seems to be the case for the melts of shield closure gear and thermocouple mechanism.

In all melts, the furnace fume activity, over which there might be the most radiological concern, was in fact very low, i.e. unmeasurable in several of the melts, and the highest level found for a short time in the duct was only 1% of the permissible whole-time work place concentration. Where surface contamination was present, most positively shown for the boiler tube material, the whole of the ^{137}Cs entered and stayed in the slag, whereas most of the ^{60}Co but not all was transferred to the metal.

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

Contractor: Ecopol S.A., Paris, France

Contract N°: DE-D-003-F Work Period: January 1981 - March 1982

4.3.1. Objective and Scope

Thermosetting resins possess the characteristics necessary to produce particularly durable and resistant coatings. The aim of this research is to demonstrate the feasibility of applying thermosetting resins on contaminated metal components in layers of various thicknesses and to evaluate their effectiveness in immobilizing the contamination both for components to be placed directly into an ultimate repository and for components to be cut up and further conditioned. The work involves tests on non-radioactive and on active specimens.

4.3.2. Work Programme: See Ref. 1, Paragraph 4.2.2.

4.3.3. Progress and Results

During the last three months of the work period, various types of commercial resins were tested in order to assess the resistance to mechanical shock and the hardening time of the coat. The best results were obtained for a Blancomme type 03 FT1 resin (layer thickness: 400 μm) giving the shortest drying time (1 h at ambient temperature) and acceptable shock resistance by pendulum test.

4.3.4. Conclusions

This research work has shown that thermosetting resins (commercial liquid or powder epoxy and polyester) can be applied on metal components of various geometries in layers of up to 700 μm thickness (commercial standard: 30-100 μm); a particularly thick coating of 1.8 mm (three layers) was realized with a powder epoxy resin.

The effectiveness in immobilizing contamination was assessed on metal samples artificially contaminated with ^{137}Cs. Smear counts were found to be reduced by a factor of 10^4 after application of a two-layer coating of 55 μm total thickness.

The best application technique for these resins seems to be electrostatic projection with commercial spray guns. This technique seems to be well adapted to application in the nuclear field, including remotely controlled application.

4.4. 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

4.4.1. Objective and Scope

The aim of this research is 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.

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

4.4.3. Progress and Results

Work in 1981, aimed at removing cobalt from steel by a melting process of the Electro-Slag Refining type, had shown very low cobalt removal coefficients, due to the low solubility of cobalt phosphate in the slag. Since these results did not fit with the aim of the research, an alternative process, called 'controlled oxidation of steel', has been investigated in 1982. This process involves the introduction of an oxygenous gas flow into a steel/slag melt and was expected to result in:

- a metal phase amounting to about 1/10 of the stainless steel input and containing about 99% of the cobalt and the bulk of the nickel, to be disposed of as radioactive waste;
- a slag phase consisting of iron and chromium silicates, which could possibly be re-used for the production of stainless steel.

The new process was first studied by calculations and tests on 1 kg sample scale, in order to get reliable data on chemical exchanges taking place in the processing of stainless steel (AISI 304). Preliminary results showed that 1 to 10% of the cobalt present in the steel feed were transferred to the slag.

These results were then checked with respect to changes of slag composition which could be needed to meet fluidity or refractories requirements. These tests, carried out on a 20 kg sample scale, resulted in transferral to the slag of 8% of the cobalt and 2% of the nickel.

The study also showed that, for stainless steel, a two-step process is required: dechroming of the steel, and oxidizing of the iron. The nature of the various slags used in the two steps as well as the operating conditions and limits of the process were determined.

The favourable preliminary results (transferral of only 1% of the cobalt to the slag), obtained with 1 kg melts, could not be reproduced with 20 kg scale melts. Possible reasons for this are being analysed. It seems at present that the amount of cobalt transferred to the slag would be too large for re-use of the slag.

4.5. 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 - December 1983

4.5.1. 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.

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

4.5.3. Progress and Results

4.5.3.1. Literature survey

A literature survey was carried out to establish the known uses of sodium silicate solutions as a binder in the field of pelletizing, briquetting and the consolidation of powdered materials. Mechanisms of setting were investigated, including the use of chemical agents which render the binder insoluble. Consideration was given to existing methods of dust control with a view to incorporating known separation techniques into dust handling processes.

4.5.3.2. Material properties affecting dust treatment

Hardened masses of treated dust were formed from mixtures in slurry form of concrete dust and sodium silicate solutions. This was achieved by simple casting techniques and no further chemical or physical treatments were necessary to cause hardening of the slurry into solid blocks.

The efficiency of the binding of concrete dust by sodium silicate solutions was found to depend on many factors associated with the binder, concrete dust and the type of mixture. The effects of these factors on the binding of concrete dust were investigated in terms of setting times of mixtures and of tensile and compressive strength of hardened material.

Affect of sodium silicate solutions: Tests were carried out using sodium silicate solutions with silica/alkali ratios of 2.00, 2.85, 3.30 and 3.65, with solids contents in the range of 20-40%. Setting times were found to depend on the silica/alkali ratio and, for solutions with equal ratio, they depend on the solids content. The strength of binding was sensitive to both the silica/alkali ratio and solids content of the silicate solutions. Below 25%, solids strength was low, whereas above 35%, loss of strength due to changing mix characteristics is observed. Strong binding was achieved with a silica/alkali ratio of 2.85 and solids content of 33%.

Affect of concrete dust: The behaviour of concrete dust and sodium silicate solutions were dependent on the chemical nature of the dust. Tests were carried out using concrete dust from three different sources. These were 40 days, 5 years and 15 years old concrete. Analysis of the three concrete dusts gave the respective pH values as 12.2, 12.4 and 8.7. Mixing characteristics and setting times for the mixtures were affected by the alkalinity of the dusts. Dust with a pH of 12.4 gave shorter setting times than dust with a pH of 8.7. Greater quantities of sodium silicate solution were required to treat the more alkaline dust. These differences in behaviour are attributed to the concrete being manufactured from different cement types. Hydration of Ordinary Portland Cement releases free lime giving the concrete a pH of a saturated lime solution (12.4). Pozzolanic cement, although releasing free lime in the early stages of hydration, gradually causes a depletion of this lime with age, resulting in a concrete of lower pH. Differences in behaviour are associated with the lime content of the concrete.

Affect of mixture characteristics: The rate of setting of mixtures of concrete dust and sodium silicate solutions was found to be sensitive to the relative proportions of dust to solution. This sensitivity is related to the chemical nature of the setting mechanism.

4.5.3.3. Process requirements

The development of the engineering features for the process has been considered and alternatives were identified. From this study, refined development priorities have evolved for the assessment of the suitability of equipment for transport, mixing and pelletizing the dust/silicate mixture. The need for developing a pelletizing process has been recognized and techniques are being investigated. The most critical part of the development was identified as the mixing of the dust and silicate, to produce complete wetting of the dust by the silicate, and the interaction with the pelletizing processes for each of several stages of dust separation, in order to control particle size at each mixer and pelletizer.

4.6. 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 - December 1983

4.6.1. 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.

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

4.6.3. Progress and Results

4.6.3.1. Choice of the coating product

An epoxy paint modified with polyolefines (M.M.D. Titania E.P. series 5.23) was chosen for this investigation. With this product, which is similar to most paints generally used in nuclear industry, there is a large experience on maintenance cycles of previously painted structures. It can be applied, with good durability results, on both steel and concrete, even on surfaces which have not been perfectly prepared. The addition of polyolefines gives it plasticity and improves its adhesion, and it contains a non-toxic anticorrosive pigment (zinc phosphate). Its cost is moderate (about 7,000 Lit/kg).

4.6.3.2. Tests for conventional characterization

To characterize the chosen product, the following tests were carried out:

- mechanical resistance tests: abrasion, impact, drawing, bending, hardness, adhesion;
- chemical resistance tests: in salt spray (fog), in moisture, in water, in SO_2 atmosphere.

The samples for these tests were: sandblasted steel, rusted steel, steel with detaching rust, steel painted with spray cycle (aged), steel painted with polyurethan cycle (aged).

Good results were obtained in all chemical resistance tests, particularly the moisture and fog tests, which were continued for a second cycle. In the mechanical tests, the best results were obtained with specimens coated with two layers of paint (total thickness: 160-180 μm).

Some standard tests were carried out on concrete samples, dipped in water, in acidulated water and in alkalized water.

4.6.3.3. Non-conventional tests

The test samples were small disks or boards made of sandblasted steel, rusted steel, eternit (asbestos cement).

Smear tests (standard test for checking the transferable contamination) were performed on samples contaminated with a solution of uranyl nitrate and coated with one, two and three layers of paint.

Leach tests were carried out on contaminated samples coated with one, two or three layers of paint. Contamination release was determined by analysis of uranium in water. So far, these tests have not evidenced contamination release.

Irradiation resistance tests according to standard UNI 7804, on samples painted with two layers, were started at the ENEA Nuclear Center at Rome.

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. Reference nuclear power plants

Virtually all the major nuclear power plants in the European Community belong to one of three generic types - Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR) and Gas-cooled Reactor. Studies have therefore centred on reference plants representative of these three generic types, i.e.:

- Type 1: European PWR of the 3,800 MWth/1,300 MWe class;
- Type 2: European BWR of the 2,800 MWth/950 MWe class;
- Type 3: UK Magnox reactor of the 840 MWth/205 MWe class;
- Type 4: UK Advanced Gas-cooled Reactor, 1,500 MWth/620 MWe class.

5.1.3.2. Inventory of neutron-activated radionuclides

The radioactive inventories produced by neutron irradiation for the reference plants were estimated for decay periods up to 100 years using computer codes such as AKAT, REDIFFUSION and MCNID aided by axial and radial power profile measurements. A global summary of these data is given in Table 10.

Table 10. Activation inventories of reference plants

Reference plant	Total mass of neutron activated components (Mg)	Total induced activity (Bq) Time after shutdown		Main source of induced activity (10 y after shutdown)		
		10 y	100 y	Material	Mass (Mg)	Activity (Bq)
Type 1	1,040	$3.1x10^{16}$	$5.2x10^{15}$	X10CrNiNb189	18	$2.3x10^{16}$
Type 2	1,010	$2.4x10^{17}$	$2.8x10^{16}$	X5CrNi189	235	$2.4x10^{17}$
Type 3	20,070	$3.7x10^{15}$	$1.8x10^{14}$	mild steel	2,500	$2.6x10^{15}$
Type 4	51,000	$1.4x10^{16}$	$2.6x10^{14}$	mild steel	6,600	$1.2x10^{16}$

5.1.3.3. Inventory of radionuclide contamination

Potential contamination mechanisms vary widely between and within reactor types because they depend on the operating history as well as on accidental releases from the fuel. The available data were examined and best estimates derived. The following contamination inventories were estimated for the four reference plants:

- Type 1: $1.1x10^{13}$ - $1.1x10^{14}$ Bq;
- Type 2: $3x10^{14}$ Bq;
- Type 3: $6.7x10^{11}$ Bq;
- Type 4: $2x10^{12}$ - $4x10^{12}$ Bq.

Information on contamination of 'non-activated' components is less comprehensive but also less important for gas-cooled reactors. The predominant isotopes dictating package shielding are indicated as ^{60}Co,

^{134}Cs, ^{137}Cs. Whilst activities at both 10 years decay and 100 years decay were estimated, it was decided to use only the '10-year decay' figures for subsequent parts of the study, even though many reactor components may in fact remain in place for periods well in excess of 10 years, because of subsequent changes in both technology and regulations likely over that period.

5.1.3.4. Survey of existing containers

The technical data for each model of suitable container found after collecting information in Belgium, France, Germany, Italy, the Netherlands, the UK and the USA have been compiled. The packagings were classified in appropriate groups. Criteria were defined for evaluating the existing designs, for non-Type B and Type B packages. Whilst a number of the designs satisfied the selection criterion for specific waste streams, they were generally not suitable for a wide range of wastes, or optimized for payload. Useful design features were noted however.

5.1.3.5. Means of transportation

Information was compiled on the main means of transportation and of the size and weight restraints imposed by transport regulations. For the transport of heavy loads from nuclear power stations, rail would appear to be the more practicable mode of transport, though some handling by road to or from the rail-head may be necessary. A view has yet to be formed on the design for packaging to be transported by water or sea.

5.1.3.6. Regulations on transport and disposal

The regulations in the European Community covering transport of radioactive materials and sea disposal were compiled and assessed. Data on land disposal regulations are very limited, due to the lack of evolved policy in a number of countries. The IAEA regulations for the transport of radioactive material (Safety Series 6) remain dominant on package design, particularly for solidified intermediate level wastes.

5.1.3.7. Definition of possible conceptual designs

Conceptual designs of large containers were initiated. General criteria were evolved covering size, shape, handling, fixings and fittings as well as materials of construction. These criteria were elaborated further to cover transport under the IAEA Transport Regulations (Safety Series 6) under Type B or non-Type B categories.

This study has led to the definition of the following types of transport container for further elaboration, all as large as possible within the limits set by the railway freight gauge:

- a cylindrical or square-section Type B container;
- a rectangular self-shielded container of modular design to take large pieces of Low Level Solids (LLS) waste;
- a cylindrical container to transport self-shielded inner containers as LLS;
- a cylindrical shielded container to transport LLS.

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 had been limited in a first phase 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.

In 1982, a study on the methodology for evaluating radiological consequences of the management of low-level waste and a review of techniques for measuring very low-level radioactivity have been added.

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 is 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

6.1.3.1. Taking of samples

Three concrete cores were taken from the biological shield by horizontal drilling from the outside at three different levels, i.e., reactor core bottom level, reactor core mid plane and reactor core top level. A number of samples were taken from the concrete cores at distances of up to 120 cm from the inner shield surface.

6.1.3.2. Measurements for gamma-emitting isotopes

Gamma emitters were measured on all samples. Figure 11 gives the specific activity in concrete as a function of depth for the reactor core mid plane. Decay time after final shutdown of the reactor was about five years at the moment of measurement.

In the steel reinforcement bars, gamma activity was about 0.2 µCi/g for ^{60}Co at 13 cm distance from the inner shield surface. ^{54}Mn and ^{137}Cs activities were much lower.

6.1.3.3. Chemical analyses of concrete samples

Analyses were performed by ICP (inductively coupled plasma) emission spectrometry and by atomic absorption (see Table 11). Trace elements will be determined more precisely by activation analysis.

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 is linked to the work carried out by Kernkraftwerk Lingen GmbH (see Paragraph 6.1) and has the same objective and work programme.

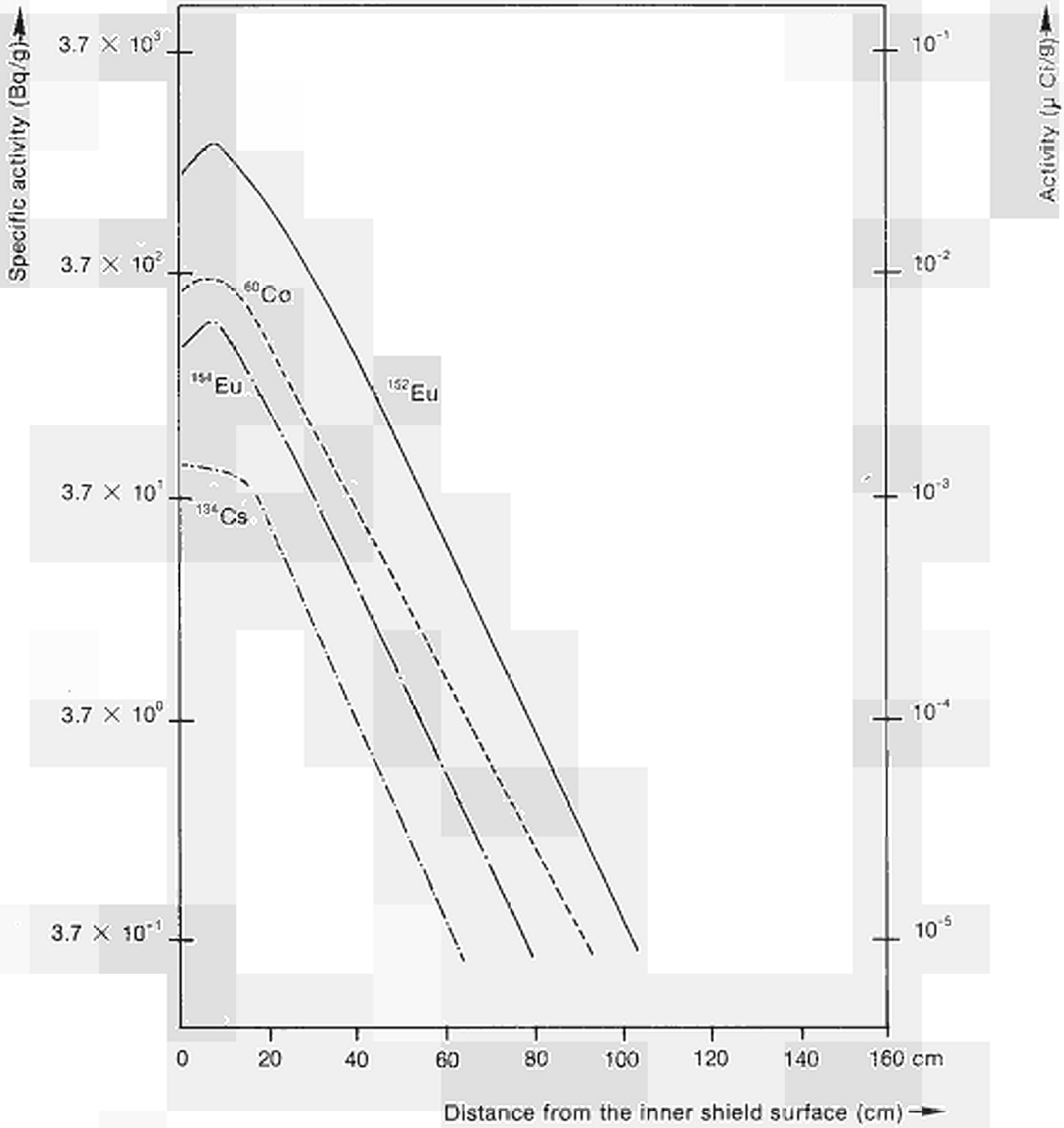

Figure 11. Gamma activity of the KWL biological shield concrete (at reactor core mid plane)

Table 11. Chemical analyses of concrete from the biological shield of KWL

Element	Content (mg/g)	Element	Content (mg/kg)
Si	335 $\pm$ 20	Ba	240 $\pm$ 20
Ca	55 $\pm$ 10	Ni	13 $\pm$ 5
Al	12 $\pm$ 3	Sm	5.6 $\pm$ 1.0
Fe	6 $\pm$ 2	Co	4 $\pm$ 2
K	5 $\pm$ 2	Cu	0.93 $\pm$ 0.20
Mg	4 $\pm$ 1.5	Cs	0.9 $\pm$ 0.1
Na	2 $\pm$ 1	Ir	<0.003
Cl	1 $\pm$ 1		
Mn	0.9 $\pm$ 0.1		

6.2.1. Progress and Results*

6.2.1.1. Taking of samples

At three levels (reactor core top, midplane and core bottom), samples were taken up to a distance of 2 m from the inner shield surface.

6.2.1.2. Activity measurements

The measurements for gamma emitters on the concrete samples show a maximum of 1.2×10^{-7} Ci/g for ^{152}Eu at 0.5 to 5 cm distance from the inner shield surface. Figure 12 shows the specific activity of the main gamma emitters at core mid plane. One sample was analysed for long-lived beta emitters. The specific activity of ^{45}Ca was measured to be 3.8×10^{-9} Ci/g.

The activity of steel reinforcement bars is dominated by ^{60}Co with a maximum of 2×10^{-6} Ci/g. ^{152}Eu was found in one sample.

6.2.1.3. Chemical analysis of concrete samples

The results of the chemical analysis are given in Table 12. The weight loss after heating to 600°C was 8.9% for Sample 1 and 7.9% for Sample 2. Trace elements will be analysed again by activation analysis.

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* Information on this research is also published in Ref. 7.

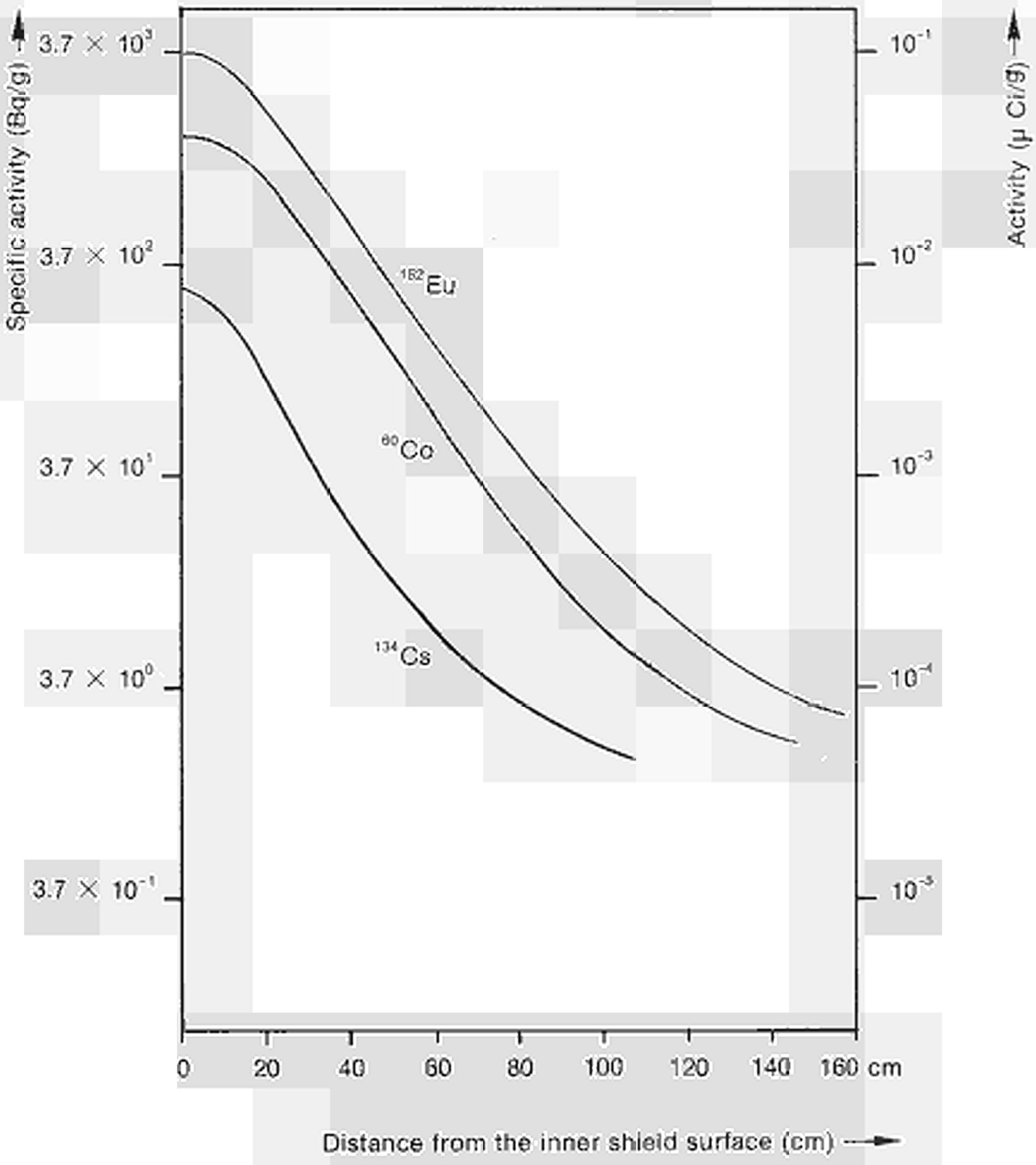


Figure 12. Gamma activity of the KRB-A biological shield concrete
(at reactor core mid plane)

Table 12. Chemical analysis of concrete from the biological shield of KRB

Element	Content (mg/g) Sample 1	Content (mg/g) Sample 2	Element	Content (mg/kg) Sample 1	Content (mg/kg) Sample 2
O	455	441	Ba	990	2540
Si	233	229	Cu	330	310
Ca	202	210	Co	130	130
C	26	27	Ni	110	320
Al	22	17	Cs	63.9	60.1
Fe	11	14	Ho	45.6	36.7
Mg	11	9	N	<3	<3
H	10	9	Eu	<0.7	<0.7
K	5	3			
Na	3	3			
Sm	<1.3	<1.3			
S	1	1.5			
Gd	<1	<1			

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 are:

- 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

6.3.3.1. Activation calculation

As a preliminary step, in order to evaluate the neutron flux and activation distribution, a three-dimensional calculation was performed of power and water densities and fissile nuclides concentrations at the end of the 5th cycle, which virtually represents the end of the reactor life.

On the basis of the results, an ANISN computer-code evaluation will be performed with the purpose of providing cross sections reduced from a hundred to 27 groups in order to decrease computer time for the further evaluation. This DOT code calculation will be performed to estimate neutron fluxes at the vessel. The results of this further calculation, after comparison with the effective power distribution related to the reactor operational life, will be used as input data for an ad-hoc computer code, to evaluate vessel materials activation.

6.3.3.2. Analysis of samples

Vessel steel composition data were lacking as concerns the impurities to be considered in activation calculations, i.e. Co, Nb, Ag, Mo. Therefore, it was decided to analyse steel samples obtained during manufacturing. Since atomic absorption analyses gave significant figures only for Co and Ag, complementary analyses were performed by neutron activation followed by high-resolution gamma spectrometry. The samples were activated in the TRIGA MARK II reactor of Pavia University and a Ge(Li) detector coupled to an ORTEC multichannel analyser was used for measurements. Results are shown in Table 13.

The work programme includes also measurements of gamma activity on irradiated samples. These samples, mounted close to the reactor core at the beginning of operation, were periodically taken out in the frame of the in-service inspection programme. Unfortunately, the decision to decommission the plant has stopped the in-service inspection programme. Therefore, the date for these experiments is uncertain.

Table 13. Contents (ppm) of impurity elements in the vessel steel of the Garigliano reactor

Element	Sample 1	Sample 2
Co	175 ± 9	166 ± 8
Ag	1.10 ± 0.11	1.17 ± 0.12
Mo	5800 ± 406	7200 ± 400
Nb	238 ± 47	201 ± 32

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 current research is to carry out trace element assessment of low-alloy and stainless steels with particular reference to sources of long-lived gamma emitters. Both neutron activation analysis and atomic absorption will be used with the aim of characterizing the variability of Nb, Mo, Co, Eu, Sm, Ho and Ag concentrations. Using established data libraries, long-lived activities in representative reactor fluxes and spectra will be assessed for typical sample compositions.

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

6.4.3. Progress and Results

6.4.3.1. Neutron activation analysis

Most of the work during 1982 was aimed at the development of a technique for the determination of niobium. Theoretical studies have shown that niobium often determines the long-term gamma activity in irradiated steel beyond about 90 years cooling. It is a strong gamma emitter and may dictate special storage and disposal provisions for some LWR components.

The activation analysis of niobium is being carried out using the $^{93}Nb(n,\gamma)^{94m}Nb$ reaction. ^{94m}Nb has a half-life of 6.3 min, but, unfortunately for this work, decays mainly by a highly internally converted isomeric transition to the ground state of ^{94}Nb. Only 0.5% of ^{94m}Nb decays go by beta decay to an excited state of ^{94}Mo with a resulting 871 keV gamma ray. To detect ^{94m}Nb, there is thus a choice between measuring the relatively intense K X-rays at 16.6 keV and 18.6 keV or the very weak gamma rays at 871 keV. Because of the low energies of the X-rays, there are severe self-absorption problems encountered in their measurement. On the other hand, the low intensity of the gamma rays leads to a very poor analytical sensitivity for niobium because of the presence of other strong gamma emitters.

In an attempt to overcome these problems, a chemical separation process for niobium was devised. The process was based on the extremely low solubility of niobium in concentrated hydrochloric acid. 1 or 2 g samples of steel were boiled in 75 ml of concentrated hydrochloric acid for 30-60 min. Most of the steel goes into solution, leaving a black residue containing the insoluble niobium. The residue was removed by filtration and then irradiated in a mainly epithermal neutron flux. This process leads to lower detection limits for both the K X-rays and the 871 keV gamma rays. It has since, however, been pointed out that the niobium content of steel is shared between that present 'in solution' and that contained within precipitates, mainly as niobium carbide. It appears that the dissolution technique described above releases only the precipitated portion, with the rest being dissolved along with the rest of the steel. It has also been shown that the ratio of niobium collected to that lost depends crucially on the thermal history of the steel and on the carbon to niobium ratio in the steel. It is concluded that this separation technique is not reliable and attention is now turning to other analytical techniques for the determination of niobium.

Preliminary investigations were carried out into the potential of neutron activation analysis for the elements Mo, Ag and Sm using the short-lived isotopes ^{101}Mo (14.6 min), ^{108}Ag (2.4 min), ^{110}Ag (24.4 s) and ^{155}Sm (22.4 min).

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. Concrete samples from biological shields of about ten reactors in the European Community will be analysed.

6.5.2. Work programme

The elements to be determined in concrete are chlorine, calcium, nickel, cobalt, niobium, samarium and europium. All these elements produce long-lived radionuclides when they are irradiated in concretes used for reactor shielding. Neutron activation followed by gamma spectrometry will be used for the quantitative determination of these elements in order to subsequently calculate the activity due to long-lived radionuclides.

Calcium, cobalt, europium and samarium will be directly identified by gamma spectrometry after irradiation. The technique used for these elements will be easily extended to concrete samples from various sources.

For chlorine and nickel, which cannot be determined directly, chemical separation procedures were elaborated. For chlorine, the irradiated concrete sample is solubilized by alcaline melting. The residue is dissolved in water, then hydrochloric acid is added as carrier. The addition of a solution of silver nitrate gives a silver chloride precipitate which contains the ^{38}Cl. In using this procedure the interfering elements are eliminated. For nickel determination, the complex beta nitroso - alpha napthol - cobalt-58 ($^{58}Ni(n,p)^{58}Co$) is extracted by carbon tetrachloride after concrete alcaline melting.

6.5.3. Progress and Results

A preliminary study was performed on ordinary concrete to check the procedures and to determine the best analytical method. In the chosen method the concrete samples are pulverized in a tungsten carbide mortar.

As a first application, four concrete samples from biological shields (three from the Lingen reactor and one from KRB-A Gundremmingen) were analysed. Ca, Co, Sm, Eu, Ba, Cs and Ni contents were determined and a series of other elements (Lu, Nd, La, Fe, Ta, Sb, Yb, Sc, Th, Rb, Cr, Ce and Hf) were identified.

The most difficult task is the analysis for niobium. The most obvious reaction, i.e. $^{93}Nb(n,\gamma)^{94m}Nb$ with a 6.3 min half-life, cannot be used due to the existance of more powerful gamma emitters covering its characteristic gamma rays. For the development of a convenient method, the reaction $^{93}Nb(n,2n)^{92}Nb$ with a half-life of 10.2 days (gamma ray 934 keV) was chosen. For this reaction, an interference exists due to molybdenum by $^{92}Mo(n,p)^{92}Nb$. Therefore, molybdenum is measured independently and the interference is substracted.

The method applied was to dissolve the irradiated concrete sample by alcaline fusion with Na_2O_2, followed by adding water and nitric acid. After addition of iron carrier, iron hydroxide containing the niobium was precipitated. After elimination of chlorine and silicon, the counting takes place. For the reactions chosen, 1 ppm of Mo gives an apparent Nb content of 2.16 ppm. First measurements on concrete showed a Mo content in the order of 2 ppm and an apparent Nb content of 12 ppm. This would mean that the real Nb content is about 8 ppm. In case of a too important Mo interference in future samples, a method for separating niobium prior to irradiation has to be developed.

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

Contractor: National Radiological Protection Board, Chilton, United Kingdom

Contract N°: DE-F-006-UK Work Period: September 1982 - December 1983

and

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

Contract N°: DE-F-007-F Work Period: September 1982 - December 1983

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

6.6.2.1. Definition of the work

Starting from existing models, the methodology should be adapted to the base cases of light-water cooled or gas cooled nuclear power plants finally shut down. The decay time before dismantling the plant will be a parameter for this study and 5, 25 and 100 years should be considered. All strongly activated materials and all liquid waste have been disposed of and surfaces have been decontaminated to reasonably low residual contaminations. Consequently, the materials to be taken into account will have activity levels rather near to limits for unrestricted release. For the purpose of this study, the following values will be considered as limits for unrestricted release: 0.37, 3.7 and 37 Bq/g of specific activity.

The methodology should be clearly presented and numerical examples for typical plants, sites, decay times, waste treatment and management modes will be calculated, giving radiological consequences and cost. The areas in which experimental research is needed to eliminate important uncertainties of input data, will be identified. Additionally, easy-to-use relations and graphs will be developed, allowing the computation of upper bounds of doses and costs.

The methodology will cover exposure of workers during treatment, transport and disposal operations, and exposure of the public during these operations and after recycling and disposal. The probabilities of exposure for the various pathsways will be given. The methodology should allow for calculating individual and collective effective dose equivalents, and dose equivalents to most the important body organs; it should cover all radiologically relevant exposure pathways.

Collective dose commitments will be calculated over various time periods (infinity and truncated for instance at 50 y, 500 y, 10,000 y). The sensitivity of the dose results to a variation of the most important parameters used in the models will be analysed. A cost-benefit analysis of the recycling/disposal methods envisaged will be performed.

6.6.2.2. Areas to be studied

The areas to be studied and the mainly responsible organizations are as follows.

Definition of waste (CEA): The quantities of waste and the radionuclide inventory for the above-mentioned generic plants and conditions will be established on the basis of available data. The main materials will be concrete and steel, but the inventory has to show, if it is necessary to deal specifically with other materials, e.g. copper and lead. Concrete with non-separable iron re-enforcement bars is a category to be distinguished. Graphite, having disposal routes of its own, will be excluded.

Recycling (CEA): Two options will be considered for recycling, i.e.:

- after treatment of the material the further destination is unknown and uncontrolled;
- the destination of the recycled material is known (reuse in nuclear installations or as containers for radioactive waste, for civil works under water, etc.).

Discharge to refuse tips (NRPB): The refuse tip will be a normal municipal refuse tip without any further control after the end of the disposal operations. Two generic sites differing in hydrology and soil conditions will be considered.

Burial at the reactor site (NRPB): Two or three generic sites will be evaluated. The extreme cases will be a site on a large river, like the Rhine or Rhone, on the one hand, and a site on the sea-shore, on the other hand.

Disposal in the sea (NRPB): Disposal in the sea, either near the shore or to the deep sea, will also be considered since these are possible alternatives to burial.

6.6.3. Progress and Results

The main work was to determine the radionuclide content and quantities of waste and to define the disposal options in detail.

6.6.3.1 Disposal at municipal refuse tips

It will be assumed that the wastes are placed directly in simple trenches, which incorporate minimum engineering. Several studies have been published which attempt to model underground water flow at landfill sites and these are currently being reviewed in order to define two generic sites which differ in hydrology and soil conditions. Coefficients defining the release of radionuclides from the wastes to the percolating underground water were estimated from data obtained for the release of metals from landfill sites.

6.6.3.2. Disposal at reactor sites

The range of possible burial facility designs has been reviewed in order to define a reference structure. It is expected that the depth of the trenches will be 10-20 m and that some engineered features will be incorporated, for example a concrete lining and concrete/clay cap. Consideration is also being given to the most suitable from of waste packaging.

Two reference generic sites will be considered. The first of these is a coastal site and is shown in Figure 13. It will be assumed that underground water flow is entirely within the gravel aquifer and is towards the coast. The second site is situated on a large river but further information is needed before the exact details of the site are defined.

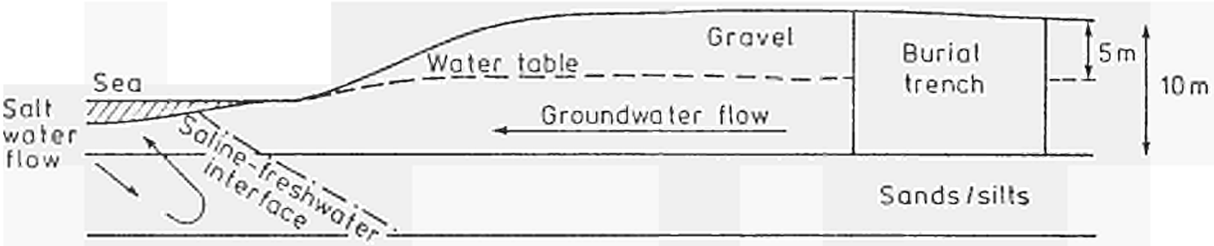


Figure 13. Generic coastal reactor site

6.6.3. Disposal into the sea

New models are being developed at NRPB for the leaching and dispersion of radionuclides from wastes which have been disposed of in the deep ocean and will be used to assess the radiological consequences of this disposal option. An existing model for the dispersion of radionuclides in the coastal seas of Northern Europe will be used in the radiological assessment of disposal close to the shore.

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 - January 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

The techniques and procedures of measurement to be considered relate to the dismantling of nuclear power plants shut down after a working period corresponding to design life; before dismantling, a decay time of 5 to 100 years is observed; all strongly activated materials and liquid waste have been removed and surfaces, where necessary, have been decontaminated with usual procedures. Consequently, only low-level activities are present.

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

General aspects (GKSS and CEA): General aspects of measuring and controlling low-level radioactivity will be discussed, e.g. definition of radioactive waste by licensing authorities, necessity of homologation, dependence on possibilities of waste disposal.

Existing experience (GKSS and CEA): The existing experience will be assembled and documented. The description will include hardware, methods and strategies, main results (including information on necessity of further storage or release of the measured materials), cost and disadvantages of the methods. Special attention will be paid to sweating of activity after decontamination.

Available hardware (GKSS): Available hardware, excluding sophisticated laboratory equipment, will be reviewed. Means to measure soil samples will be included, but air measurements will not be considered. Main characteristics, like precision, limitations and costs, will be given. Homologation by licensing authorities and compliance with codes and standards will be indicated.

Handling and application (GKSS): A description of handling modes and applications, including difficulties, limitations and time for execution, will be given.

Procedures (CEA): The procedures are related essentially to strategies for measurement and control of slightly radioactive material. Probabilistic methods and statistics will be applied. If the history of the items to be measured is known, the arrangement of measuring points could be adapted consequently. The possibility of sweating of radioactivity will be considered. Costs will be established as a function of the precision sought. The procedure to measure a typical isotope mixture in the laboratory and to limit field measurements to one or several lead isotopes, will be investigated.

Possible improvements (CEA): Possible and necessary improvements for hardware and methods will be shown and areas for future research will be suggested.

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 order to assess the influence of nuclear power plant design features on decommissioning, a catalogue of possible design features facilitating decommissioning of Advanced Gas-cooled Reactors (AGRs) will be established. On the basis of existing studies, this catalogue will show the advantages and disadvantages of each design feature, and the particular effects on cost and radiation exposure, taking into account the construction, operation and decommissioning of the plant.

The aim of this research is to consolidate data for the use of reactor designers:

- to set up methods of minimizing the end-of-life active inventory;
- to establish if equipment (or access and support arrangements) should be installed during the construction phase, subject also to operational restrictions;
- to check the feasibility of using removable components including concrete biological shielding.

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 overall programme for decommissioning includes delayed dismantling and the capability of removing activated solid waste from the debris storage vault either during or at the end of station life.

The major effort during 1982 has been the assessment of the expected neutron-induced radioactive inventory corresponding to ten years after shutdown and 100 years after shutdown. To calculate these inventories, it was necessary to establish the elemental composition of all the materials in the zones affected by the neutron fluence. For convenience, the reactor components within the prestressed concrete pressure vessel and including the concrete pressure vessel, have been classified into zones. Outside these zones, neutron fluxes are low and induced activity levels are insignificant. The work of identifying all the components, the material, the material specification and the operating conditions is in an advanced stage, though still subject to the uncertainties in elemental composition. Large variations in some impurity levels are evident from the results of chemical analyses of graphite of different batches and reported variations in manufacturers' specifications for other materials. This means at times, it will be necessary to assume pessimistic (i.e. high) values for impurities. The remainder of the work of establishing the 10-year and 100-year after end of life inventories is still proceeding. This work will illustrate the design significance of minimizing elements which produce long-lived active species.

The prestressed concrete pressure vessel inner surface becomes mildly active during the station life and work is proceeding to establish the activity levels of this very large mass with a view to providing information which will assist in establishing waste disposal routes.

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 - December 1983

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

The research can be considered in four main parts, namely:

a) A literature review and study to define and understand the main problems posed by decommissioning.

b) The generation and development of ideas and features that will overcome the problems.

c) An assessment of these ideas with respect to design, construction, operation and cost of the installation.

d) A statement of those features and ideas which appear commercially and technically feasible.

At the end of 1982, item a above is complete and work on items b and c is well advanced.

7.2.3.1. Literature review and study

The literature review identified 110 references of which ten were selected for detailed study. The main conclusion of the review is that, generally, all the references cover very similar ground with little information on actual decommissioning experience. However, the literature did point to several features likely to aid the decommissioning of future plants.

Many of the problems associated with the dismantling and removal of large activated structures of composite concrete and steel construction arise from the time limitation on personnel access to activated zones in order that such personnel do not receive a radiation dose exceeding the acceptable limit. The main problems are as follows:

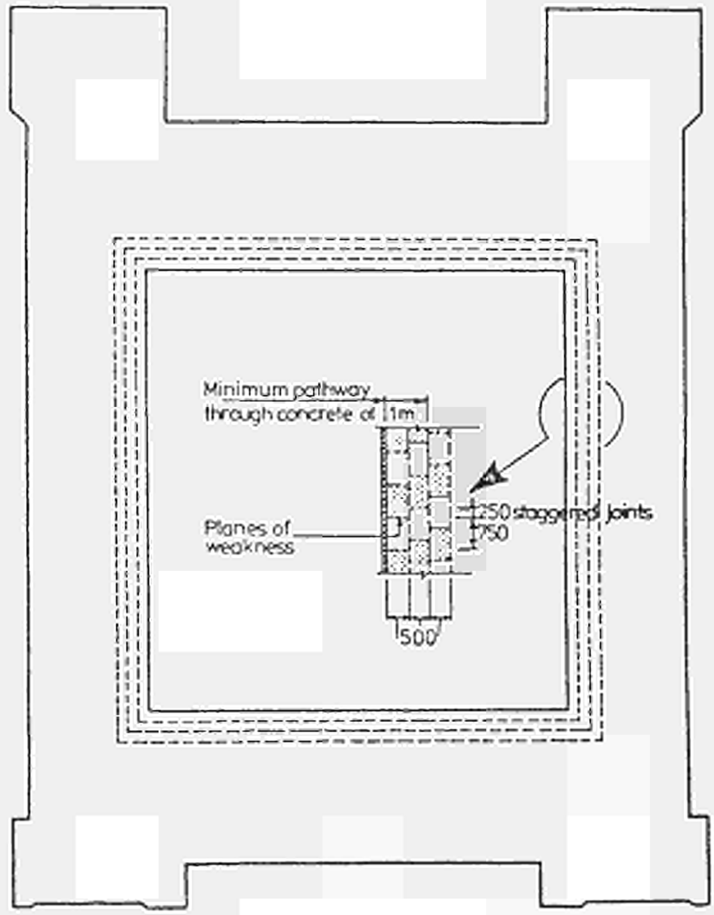
- Control of smoke, fumes and airborne dust associated with conventional explosive and cutting techniques.
- Separation of bonded reinforcement and bonded prestressing tendons from fragmented concrete. The presence of cobalt-60 in conventional steels presents a particular radiological problem.
- Separation of steel liners and penetrations from concrete. Again the radioactive isotopes present in the steel give rise to radiological problems.
- Controlled destressing and removal of certain types of prestressing tendons.
- Safe packaging and removal from site of activated steel and concrete debris, fluids, residual dust, etc.

The literature study and discussions with others indicate that the activated concrete occupies a relatively thin layer surrounding the main liner and, for the purpose of the research, this layer is deemed to be 1 m thick.

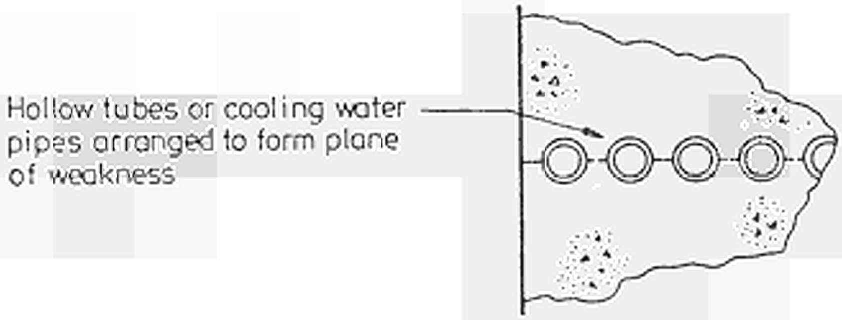
7.2.3.2. Proposed design features and ideas

The possible solutions that have emerged to overcome the above problems fall into the following four categories:

- Planes of weakness: The ideas within this category envisage built-in planes of weakness between the activated and non-activated concrete, and within the activated concrete itself so that this can be broken up and removed with relative ease. Figure 14 illustrates this idea.
- Features to facilitate forceful break-up: These ideas imply building in to the activated concrete, mechanical, chemical or other devices which can be triggered to cause disintegration of the concrete. One means of providing for such devices is shown in Figure 15 (a).
- Selective use and location of materials: These ideas are based on keeping the activated concrete as free as possible of materials which are difficult to remove, either from radiological considerations or because of their location and geometry.

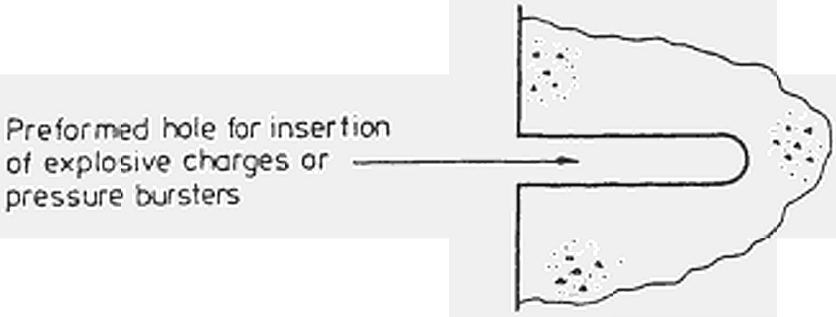


IN-BUILT PLANES OF WEAKNESS
Prestressed Concrete Pressure Vessel

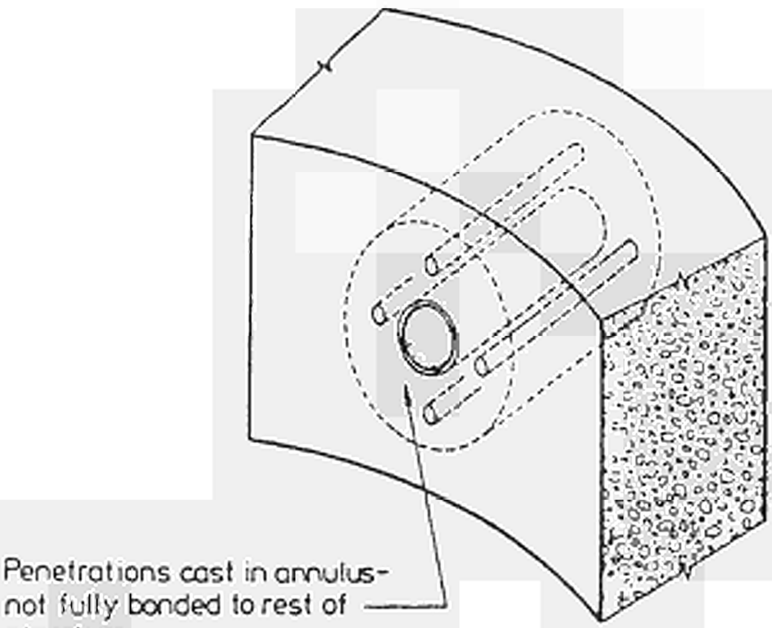


PLANES OF HOLLOW TUBES

Figures 14. Proposed design features for concrete structures



PREFORMED CAVITIES
(a)



(b)

Figures 15. Proposed design features for concrete structures

- Removal of liner and penetrations: Possible solutions here depend on redesigning the liner and penetration details to make these items more easily detachable from the concrete. A typical arrangement for a group of penetrations is shown in Figure 15 (b).

It is likely that application of some of the above solutions will require the development of remotely controlled equipment. Within the above four categories, approximately 30 separate ideas have come forward and are currently under consideration.

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: February - December 1982

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 substition of cobalt alloys by other materials would reduce the overall radiation level of primary systems of nuclear power plants to be decommissioned.

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

The objectives of this research are:

- to demonstrate the erosion-corrosion resistance of substitutes in experiments;
- to establish a comparative list, taking the most current cobalt alloy as a reference;
- to determine possible correlations of the erosion-corrosion behaviour with material characteristics.

7.3.2. Work Programme: See Ref. 2, Paragraph 7.3.2.

7.3.3. Progress and Results

7.3.3.1. Test materials and conditions

The test materials (Table 14) include four welded surface layers, one detonation gun coating and two structure materials. The reference material is Stellite 6 (welded layer).

The tests were carried out at the following conditions:

- temperature: 150°C;
- medium: demineralized water with 750 ppm boron as boric acid and usually less than 0.01 ppm O_2;
- flow velocities: 65, 44 and 2 m/s;
- duration of test runs: 300 + 400 h.

7.3.3.2. Test results

Weighting of the specimens before and after the tests showed that weight changes were smaller than the inaccuracy of about 1 mg, due to small damages during mounting and dismounting of the specimens.

Scanning electron microscope examinations were carried out before testing and on specimens having been exposed to the highest flow velocity (65 m/s). The examinations showed that five materials (Colmonoy 4, 17-4 PH, LC-1C, Tribaloy T 700, Nitronic 60) had been slightly smoothed. Moreover, small selective attack was found on the Colmonoy specimen. The specimens of the other three materials (Stellite 6, Pantanax 25, Antinit Dur 300) were not affected by the flowing medium.

The metallographic sections confirmed that there was no or very small selective attack. The maximum attack was about 6 µm and was found on the Colmonoy 4 specimen having been exposed to the highest flow velocity (65 m/s).

Surface roughness measurements, carried out before and after the tests, indicated smoothing of the 17-4 PH steel. The changes of roughness values of the other materials were smaller than the inaccuracy (scattering) of the measurement method.

7.3.3.5. Conclusions

All tested materials showed a high resistance against erosion-corrosion. The small differences found indicate the following order of decreasing resistance against erosion-corrosion:

Table 14. Composition and hardness of the test materials

Material designation	Composition (wt%)													Vickers hardness HV 20
	C	Si	Mn	Cr	Mo	Ni	N	W	V	Fe	Co	Cu	B	
Welded layers:														
Pantanax 25/0 Mo-G	2.5	0.4	0.9	25	3.1	-	-	-	0.4	bal.	-	-	-	470-480
Tribaloy T 700	-	3	-	15	32	bal.	-	-	-	-	-	-	-	450-470
Alloy E 3	0.11	5	6.5	21	-	7.8	-	-	-	bal.	-	-	-	300-340
Colmonoy 4	0.45	2.25	-	10	-	bal.	-	-	-	2.5	-	-	2.0	450
Detonation gun coating:														
LC-1C	Cr_3C_2 + 20% NiCr 8020													
Structure materials:														
17-4 PH	0.006	0.36	0.46	17	-	3.9	-	-	-	bal.	-	3.6	-	350
Nitronic 60	<0.1	4	8	17	-	8	0.13	-	-	bal.	-	-	-	250-270
Reference material:														
Stellite 6	1.1	1.5	1	30	<1.5	3	-	4.5	-	3	bal.	-	-	430-460

- Pantanax 25
- Stellite 6 (reference material)
- Antinit DUR 300
- Nitronic 60
- Tribaloy T 700
- LC-1C
- 17-4 PH
- Colmonoy 4

The resistance against erosion-corrosion is not connected with the hardness of the material. The main influence seems to be corrosion resistance, i.e. chemical, not mechanical properties.

As a follow-up to this work, it is planned to start in 1983 tests using water with suspended magnetite particles.

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 by neutron activation of cobalt-60 in the thermal reactors. This strong gamma emitter will be the dominant source of radiation exposure during decommissioning in most cases. Niobium-94 will become an important source of gamma rays after decay periods of several decades.

A study is being made of the origin and possibilities to reduce the cobalt and niobium content of the materials which are used within the neutron field. The study includes the following items:

- a review of the current material specifications for cobalt and niobium impurities;
- evaluation of the possibility to reduce the cobalt and niobium impurity content by using low cobalt ores and feedstocks;
- evaluation of the possibilities to reduce the cobalt and niobium impurity content by special treatments;
- theoretical assessment of the radioactivity in steel having the present cobalt and niobium specification and a specification based

on the possible routes to low cobalt and niobium steels, considering thermal reactor and fast reactor fluxes.

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

7.4.3. Progress and Results

A literature review, supported by some additional calculations, was made in order to identify the potential sources of long-lived gamma radiation in reactor steel structural components. Besides cobalt, other residual elements in steel which may give rise to a gamma radiation hazard are niobium, silver and, of less significance, europium. The decay of ^{60}Co (half-life 5.27 years) dominates the gamma dose rate at reactor shutdown and for several decades afterwards, but eventually the longer-lived ^{94}Nb (half-life 20,000 years) and ^{108m}Ag (half-life 127 years) can become more important. The high burn-up cross-sections of gamma-emitting europium isotopes render this element unimportant except in components which experience a relatively low neutron flux, where ^{152}Eu (half-life 13.3 years) can contribute a significant proportion of the overall, rather low, gamma dose rate.

Chemical analyses of reactor grade AISI 304 stainless steel have been carried out to determine the typical residual levels of potential sources of gamma radiation which are achieved in present-day steelmaking. Four heats of the steel, with cobalt contents in the range 140-350 wt ppm, were analysed. The niobium content of the steels was generally <25 ppm but up to 85 ppm was present in one sample. The silver content was generally <0.1 ppm but in one case reached 0.8 ppm. The highest niobium and silver levels were recorded in two different heats of the steel, indicating that in practice either element could give rise to the highest gamma dose rate after, for instance, a 100 year decay period. The europium content of the analysed samples was in all cases $<4 \times 10^{-4}$ ppm, which is too low to present a significant radiation hazard, whatever the neutron flux.

The major sources of cobalt, niobium and silver in steels are scrap (low-alloy and stainless steels, nickel-based alloys etc) and other raw materials (e.g. ferrochrome) used in steelmaking. Since these three elements are not easily oxidized during steelmaking, they are retained in the finished steel. Lower residual concentrations can be achieved by using higher purity raw materials such as:

- Swedish sponge iron (Harwell analysis: 45 ppm Co, 6 ppm Nb, <0.04 ppm Ag, 0.27 ppm Eu);
- nickel granules (Harwell analysis: 7 ppm Co, <1 ppm Nb, 0.15 ppm Ag, $<4 \times 10^{-4}$ ppm Eu);
- chromium metal (manufacturer's analysis: 0.3 ppm Co, <0.01 ppm Nb, <0.02 ppm Ag).

According to the above analyses, an AISI 304 steel (72% Fe - 18% Cr - 10% Ni based) manufactured from these higher purity materials would be expected to contain about 35 ppm Co, 5 ppm Nb and <0.05 ppm Ag. At such low residual levels in the steel, the radiation dose rates arising from niobium and silver would be less than arises from ^{59}Ni. Although the sponge iron analysis revealed an appreciable europium content (0.27 ppm), such a level could not give rise to a high gamma dose rate and, in any case, europium is easily oxidized and could be refined out of the steel.

Based on raw materials costs, a 35 ppm Co AISI 304 steel would, at around £1000 per tonne, be about 2-3 times more expensive to produce in ingot form than a normal production heat. In a PWR, for example, stainless steel structural components total only about 60-70 tonnes. The additional expense of using the higher purity steel would therefore be small in absolute terms as well as relative to the costs of component manufacture and, otherwise, possibly delayed reactor dismantling.

7.5. Removable Coatings for the Protection of Concrete against Contamination

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

Contract N°: 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

7.5.3.1. Critical analysis of existing knowledge

As a first step, an investigation was made on the use of removable coatings in nuclear industry. In Italy, removable coatings have been applied so far seldom and only as a decontamination technique. More extensive application is reported from other countries, expecially France, U.K. and U.S.A.

A variety of removable coatings, mainly vinyl based, either organic solutions or water dispersions, were used to decontaminate carbon steel or concrete surfaces at TMI-2. Decontamination factors up to 50 are reported.

Application as protective coating is reported too. This makes it possible to decontaminate the component covered by removal of the coating.

7.5.3.2. Experiments

In order to set up a test programme, the following general requirements for a removable coating were considered:

- it should durably and efficiently protect against contamination;
- it should be easily removed in order to reduce the decontamination costs, either in terms of the occupational radiation dose or in terms of total waste arisings.

On this basis, a series of tests is being performed on selected coatings, in order to evaluate mechanical characteristics, chemical and radiation resistance and decontaminability. The first results show the interest of a multilayer coating consisting essentially of a normal decontaminable layer, protected by a strippable coating.

7.6. Characterization and Improvement of Coatings Protecting Concrete against Contamination

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

Contract N°: 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 decontamination by washing or by removing the coating. Normally, these coatings do not assure efficient long-term protection against the penetraction 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 decontaminate 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

7.6.3.1. Decontamination tests on strippable paints

Decontamination tests were carried out according to the French norm NF T 30 901 of November 1976. These tests determine both the susceptibility to radioactive contamination, required in French stations to remain below 20%, and the efficiency of decontamination, which must be above 85%.

Of the numerous tests performed, the majority proved entirely negative, since most strippable coatings possess a feature inconsistent with the requirements of the norm NF T 30 901: they fix and hold radioactive contamination.

7.6.3.2. Tests on composite systems

To offset this disadvantage, many trials were carried out on multilayer systems up to 0.4 mm thick, built up as follows:

- on the concrete, a strippable coating with all the requested properties but non-decontaminable;

- a finishing layer of the standard kind used in French commercial nuclear power stations, to render the system decontaminable.

The main difficulty here lies in the choice of components which are compatible to assure the homogeneity of the system as a whole. An interaction between the final layer and the undercoating can, for instance, deprive the latter of its strippability.

7.6.3.3. Products selected

Two coatings have so far passed all tests successfully. One of them (trade-mark Meyer 2258) belongs to the first category tested: strippable white paint based on vinyl acetochloride in methylene chloride solution laid on concrete in two identical 250 µm thick layers. Temperature resistance, artificial weathering, chemical corrosion resistance, and decontamination tests were entirely satisfactory. This chloride based coating must be applied by specialized staff wearing masks.

The second coating selected (also produced by the Meyer Company) is a composite system consisting of:

- two layers of Pelableau 2000 (trade-mark), a strippable water-based acrylic emulsion paint, which is non-inflammable, odourless, and non-toxic and which fixes contamination;
- one layer of ISOX 1817 (trade-mark), a decontaminable polyurethane paint without effect on the strippability of the underlying Pelableau layer.

These encouraging results, which have been confirmed by tests in progress on multilayer systems and simple strippable paints from several firms, show that a satisfactory combination of properties can be found.

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

The first step will be a review of the existing decommissioning methodologies. The most critical aspects in costs and operational radiation exposure will be identified taking as a reference design a standard 1000 MWe Pressurized Water Reactor.

A closer analysis of the plant systems, where modifications in design could be useful, will be the next step. Special attention will be paid to systems which are required to be operational during the period between final shutdown and dismantling.

The space needed for access to components for dismantling operations and for shielding will be analysed and optimized; this analysis will include means for lifting and transport inside the controlled area. The requirements for cutting pipes and heavy equipment will be studied with a view to the use of existing handling equipment provided for normal reactor operation.

The possibilities for separating materials with different activity levels will be studied. The decontamination facilities needed for decommissioning purposes will be examined and, in particular, the possibility of designing the normal operation equipment for decontamination and secondary waste treatment will be considered.

7.7.3. Progress and Results

First, a bibliographic research into decommissioning procedures and design features facilitating decommissioning was carried out, and general aspects were pointed out.

Then, definition of a general logic tool to define a decommissioning plan has been started. The future standard nuclear power station of ENEL ("Progetto Unificato Nucleare") has been chosen as a basis for the analysis. The methodology is being developed for a preliminary design, considering:

- expected plant life history to predict the activity level of each sub-system;
- sub-systems and devices to be maintained in operation after the final reactor shutdown.

With these input data, one can derive three logic schemes (graphs), i.e., operational graph, lay-out graph, safety graph. Each graph represents a matrix of the interconnections of various sub-systems with the estimated relevance of each interconnection. These three matrixes allow to build several possible dismantling sequences respecting all the operational, lay-out and safety requirements.

A best sequence will be chosen by optimizing with respect to the most critical variable (e.g., minimum cost or minimum risk for the operators). The critical items of the sequence will then be pointed out. Their examination will produce the necessary feedback for possible modifications in plant design.

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 is to select and analyse realistic design concepts to minimizing 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

At present the biological shield in a nuclear power plant is constructed of concrete. Two main parameters lead to the activation of the concrete. These are the flux of neutrons escaping from the reactor and the quantity of isotopes in the concrete with a potential to transmute to a long-lived radioactive isotope as a consequence of a nuclear reaction. In order to reduce the activation in the concrete, one or both of these parameters must be reduced in value.

To reduce the neutron flux in the concrete it would be necessary to insert some neutron-attenuating material in the space available between the reactor and the concrete. Such a material must be capable of being inserted in the space available and withstanding the environment, and of reducing the activation of elements in any concrete which is beyond that material to below the datum value separating radioactive wastes from non-radioactive wastes. It must not give rise to an even greater problem at the dismantling phase (for example due to activation within the attenuating material).

The other parameter relates to the chemical composition of the concrete itself. The elements in concrete giving rise to radionuclides have been identified by numerous studies. Recommendations that quality assurance programmes ensure those elements are kept to a minimum have been made. However, control of the quantities of trace elements is difficult to achieve, since even 'parts per million' quantities of these can lead to significant activity.

The research work will therefore concentrate on achieving a reduction in the neutron flux in the concrete.

7.8.2.1. Alternative designs and candidate materials

The initial part of the work is to study the variations in biological shield designs in the various plant designs which are known. This study is aimed at examining the suitability of altering the design to include an inner neutron-attenuating layer. Candidate materials for this purpose will then be identified. This is done by surveying current neutron shielding materials and investigating their properties to see if they can be used in the environment between the reactor and the concrete shield. A list of candidate materials will be drawn up and reported.

7.8.2.2. Detailed analysis of two selected designs

Dependent on the number of candidate materials identified in the preceding work, the most promising materials and designs will be indicated. Taking two of these, the following analytical procedure will be carried out for each:

- the thickness of concrete for structural requirements will be estimated;
- neutron flux computations will be carried out to determine the thickness of the material required to meet the radiation protection requirements;
- the neutron flux computations will be used to generate three group (thermal, epi-thermal and fast) neutron fluxes for activation computations;
- the dismantling technique will be examined and the radioactive waste arisings compared to existing arisings;
- the costs for the conceptual design will be evaluated taking into consideration costs at the stages of construction, operation,

dismantling and waste management.

7.8.2.3. Comparative evaluation of each conceptual design

A comparative table for the materials and designs analysed will be drawn up against current designs and reported. Comments on the advantages and disadvantages will be made and the feasibility of each conceptual design will be addressed.

7.8.3. Progress and Results

7.8.3.1. Selection of design concepts

Phase 1 of the work involved a study of the various designs of biological shield for a variety of PWR and BWR power stations. Possibilities for altering the design to include neutron-attenuating materials were examined. The approach has been to insert a neutron-attenuating material between the reactor and the concrete biological shield. In addition, potential neutron-attenuating materials were identified by surveying available shielding materials and studying their properties to assess the feasibility of their inclusion in a design.

As a result of this phase of the work, a BWR design (the General Electric BWR/6) and a PWR design (the Kraftwerk Union PWR) were selected for further analysis. Both of these power station designs have a biological shield comprising two walls with the inner sacrificial wall non supporting. For these designs the concept involves using a suitable neutron-attenuating material other than concrete for the construction of the inner wall. Thirteen candidate materials were identified for consideration in design concepts.

7.8.3.2. Assessment of selected concepts

In Phase 2 of the work, the attenuation performance of the various candidate materials for both designs was assessed. Attenuation analysis was carried out to compare the attenuation performance of the candidate materials when substituted for the concrete in the inner sacrificial wall. The computer code ANISN in conjunction with the data library BUGLE80 was used for the attenuation calculations. As a result of this comparison, a shortlist of materials was drawn up for activation analysis using the isotope generation and depletion code ORIGEN2.

The investigations have looked at the nuclear aspects but have not examined other aspects, such as the engineering problems, which have to be considered to establish the overall feasibility of the concept.

From the information available on the shielding materials and the analysis carried out so far, it can be concluded that significant reductions in neutron flux and activation levels can be achieved in both the inner and outer shield walls.

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

Contractor: Dyckerhoff & Widmann, München, Germany

Contract: 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. Work Programme

7.9.2.1. Reduction of the activation of the biological shield

On the basis of chemical and activation analyses of the base materials for the reinforced concrete civil work surrounding the pressure vessel of a typical 1300 MWe Pressurized Water Reactor, the possibilities for reducing the level of activity in decommissioning by selection of the raw materials will be investigated. Elements leading to radio-isotopes with half-lives of more than four months, such as Ce, Ca, Co, Ni, C, Nb, Fe and Sm will be considered. Computations of neutron fluxes for designs with alternative shielding materials will be performed in order to evaluate the efficiency of alternative neutron shields. For the best alternatives of materials and design, the radioactive inventory after full reactor life and for various decay times will be calculated.

7.9.2.2. Alternative design facilitating the dismantling of biological shields

On the basis of the current criteria for the design of biological shields, an alternative design facilitating dismantling will be examined with a view to reducing radiation exposures during dismantling operations.

7.9.2.3. Analysis of the consequences of the alternative shield designs

This analysis will cover the following aspects:

- techniques for construction and dismantling of the shields;
- possible consequences for conditioning, transport and disposal of radioactive waste;
- construction cost of the biological shield;
- needs for future research on materials for biological shields.

7.9.3. Progress and Results

The KWU design of the biological shield and supporting structure for 1300 MWe Pressurized Water Reactor plants was chosen as a reference for this study. Input data for neutron-flux and activation calculations were compiled.

The following load cases are to be considered in the design:

- load cases during reactor operation:
  - . temperature difference (50°C);
  - . maximum conceivable accident (MCA);
  - . loss of coolant accident (LOCA);
  - . leak of the reactor pressure vessel;
  - . break of a reactor coolant pipe;
  - . quasi-static and dynamic jet forces;
  - . water pressure (78 kN/m^2);
- special load cases (external incidents):
  - . vertical earthquake intensity: $\pm$ 2.4 m/s^2;
  - . horizontal earthquake intensity: $\pm$ 2.0 m/s^2;
- tilt and sliding resistance.

Samples were taken from the biological shield concrete of various 1300 MWe Pressurized Water Reactor plants. The samples were identified and documented with respect to their mixing ratios, dates of production and strength values. Results of the chemical analyses are given in Table 15. These data will be completed by activation analyses for trace elements.

Table 15. Main constituents of the biological shield concrete of various nuclear power plants (wt%)

Constituent	KKG	KWG	KKI-2	KKP-2	KKU
SiO_2	65.1	55.86	62.70	68.55	75.60
TiO_2	0.28	0.16	0.09	0.14	0.15
Al_2O_3	4.67	4.42	2.12	5.08	4.56
Fe_2O_3	5.10	1.26	0.80	1.41	1.33
MnO	0.40	0.05	0.02	0.03	0.02
MgO	0.83	0.63	2.34	0.57	0.34
CaO	12.1	19.65	17.57	12.98	8.70
Na_2O	0.53	0.47	0.16	0.77	0.64
K_2O	1.20	1.37	0.50	1.54	1.68
P_2O_5	0.11	0.07	0.05	0.07	0.14
SO_3	(0.40)	0.38	0.47	0.57	0.41

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 - December 1983

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

The first step will be to define a documentation language suitable for documentation in nuclear power plants and to estimate the volume of documents to be processed. This includes:

- analysis of the structure and content of the existing relevant information;
- determination of a suitable classification by characteristics and

of the basic vocabulary;

- determination of the structure and content of the data set for documentation and retrieval;
- determination of the parts of the available information that are to be stored and estimation of the required storage capacity.

Thereafter, the data management processes will be determined and the programmes needed will be specified. Existing software and routines will be indicated.

The third step will be to specify suitable hardware including system software. The characteristics of the mass memory and of the user peripherals will be determined.

A suitable system for a micro-film system will be proposed. Finally, the operation of the whole system will be projected and described. This will include description of the management of the documentation process, management of filming, refreshment processes and means to assure protection of the system.

7.10.3. Progress and Results

Most of the existing documentation and registration systems are quite unsuitable to retrieve the information relevant to decommissioning which is hidden in the documents originating from specification, design, construction and operation of the plant. To develop specifications of a system which is suitable to this purpose, co-operation has been established with a utility which is decommissioning a nuclear power plant.

The data generated during the lifetime of a power station may be divided into the following parts:

- project data;
- data documenting the operating process;
- data documenting the final shutdown and the close-up phase;
- data documenting the decommissioning process.

A documentation language which is suitable to describe the information content of these four partial data stocks has to cover the following categories:

- segments (and subsegments) of data stock;
- types of documents;
- organizational framework;
- object system;

- functions and processes (within the plant's operation);
- inventory of nuclear activation;
- buildings, areas, rooms;
- activities (during construction, operation and decommissioning of the plant);
- documentation of the documentation system's hardware and software.

Proposals for the basic vocabulary to be used within these categories, for structure and contents of the data record, which stores the "Documentation Unit", were drawn up.

Initiating the software specification, the data management procedures to be executed in the course of describing the documentation data base, of inserting the documentation data and of retrieving, were determined, i.e.:

- insertion of the definition and description of the data bases and of the documentation language;
- loading and dumping of data files;
- on-line and off-line data input;
- on-line and off-line retrieval;
- alteration of the description of the data bases and of the documentation language.

The basic operations needed to execute these data management processes were specified.

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