Commission of the European Communities

nuclear science and technology

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

Second annual progress report (year 1981)

Report

EUR 8343 EN

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Directorate-General
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FOREWORD

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

The Council of the European Communities has adopted the programme in March 1979 (Ref. 2), 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 organisations and private firms in the Community under cost-sharing contracts with the Commission of the European Communities. The Commission budget planned for this five-year programme amounts to 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*.

In the initial period of the programme, the work to be undertaken has been defined in detail and research proposals have been called for and carefully examined, project by project. Accordingly, the first Annual Report described the work programme of most research contracts relating to the Projects N° 1 to 4 and the orientation of the Projects N° 5 to 7.

The present report describes the work programme of further contracts awarded through 1981 - most of them relating to the Projects N° 5 to 7 - and the progress and initial results of the research relating to the Projects N° 1 to 5. The Commission staff in charge of the programme during 1981 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

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 the extent of degradation of building materials have been made at four nuclear power stations in an age range of 15 to

21 years. The objective has been to determine the required extent of repairs and renovations to ensure that plant and building integrity is maintained for both the expected remaining part of the station operational lives and for a further period of about ten years to cover defuelling and removal of reactor peripheral plant.

The results of these investigations indicate that built-up bituminous felt roofing, coated metal wall cladding, metal wall louvres, sealing materials round doors, windows and wall penetrations and active drainage are areas where failure of materials has occurred, in some cases in a period as short as 14 years and, therefore, demand on-going maintenance for as long as these items are required to protect plant and services essential or useful to the decommissioning task.

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

Reports of surveys already made independently of this contract of the condition of buildings and structures at two nuclear and six coal-fired power stations, ranging in age from 15 to 28 years, are being studied and analysed. Although the contract research programme was originally based on nuclear power stations only, it is considered sensible to make use of available information from conventional stations, particularly those of greater age than the nuclear stations, since many constructional materials are common to both types and are in similarly exposed usage. This analysis has only recently commenced and the final results are not yet available, but problem areas identified, where material degradation has occurred, also include built-up bituminous felt roofing, roof asphalt, coated metal wall cladding, metal window frames and structural steel framing where the protective coating has been inadequate.

1.1.3.3. Sampling and testing of materials from nuclear power stations

Investigation into the extent of sampling and testing necessary to produce information of value for the full range of building materials used, even with samples from an unrepresentatively small number of nuclear stations, showed that such sampling, testing and reporting will have to be limited to key areas. These will be identified by the results of the work outlined in the foregoing paragraphs, where material performance has been poor in comparison with the expected life, but excluding materials such as bituminous felt which have a known average life below 30 years.

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

The plant components were classified according to their function and/or the nature of the material i.e.:

- concrete confinement;
- metal structures;
- circuits and pipes;
- ancillary equipment (auxiliary systems, ventilation, electrical equipment).

A catalogue of the components and equipment which are to be maintained, either in actual operation or in stand-by, has been prepared.

1.2.3.2. Study of aggressive agents

Using information collected through inquiries and literature research, the aggressive agents have been studied, in particular:

a) for concrete structures: weather conditions, i.e. temperature, humidity, salinity, wind, freeze/thaw cycles;
b) for metal structures: humidity, salinity, temperature, acidity, mechanical stress;
c) for electrical components (motors, power supply, relays...): mechanical stress (vibrations...), thermal and radiation stress.

Items a) and b) are almost complete. A technical appendix has been devoted to the behaviour of ancillary equipment to be considered specifically during Stage 1 or before reaching Stage 2.

1.2.3.3. Radiological aspects

Bibliographic data have been used to characterise the activity levels versus time, limiting this inventory to the identification of the causes of hazards to be considered:

- for accessibility of the components to be surveyed or maintained during Stage 1 or Stage 2;
- in case of failure of a safety barrier due to aging.

In order to confirm and complete the available data, information will be collected from measurements carried out in nuclear power plants during shutdown periods.

1.2.3.4. Preventive and curative measures

Preventive and curative technical measures have been developed. The work has been completed for the ancillary equipment; for the other components, the work is still in progress.

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. Delegation of an Expert to the USNRC* in Relation to the Clean-up of the TMI-2 Plant

Contractor: Comitato Nazionale per l'Energia Nucleare, Rome, Italy

Contract N°: DE-B-001-I Work Period: October 1980 - April 1981

2.1.1. Objective and Scope

The decommissioning of a nuclear power plant which has met with a major loss of coolant accident would need decontamination procedures different from those required in the case of a plant with a normal operation history, because:

- the degree and the radioelement composition of the contamination are different;
- the activity may be spread over places which are normally not contaminated and where access is difficult;
- large quantities of radioactive liquids and gases must be treated.

* United States Nuclear Regulatory Commission

The reactor N°2 of the Three Mile Island station (United States) offers valuable information on the contamination of a nuclear power plant which has met with an accident of this kind, and illustrates the problems encountered and the methods of decontamination used. The aim of this contract is to provide the Community with this information.

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

2.1.3. Progress and Results

The contractor had delegated an engineer to the USNRC to participate from October 1980 to April 1981 in the safety analysis of the clean-up operations proposed and performed by the owner of the TMI-2 plant. The delegate has reported on this assignment, covering mainly the following subjects:

- radioactive waste treatment (processing of released reactor coolant water, solidification of decontamination solutions);
- radioactive waste storage and shipment;
- radioactive discharges (essentially krypton-85);
- occupational exposure;
- population exposure due to krypton-85 venting;
- surface decontamination (auxiliary and fuel handling building, reactor building);
- information obtained through containment entries (radiation levels, condition of equipment);
- risks of radioactive leakages to the underground water.

2.2. Decontamination of Concrete Surfaces by Flame-scarfing

Contractor: Salzgitter A.G., Salzgitter, Germany

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

2.2.1. 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, radiological protection of the workers, feasibility of directly exhausting the combustion products to prevent recontamination, and magnitude of the decontamination factor.

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

2.2.3. Progress and Results

2.2.3.1. Preliminary tests

Preliminary tests were made with a hand-held multiple-nozzle heating torch and propane gas, an oxyacetylene cutting torch, and a mechanically operated flat-section torch with acetylene/oxygen supply. Torch form, gas admission pressures and torch feed rates as a function of concrete composition have been optimised. The tests showed that best thermal input is obtained using a flat-section jet torch.

2.2.3.2. Inactive tests on coated and uncoated concrete surfaces

Following the preliminary tests, flame-scarfing was tested on bare and coated concrete surfaces, in order to investigate whether the technique can be applied to coated surfaces. For this purpose, two nonradioactive specimen concrete plates (1 m x 0.8 m x 0.15 m square) were manufactured according to the concrete specification for nuclear power plants (density: 2.39 g/cm^3), and one of the plates was coated with epoxy resin (film thickness about 0.1 mm).

Flame-scarfing of the concrete surfaces was carried out in several parallel strips using a flat-section jet torch of 100 mm width. An acetylene-oxygen mixture with a mix ratio of 1:1.25 was used as fuel gas. The hand torch was manipulated by means of a remote-control feeding unit. The feed rate was set at a maximum of 0.5 m/min, which is the maximum rate possible without altering the equipment. The tests were carried out within an existing cabin that permits the arising dusts to be sucked off and determined quantitatively.

2.2.3.3. Results

In the tests with the coated specimen, the coating was completely burnt off by a single application of the flame and over the full width of the torch. The very thin concrete layer that does not contain aggregates and lies immediately below the coating was also removed at the same time. The cabin room air, however, was loaded with only a small part of the burnt-off material in form of aerosols. Most of the removed layer (approx. 1-2 mm) remained on the scarfed surface or fell down to the cabin floor. The quantity of the material dropped on the floor was not determined gravimetrically.

The aerosols arising during the flame scarfing tests were passed through an extraction duct without filtering, and isokinetic samples were taken and analysed. In addition, the dust resulting from subsequent treatment of the scarfed surfaces by brushing and grinding was sucked off with an EM 100 dust collecting probe (Messrs. Sartorius) in the form of isokinetic samples and subjected to a gravimetric analysis. The following values were obtained from an evaluation of the precision filters:

Test No.	Concrete finish	Surface treatment	Dust concentration (mg/m^3)
1	bare	flame-scarfing	26
2	bare	brushing, grinding	165
3	coated	flame-scarfing	22
4	coated	brushing, grinding	144

2.2.3.4. Experimental work in preparation

Further tests will be conducted on coated and bare surfaces, together with controlled filtering in the test plant's extraction duct of the aerosols arising from the flame-scarfing and during the subsequent treatment of the scarfed surface. Measurements of aerosol retention by the filters will be done and the main process parameters will be further optimised.

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

Contractor: Alsthom-Atlantique Neyrtec, Grenoble, France

Contract N°: DE-B-003-F Work Period: October 1980 - September 1981

2.3.1. Objective and Scope

Cavitation is the formation of vapour cavities in zones of a flowing liquid where pressure is low due to high local velocity. Subsequently, the cavities collapse abruptly, and if this takes place near the surface of a solid, the latter may be eroded.

The object of this research is to assess the feasibility of a surface erosion technique employing a cavitation flow of very high velocity. A major advantage of such a technique, as compared with other decontamination techniques, would be to operate with pure water and, hence, to produce an effluent which can be treated easily, resulting in a small quantity of secondary radioactive waste.

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

2.3.3. Progress and Results

2.3.3.1. Hydraulic design of the cavitation element

Using an experimental loop with a maximum flow-rate of 90 l/s at 6 bar, tests have been performed in order to optimise the geometry of the cavitation element.

The chosen basic configuration of the cavitation element (see Figure 1) comprises a flow obstacle, for generating a large number of vapour cavities, and a wedge-shaped piece, creating a pressure increase, to produce abrupt collapse of the cavities. Compared with the characteristics considered for the real application of the technique, the dimensions of the cavitation element were four times larger and the flow velocity (20 m/s) correspondingly lower, so that hydraulic similarity (i.e. equal Reynolds number) was obtained.

Transparent walls of the test section enabled visualisation of the formation and collapse of vapour cavities, using fast-cinematography techniques. The impacts on the target wall produced by collapsing vapour cavities were recorded using high frequency acoustic sensors and an oscilloscope.

The geometric characteristics of the cavitation element (height, form and position of the obstacle, angle of enlargement, etc.) have been varied and optimised for producing a large number of intense impacts. The influence of a gap between the base of the obstacle and the adjacent wall - such a gap may be inevitable in the real application of the technique - has also been studied; a 2 mm gap reduced slightly the intensity of the impacts, but increased slightly their frequency.

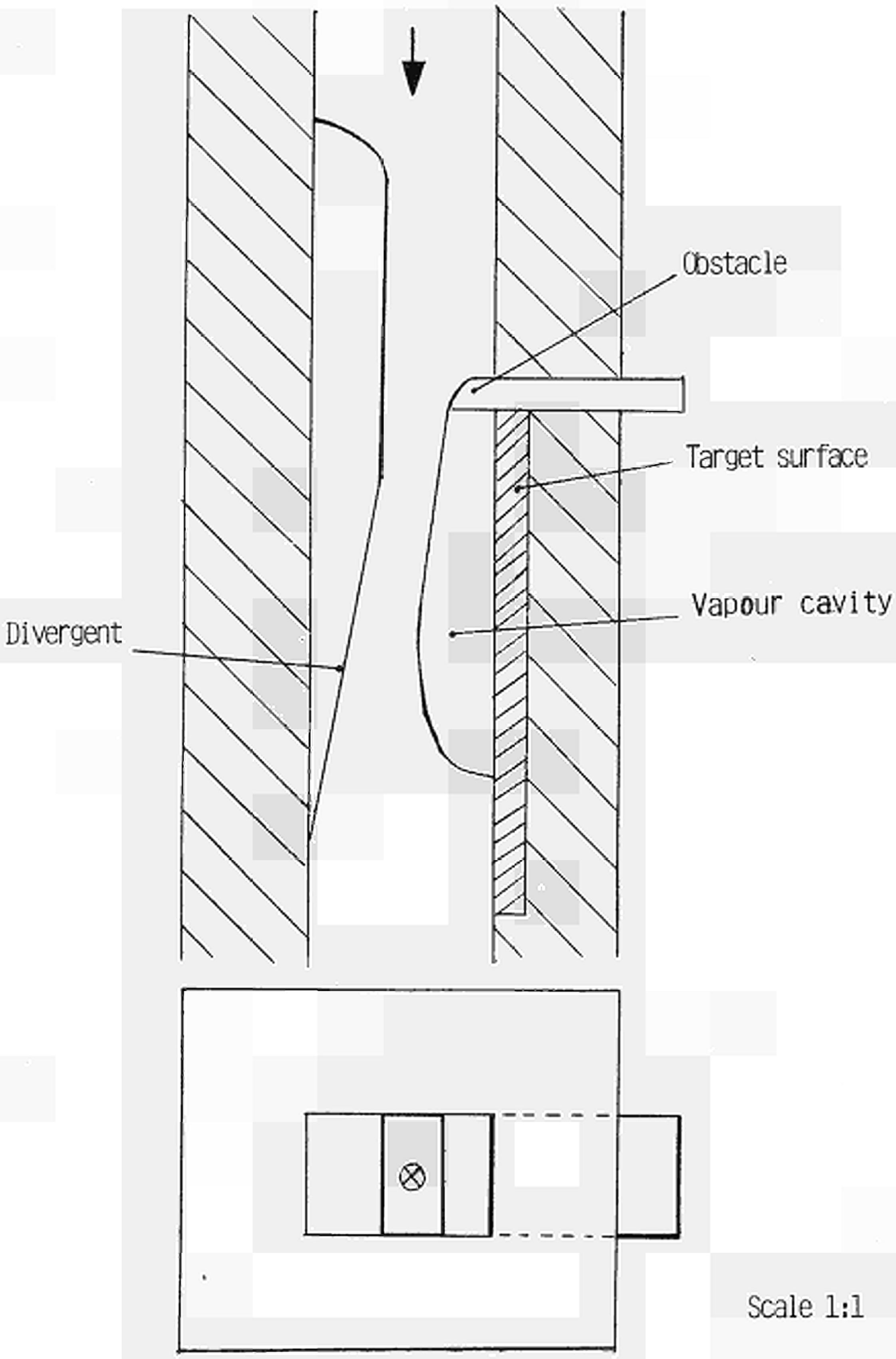


Figure 1. Test section for hydraulic design of the cavitation element.

2.3.3.2. Erosion tests

Erosion tests have been performed with a pressure of 58 bar upstream the cavitation element which had a minimum flow section of 0.5 cm x 1 cm. The flow rate was about 3 l/s, corresponding to a maximum flow velocity of 60 m/s. Since the characteristic dimensions of the cavitation element were half of those used in the low-pressure tests described in the previous paragraph, Reynolds numbers were in the same range.

Transparent side-walls of the test section enabled visualisation of the cavitation flow. The cavitation element was immerged in a water-filled tub and an adjustable slide plate was placed at the outlet of the cavitation element to the tub. The slide plate made it possible to vary the outlet flow section and, thereby, the pressure in the zone of erosion.

Erosion tests made with aluminium alloy samples showed a decisive influence of the outlet flow section on the pattern of cavitation and erosion. Maximum erosion was produced with an outlet section of 0.8 cm^2; in this case, a short vapour pocket was formed behind the flow obstacle. After 8 h exposure to this type of cavitation, the volume of material removed, as determined by surface profile measurements, was 1.2 mm^3. On an anodised aluminium sample treated for 1 h, the anodisation layer was removed on an area of about 0.8 cm^2.

On a stainless steel sample treated for 1 h, a mark of very slight erosion was visible.

The tests have also shown erosion of the side-walls of the cavitation element.

2.3.3.3. Extrapolation of the erosion tests

For extrapolation of the results of the erosion tests, it has been assumed,

- that the erosion rate increases with the sixth power of the flow velocity or, accordingly, with the third power of the pump pressure (at constant dimensions of the cavitation element);
- that the erosion rate is inversely proportional to the characteristical dimension of the cavitation element (at constant flow velocity);
- that it is feasible to reduce the characteristical dimension of the cavitation element by a factor of three.

Taking as a basis the erosion rate obtained on an aluminium alloy sample (i.e. 1.2 mm^3 in 8 h), it has been estimated that of the order of 1 m^2/h of surface could be cleaned (removal of 1 micron surface layer) with a pump pressure of 300 bar and a flow-rate of 7 l/s.

2.3.3.4. Proposal of further development

It is proposed to develop a new type of cavitation element, based on a circular geometry. The water would 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. This geometry eliminates the side-walls of the cavitation stream and, thereby, the problem of their erosion and the associated energy loss.

The contractor has proposed a follow-up contract having as its subject the development of the new cavitation element.

2.4. Composition of Contamination Layers and Efficiency of Decontamination

Contractor: Kernkraftwerk Lingen GmbH, Lingen, Germany

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

2.4.1. Objective and Scope

The purpose of this research is the characterisation 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 shutdown in 1977 after nine years of operation).

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

2.4.3. Progress and Results

2.4.3.1. Sample preparation and non-radioactive tests

At three locations of the primary reactor circuit, tube sections having carried steam, condensate and reactor water, respectively, have been cut out and segmented. The tube having carried reactor water showed a dose rate of 16 R/h measured at the outer surface.

Nonradioactive tests with an electrolytical method have been made in order to determine the operational parameters for removing 1 micron of wall thickness at a time without a smoothing effect. With the electrolytical solution used and an optimum current of 4.2 A, 1 micron is removed on an area of 100 cm^2 in about three minutes. The removed thickness was measured using three different methods and good agreement was found.

2.4.3.2. Penetration measurements

The first measurements of the penetration depth of radioactive isotopes into the base material have been carried out on two tube samples from the steam system. After removal of the oxide deposit from the inner surface of the tube, the electrolytic removal of the base material took place in consecutive layers of 1 micron thickness. After removal of about 10 micron, the specific cobalt-60 activity changed only negligibly. This specific activity extends over the whole wall-thickness and originates probably from activation during reactor operation (see Figure 2).

2.4.3.3. Research on agressive chemical solutions

Preliminary tests have been made in order to determine a suitable pickling solution. Of the various pickling solutions tried out, only two (22% HCL + 5% HNO_3 and 10% HCL + 1% H_2O_2) produced satisfactory removal within 4 h.

2.5. 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

2.5.1. 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.

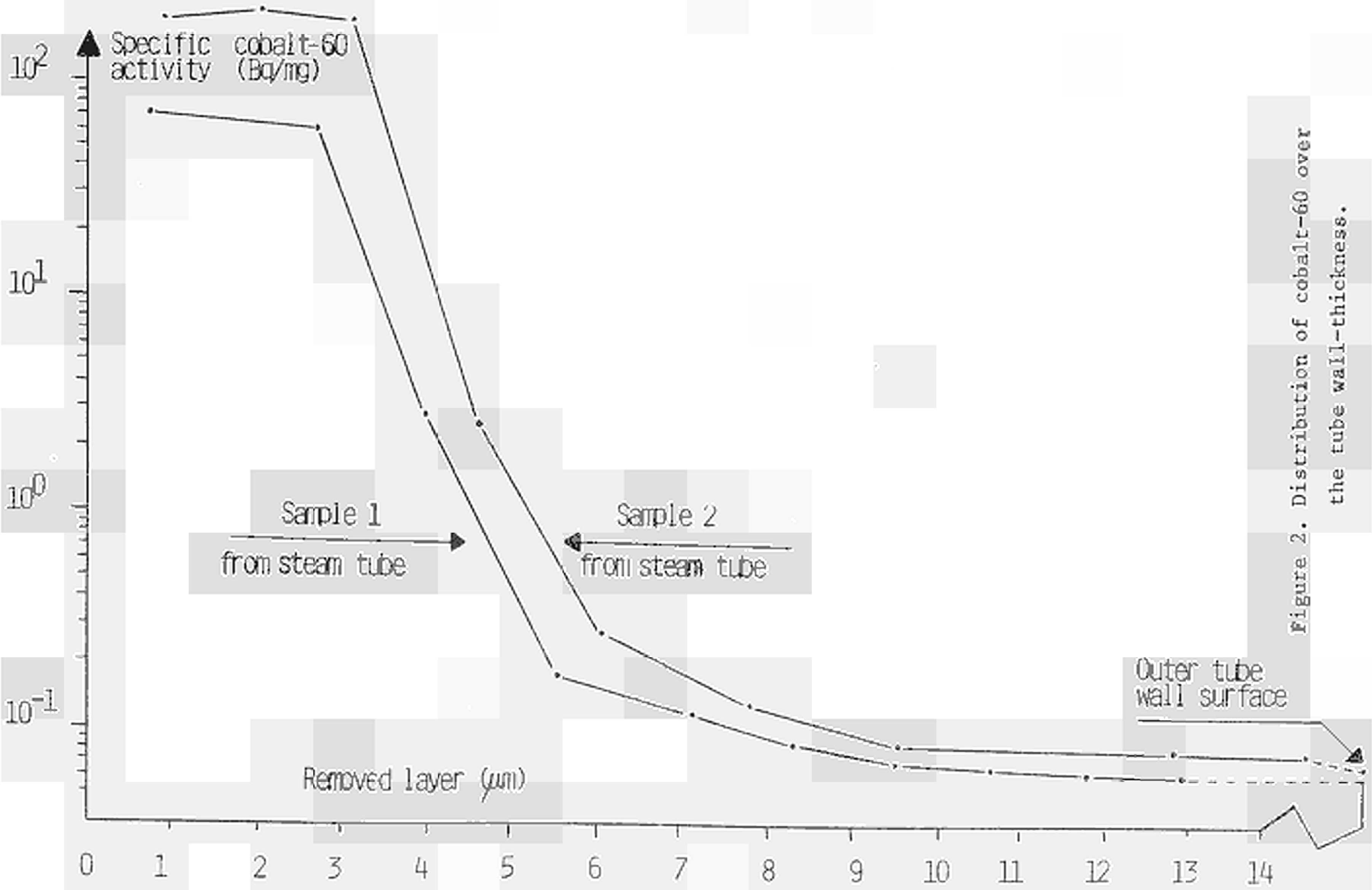


Figure 2. Distribution of cobalt-60 over the tube wall-thickness.

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

2.5.3. Progress and Results

2.5.3.1. Sample characterisation

Samples of contaminated material, mainly stainless steel, were taken from various locations of the primary circuit of the Garigliano reactor. The maximum dose rate was 4 rem/h, measured on the contaminated surface.

A preliminary characterisation of the contaminated oxide layers showed that the presence of magnetite, nickel ferrite and copper oxides can be related to the history of coolant water chemistry. Two periods were distinguished, according to the feed-water preheater materials, which had been copper-nickel alloy and Monel before 1968 and stainless steel afterwards. Correspondingly, the content of iron, copper and nickel in the primary coolant water had been much higher during the first period.

Radiometric measurements on some specimens of various primary tubing showed surface contaminations of about 100 nCi/cm^2 of gamma activity, mainly due to cobalt-60, and about 5 nCi/cm^2 of alpha activity, mainly due to plutonium-239 and americium-241 or plutonium-238, with traces of curium-244.

2.5.3.2. DECO loop

Design and assembling of the DECO loop have been completed and commissioning will be terminated by March 1982. The flow-sheet of the loop is shown in Figure 3.

2.5.3.3. Static decontamination tests

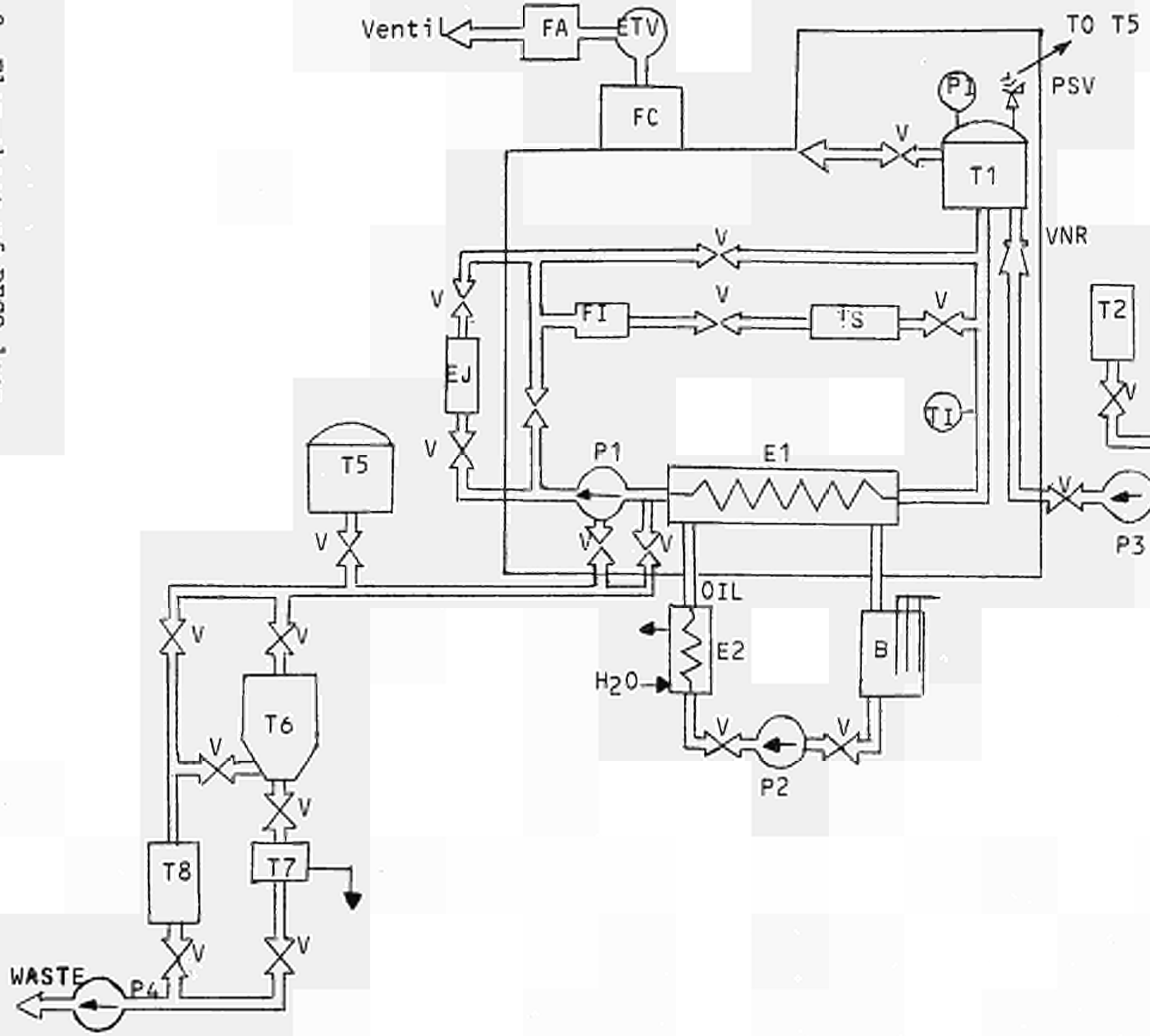
Static batch tests have been made, using the following hard etching solutions (with constant ionic strength):

a) 1.5% hydrofluoric acid + 15% nitric acid;

b) 13.4% hydrochloric acid;

c) 4% hydrochloric acid + 13% nitric acid.

Both nonradioactive samples, taken from an experimental loop operated for about 7000h in BWR conditions, and contaminated samples have been treated. The results indicate a close connection between oxide features and etching effectiveness, solution c) appearing to be most effective with the nonradioactive samples, and solution a) with the contaminated samples.



P1 PRIMARY PUMP
P2 FEED EXCHANGER PUMP
P3 FEED PUMP
P4 WASTE PUMP
T1 EXPANSION TANK
T2 FEED REAGENT TANK
T3 FEED REAGENT TANK
T4 FEED WATER TANK
T5 SAFETY TANK
T6 NEUT. AND FLOC. TANK
T7 GRAVITY WASTE TANK
T8 GRAVITY LIQUID WASTE TANK
E1 PRIMARY HEAT EXCHANGER
E2 SECONDARY EXCHANGER
B BOILER
TS TEST SECTION
EJ ION EXCHANGER
V VALVE
VR REGULATION VALVE
VNR CHECK VALVE
PSV RELIEF VALVE
ETV CENTRIFUGAL FAN
FC CHEMICAL FILTER
FA ABSOLUTE FILTER
PI PRESSURE INDICATOR
FI FLOW INDICATOR
TI TEMPERATURE INDICATOR

Figure 3. Flow-sheet of DECO loop.

Static decontamination tests were made with 1 litre of etching solution and about 10 cm^2 of contaminated surface. Very high decontamination factors (greater than 1000) were obtained with solutions a) and b), resulting in residual contaminations comparable to ground level; however, the etching morphology was different: solution b) dissolved the oxide surface layer progressively, whereas solution a) caused a swelling-flaking of the oxide without dissolving the detached scales. The test temperature (30 to 80°C) appears to affect only the time required for oxide removal. Very low decontamination factors (lower than 2) were obtained with solution c).

A test with 3 wt% oxalic acid + 3 wt% citric acid, ammonia buffered to 3.5 pH, at 80°C, showed incomplete oxide removal and a very low decontamination factor (1.3).

2.6. Development of Economic Decontamination Procedures

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen, Germany

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

2.6.1. 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.

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

2.6.3. Progress and Results

2.6.3.1 Radiation levels in the plant

Dose-rate measurements have been performed in the whole plant, allowing seven zones to be defined, from less than 0.5 mrem/h to more than

100 mrem/h inside the reactor.

2.6.3.2. Contamination layers on primary circuit pipes

Samples of horizontal ferritic steel pipes of 500 mm diameter of the condensate duct and of the steam duct of the primary circuit have been examined.

On the condensate pipe, an inner surface oxide layer of 100 mg/cm^2 was found, with the following metal element contents: 70 wt% of iron, 0.6 wt% of copper and of nickel, 1.8 wt% of manganese and 0.2 wt% of chromium and of cobalt. The contamination increased from the top (0.23 kBq/cm^2) to the bottom (0.73 kBq/cm^2) of the pipe. The contribution of individual radionuclides to the total contamination varied with the location within the following ranges: cobalt-60: 16 to 55%; cesium-134: 3 to 6%; cesium-137: 41 to 77%.

On the steam pipe, an oxide layer of about 1.6 mg/cm^2 and contamination of 17 kBq/cm^2 were measured, except for a 10 cm large streak at the bottom of the pipe, where a thicker oxide layer (22 mg/cm^2) and higher contamination (60 kBq/cm^2) were found. This streak is probably due to the presence, during operation of the system, of a rill of water condensed from the saturated steam. Cobalt-60 accounts for 99% of the contamination of the steam pipe.

2.6.3.3. Decontamination tests

In preliminary screening experiments, 20 organic and inorganic acids have been tested on 4 cm x 4 cm samples from the condensate duct. Tartaric acid and citric acid were found to be comparable to the best inorganic acids in decontamination efficiency and have, because of their unproblematic application, been selected for further studies.

The tests have shown that no further decontamination effect is achieved after 1h of treatment. In order to reach the limit for unrestricted release, the treatment has to be repeated using clean solutions.

With an oxidation pretreatment (APAC type) and two consecutive treatments (3h, 92°C) with 5% tartaric acid, condensate samples have been decontaminated from initial levels up to 1 kBq/cm^2 to residual levels of 2 Bq/cm^2 or less.

2.7. 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

2.7.1. Objective and Scope

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

The objective of this research 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.

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

2.7.3. Progress and Results

2.7.3.1. Preparation and characterisation of gels

Four gels, i.e. diopside gel, glycerophtalic gel, glycerophosphoric gel and silica gel, have been selected as candidate carrier materials for decontaminants. These gels have been produced in the laboratory and tested with regard to viscosity and mass of gel adhering to a vertical stainless steel surface, as a function of temperature (0 to 50°C) and of pH value. The tests showed that the selected gels behave like a viscous liquid. Nevertheless, their properties of adherence are much better than those of aqueous solutions.

2.7.3.2. Preparation and characterisation of gels carrying decontaminants

The preparation and properties of the following combinations of gels and decontaminants have been studied:

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

The products have been tested with regard to evolution of viscosity with time (checking on chemical stability), mass adhering to a vertical stainless steel surface and degreasing power. Good stability and high viscosity have been found. For application of the products by spraying, it would be necessary to reduce the viscosity. This could be achieved by increasing the decontaminant content and/or the working temperature.

2.8. 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

2.8.1. 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 optimised will be evaluated with a view to their industrial application, taking into account, in particular, their effectiveness, limitations, costs and radiological consequences.

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

2.8.3. Progress and Results

Work during 1981 was concerned with chemical decontamination techniques.

Two types of stainless steel samples have been used in decontamination tests, i.e.:

- samples contaminated with beta-gamma emitters (cesium-137, strontium-90, cobalt-60), taken from a shelf used in a hot cell;
- uranium contaminated samples, taken from a tub used in an enriched uranium treatment facility.

The samples have been pre-decontaminated by wiping with nitric acid and degreased with caustic soda (0.5h; 60°C). Subsequently, these samples have been used to test the efficiency of the decontaminants most currently used in the decontamination facilities of the Commissariat à l'Energie Atomique (27 different treatments). In parallel tests, the aggressivity of the baths has been examined, using nonradioactive stainless steel samples (3 different grades) and determining the rate of thickness reduction and the evolution of surface roughness parameters.

Efficient decontamination of the uranium contaminated samples was easily achieved using moderately strong baths (i.e. oxalic acid + hydrogen peroxide, sodium bicarbonate + hydrogen peroxide, potassium disulphide). Decontamination of the samples contaminated with beta-gamma emitters required strong baths (i.e. hydrofluoric-nitric acid with or without oxalic acid). However, one type of moderately strong bath (sulfamic acid + ferric sulphate) resulted also in good decontamination. With mechanically marked (cold-rolling) stainless steel, the use of strong baths (containing hydrofluoric acid) is necessary for obtaining good decontamination.

Moreover, the efficiency of various reagents towards magnetite was measured. Quick dissolution of magnetite was obtained either using oxalic acid at 80°C or, at room temperature, using hydrofluoric acid at pH close to 3.

2.9. 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

2.9.1. 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 optimisation of concepts for decommissioning. The objective of this study is to prepare such information, with due regard to existing national regulations on limit values for the unrestricted release of waste arising from decommissioning operations and taking as a basis existing decontamination techniques.

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

2.9.3. Progress and Results

2.9.3.1. Regulatory aspects

Regulatory aspects have been investigated with co-operation of the Institut für Völkerrecht of the University of Göttingen. It appears that no regulatory criteria (i.e. residual activity limits and proving procedures) for free disposal of large quantities of decontaminated material are existing in Community member countries.

Limit values of both 3.7 Bq/g and 0.37 Bq/cm^2 have been assumed for the purpose of this study.

2.9.3.2. Classification of components

After classification of the components which are obviously decontaminable (e.g. airborne contamination) or not decontaminable (e.g. activated), the following aspects have been considered:

- occupational exposure;
- costs;
- mass/surface ratio;
- geometry, in particular with regard to the possibility of measuring contamination;
- nature and level of contamination.

The classification drawn up so far is given in Table 1. The components listed as borderline cases will be further investigated. A particular difficulty lies in the scarcity of information on the quantities of secondary waste arising from strong decontamination methods.

Table 1. Classification of LWR Components

(Masses without buildings, for 1200 MWe plants)

Decontaminable (examples)	Borderline cases needing further investigation	Not decontaminable (examples)
- containment - ventilation system - cables - electric motors - hoists	- primary loop, steam part (BWR) - turbine (BWR) - condensor (BWR) - spent fuel storage pitliner - steam generator, steam part (PWR) - reactor pressure vessel cover (BWR)	- activated components - primary loop (PWR) - primary loop, water part (BWR) - biological shield, inner part - building surfaces
6,000 Mg (PWR) 8,000 Mg (BWR)	600 Mg (PWR) 6,000 Mg (BWR)	7,000 Mg (PWR) 2,500 Mg (BWR)

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.

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 is 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 thermal segmentation of heavy concrete and steel components by means of oxygen lances, powder cutting, oxyacetylene cutting, and plasma cutting has been tested.

In order to gain experience in handling oxygen lances, and to determine the optimal burning parameters, cuts using 1/4", 3/8" and 1/2" lances were carried out on selected heavy concrete blocks. The oxygen consumption, lance wear, cutting speed, and the amount of slag deposit were measured in relation to the oxygen pressure of the various lance types. This revealed that for heavy concrete the 1/4" lance produces the best results. In particular, the amount of slag deposit (secondary waste) was negligible for this type of lance. This result shows that for cutting concrete in nuclear power stations, the 1/4" lance is to be recommended. However, this type of lance showed less than satisfactory results for cutting steel.

During these tests the amount of dust produced, and the distribution of dust particle sizes were recorded. When designing filter installations with a view to using this process in the nuclear field, such knowledge is essential. In addition, these dust measurements enabled the suitability of the filtering systems used during the tests (cassette filters and particle suspension filters) to be assessed. When cutting concrete and steel components, the maximum particle size lies in the region of 0.3 micron, so that for separation purposes only particle suspension filters can be used.

When testing the powder cutting process, it was shown that it can only be used for splitting concrete structures if the wall thickness does not exceed 1000 mm and sufficient space exists where the slag deposits emerge. Comparable tests have proven that the powder cutting process gives better results than the oxygen lance and oxyacetylene cutting processes in respect to cutting rate and slag deposit.

Whereas the processes described so far are suitable for cutting concrete and practically all types of steel, the use of the oxyacetylene cutting process is restricted to ferritic steels. The test results underlined the known advantages of this process, such as its easy handling, little dust and slag deposit, low consumption of fuel gases, and negligible proneness to malfunction.

As a further segmenting technique, plasma cutting was tested on austenitic and ferritic plates. Here too, very good cutting rates were achieved. One disadvantage, however, was the high proneness to malfunction. When cutting items with an irregular surface, maintaining the necessary separation distance presents problems (frequent breaking of the arc and frequent nozzle changes due to varying thickness). For this reason, the use of the plasma cutting process is only advisable for cutting level surfaces, or when other processes with a lesser proneness to malfunction cannot be fallen back upon.

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 - June 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 licencee 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

After the bibliographic analysis, the evaluation of the various plasma cutting techniques has been completed. AMN, in cooperation with Messer Griesheim, analysed the actual possibility of using the plasma arc cutting technique in the decommissioning of a nuclear power plant and specified the R&D work necessary both in the field of control and in the field of cutting units and technology.

In these analyses, the Ansaldo/General Electric BWR/Mark II has been selected as a reference plant. The problems related to the application of the plasma cutting technique to this type of reactor have been analysed and a sequence in which plasma cuts mights be carried through during the dismounting of this reactor was tentatively defined.

Considering the problems revealed by this analysis, an experimental programme was planned with Messer Griesheim at the Hamburg University at the end of 1981. The scope of these tests was to optimise the cutting parameters (such as diameter of the plasma torch nozzle, distance between nozzle and workpiece, cutting speed, notching delay, voltage and current characteristics, gas flow rate, air flow) and the configuration of the cut and flash-line. The cutting tests were performed in an immersion basin with two different types of plasma torch at water depths of 40 cm and 1 m respectively.

In the field of the thermo-lance cutting technique, after the bibliographic analysis and contacts with companies qualified in this technique, an agreement was reached with Fondibeton (Padova, Italy), to perform some tests with the thermo-lance. The size of the reinforced concrete specimens to be cut has been selected in order to represent typical reinforced concrete walls existing in the reference BWR plant. The test execution is scheduled in 1982.

In cooperation with CNEN, a programme to analyse the radiological protection and safety aspects of the cutting operations will be started in the first half of 1982. The plasma torch machine has already been placed at the CNEN Casaccia Centre and the equipment and instrumentation is going to be arranged.

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

The work carried out may be divided into:

- visits to contractors and plant suppliers;
- demonstrations and field trials of saw machines;
- initiating the design and manufacture of a large diamond tipped

saw unit.

At the outset of this contract a short literature survey was made of the use of concrete sawing in the civil engineering industry. Visits were paid to the principal saw blade manufacturers, in order to understand the technical processes and limitations in respect to high quality reinforced concrete, and to firms specialising in the use of sawing machines.

To establish the basis of the technology and to check claims about performance, the technique has been demonstrated on the turbine foundation blocks at an old coal fired station. Pieces of concrete 375 mm square section and 1 m long were removed from the face of a block in a manner that could be applied to an inner face of a biological shield.

The hydraulic powered wall saw comprised a manual/hydraulic propelled carriage carrying retracting rollers, a heavy duty pivoting gearbox and a 20 kW hydraulic drive motor mounted on 3.5 m of track attached by stools to the face of the turbine block. During the trials, measurements were taken of cutting speeds, diamond segment wear, cooling water and electrical power consumption.

The sawing trial was completed successfully, but removal rates were much slower than expected. No serious dust, noise or vibration problems were encountered and cooling water supplies and electricity consumptions were reasonable. The lack of power in the power pack led to many adjustments being made to the cutting depth and speed of traverse of the saw.

From this first trial it was decided that development of a saw unit of sufficient power to cut to a depth of 1 m that could be controlled remotely would be justified. This conclusion was supported by the outcome of a second field trial carried out in March, which demonstrated a 2.1 m diameter wall saw. This successfully made a number of cuts in a retaining wall at the now disused Swindon Power Station. The complete saw unit comprised a diesel driven hydraulic power pack supplying a 35 kW hydraulic motor which was connected to the saw blade. The feed was adjusted manually and a small hydraulic motor drove the saw along a rack and pinion track securely attached to the wall. A cutting depth of 835 mm was achieved, but it was clear that if the method was to be applied to a reactor, modifications to the carriage and blade cooling system would be necessary and the driving power would have to be increased.

In November 1981, work has been started to design and manufacture a large diamond tipped saw unit and power pack with associated controls. The saw unit will be capable of being mounted on the jib head of a crane for vertical and horizontal cutting. The saw blade will be 2.5 m diameter and driven by a high torque slow speed axial piston swash plate motor of 75 kW rating. The saw carriage will be in the form of an oblong frame with two driving wheels corner mounted at one end and two castor wheels mounted centrally at the other end. As presently used in commercial operations, if a blade meets an obstruction, the overloading of the motor safeguards the saw blade, but with the uprating of the motor this protection of the blade is not available. Consequently, the relationship between the saw shaft power available and the load imposed by the saw traverse or feed is being designed to avoid saw blade damage. The testing of this saw unit should commence in April 1982.

3.4. Plasma-oxygen Cutting of Steel Pressure Vessels

Contractor: Salzgitter AG, Salzgitter, Germany

Contract N°: DE-C-004-D Work Period: April 1980 → September 1982

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, optimisation 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

During 1980 the existing test stand had been adapted and several cutting tests were performed that the permitted initial statements to be made regarding the general applicability of the combination cutter. These tests showed that the high-melting oxides in the cladding detracted from the quality of the cut, but did not make cutting impossible.

In 1981, systematic series tests were carried out for combined plasma-oxyacetylene cutting of clad plates up to 300 mm thick in horizontal and vertical cutting position. The base material used was ferritic steel grade St 37-2 (thicknesses: 150, 200, 250 and 300 mm) and the cladding was strips of plate of austenitic steel grade X 10 CrNiTi 18 9. (thickness: 10 mm).

Although there was a large difference between the gas feed rates required in the various cutting series, due to the different cutting depths, combination cuts could be carried out at speeds of 40 to 180 mm/min, because the feed rate of the plasma torch could be set to match that of the oxyacetylene torch without interrupting the arc of the plasma torch.

From the results obtained so far it appears that cuts up to 300 mm deep can be made in vertical and in horizontal position using a combined plasma-oxyacetylene cutting torch.

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, the contractor will refer to stations incorporating Pressurized Water Reactors and Gas-cooled Reactors.

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

3.5.3. Progress and Results

The research carried out in 1981 consisted of a preliminary literature search and specification, construction and testing of prototype equipment for the protection of mirrors and focusing optics from fumes and particles released in the laser cutting of steel and, more importantly, concrete. This equipment was built and operated within the limitations imposed by the existing laser system.

The cutting tests were performed using the AVCO-Everett 15 kW carbon dioxide laser installed at the Centro Ricerche FIAT, Orbassano. The welding and cutting work station equipped with a comprehensive workpiece movement system, laser optics and beam focusing was used. A basic fume extraction system is installed.

The cutting technique is as follows. The laser beam interacts with the material surface to form a "key-hole" of molten metal whose depth is proportional to the applied power and the laser/material interaction time. When the beam is moved relatively to the material (or viceversa), the molten zone flows behind the initial "key-hole". If a suitable gas jet is directed onto this zone, the molten metal is expelled mechanically and vapour removal greatly facilitated. If the gas is also reactive, certain materials (mainly ferrous) initiate a violent exothermic reaction which increases the laser cutting ability due to the higher temperature produced and the more volatile reaction products which are easier eliminated. In thick sections, the cut surfaces deriving from the two reactions, i.e. laser melting and chemical reaction, are often distinguishable both in depth and appearance, depending on operating conditions. The laser reaction cut surface is normally smooth with slight and regularly spaced stripes, while the other surface is heavily ribbed and irregular.

In the course of these checks, using laser powers up to 14 kW, preliminary process parameter have been defined for selected materials. Steel sections up to 100 mm have been cut. The obtained results, apart form giving relationships between laser power, cutting velocity and material thickness, show that as the angle of incidence (respect to the vertical direction) of the assistant gas decreases towards 10°, cutting efficiency rises. The cutting performance improves if the assistant gas pressure is risen to 6 or 7 atm, but falls back again at 10 atm (with 25 mm thick material).

Under the experimental conditions applied, the following preliminary conclusions can be drawn, which will be checked by more specific experiments in the next phase of this research:

- Beam focus: moving the beam from the material surface inward the material could allow a velocity increase of up to 20%.
- Laser power: at the power levels used, an increase of 1 kW allows

a 10% average velocity increase.

- Material thickness: 1 mm increase in thickness corresponds to an average velocity decrease of 10%.
- Material composition: alloying elements have a marked negative influence on the velocity.
- Reactive assistant gas: the change from air to oxygen allows a large increase in cutting velocity. Assuming similarity between the steel grades UNI C 10 and Fe 42 C, speed increases up to 300% seem possible at 9.5 kW.

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 optimise 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

3.6.2.1. General basic studies

These studies include:

- literature study of present day demolition and quarrying techniques;
- studies of reactions which might be utilized in the demolition of reinforced concrete structures (e.g. stress wave focusing and reflection, chemical reactions);
- outline of the demolition relevant characteristics of the reference concrete structures considered, as defined under 3.6.1.

3.6.2.2. Assessment of model techniques

The validity of model techniques will be assessed together with a full dimensional analysis to determine where non-scaling effects may be important (about 6 tests).

3.6.2.3. Studies of explosive charge effects

The following effects will be studied:

- size, placing and spacing of the explosive charges;
- firing sequence;
- use of high or low explosives.

The best mechanism to induce maximum damage by material shattering or face spall will be assessed (about 15 demolition tests).

3.6.2.4. Studies of lined concrete structures

Steel-lined concrete structures will be studied to assess whether explosives may be used effectively to cut up the liner and the most efficacious way in which this can be achieved, e.g. by the use of liquid explosives filled in the cooling ducts of the liner (about 2 demolition tests).

3.6.2.5. Assessment of stress wave disintegration

The feasibility of stress wave disintegration will be assessed including studies of stress focussing. This assessment will be done mainly by computation, but includes about 2 demolition tests on concrete specimen of an appropriate size.

3.6.2.6. Studies of special effects

The following effects will be studied (about 12 demolition tests) :

- various methods of tamping the holes in which the explosive charges are placed;
- shaped charge effects for stripping cover, cutting reinforcement and boring.

3.6.2.7. Experiments with simple models

Experiments with slabs and open ended hollow cylinders will be carried out to determine how controlled disassembly within the integral dust screen can be effected (about 18 models of an approximate scale of

1/4 of the real structures considered).

3.6.2.8. Experiments with closed ended cylinders

Experiments with closed ended hollow cylinders will be carried out to confirm that the techniques may be used with complete and monolithic structures (at least 1 model of an approximate scale of 1/4 of the real structures considered).

3.6.2.9. Studies of size effects with larger structures

Size effects are being studied in firings covered in Section 3.6.2.2. Firing under this section will be undertaken if a suitable existing structure is found and if planning permission for the test can be obtained.

3.6.2.10 Conclusive assessment

On the basis of the results of the preceding studies the explosive disassembly of the various structures considered, as defined under 3.6.1. will be tentatively assessed, including the necessary preparatory and complementary operations (e.g. liner removal, drilling of holes for explosive charges, disassembly of reinforced structures after shattering by explosive techniques, cutting of reinforcement bars, etc.). This assessment comprises:

- the description of the optimum procedures, considering the feasibility of successive dissassembly of active parts (i.e. the activated inner zone and contaminated surface layers) and inactive parts;
- discussion of explosive hazards;
- estimate of the radiation exposure of workers;
- cost estimate;
- identification of uncertainties and gaps of information and of future work required.

During the tests, high-speed cine records will be taken as required and in some firings flash X-ray methods may be used. Following tests, detailed inspection and measurements of targets will be made to indicate crater size, etc. and, where appropriate, targets will be sectioned. Some stress wave sensors and crack indicators may be introduced in targets as appears necessary.

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 immobilise 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 1981

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 and deep geologic disposal. An extension of the contract to include shallow land burial is being envisaged.

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

4.1.3. Progress and Results

4.1.3.1. Activation product inventories

Data on the impurity content of unirradiated Magnox and AGR graphite have been reviewed and reference impurity levels have been defined for each case. Inventories of activation products in the graphite of Magnox reactors and AGRs were calculated using the UKAEA reactor inventory code FISPIN. The predicted inventory of the 26 UK Magnox reactors is given in Table 2.

Table 2. Predicted radionuclide inventory (Bq) of graphite from decommissioning of the 26 UK Magnox reactors

(After 40 years reactor operation followed by 10 years decay)

^{3}H	$2.6\ 10^{15}$	^{60}Co	$5.9\ 10^{14}$	^{113m}Cd	$2.4\ 10^{11}$
^{10}Be	$1.6\ 10^{12}$	^{63}Ni	$3.0\ 10^{14}$	^{121m}Sn	$1.0\ 10^{12}$
^{14}C	$1.9\ 10^{15}$	^{65}Zn	$4.8\ 10^{9}$	^{133}Ba	$1.3\ 10^{13}$
^{36}Cl	$2.1\ 10^{13}$	^{93}Mo	$2.0\ 10^{10}$	^{152}Eu	$2.6\ 10^{12}$
^{41}Ca	$1.6\ 10^{13}$	^{93m}Nb	$1.3\ 10^{10}$	^{154}Eu	$1.1\ 10^{14}$
^{54}Mn	$6.7\ 10^{9}$	^{94}Nb	$2.2\ 10^{5}$	^{155}Eu	$3.5\ 10^{13}$
^{55}Fe	$3.5\ 10^{14}$	^{99}Tc	$3.9\ 10^{9}$		
^{59}Ni	$3.1\ 10^{12}$	^{108m}Ag	$5.3\ 10^{11}$		

4.1.3.2. Engineering studies and cost estimates

BNFL has outlined a conceptual incinerator flowsheet. After hammer milling to pieces of about 2.5 cm size, the graphite is burnt with air at 1000°C. The off-gases are cooled with water spray to about 250°C and are then passed on to regenerable sintered stainless steel filters and, finally, to HEPA filters operating at 150°C. The ash (including spent filters) from 10 t of graphite, the daily throughput, is cemented into one 200 litre drum.

CEGB and UKAEA have studied concepts for the packaging and handling of irradiated graphite. The CEGB concept is based on the use of drums, in which case the use of returnable shielded overpacks appears attractive. The UKAEA concept is to incorporate graphite in a solid matrix within a reinforced concrete box of size 2.4 m x 2.2 m x 2.2 m.

Costs of the various operations have been estimated.

4.1.3.3. Radiological assessments

NRPB has carried out radiological assessments of the envisaged disposal options. Existing models have been adapted where necessary for use in this study, and data have been assembled on the environmental behaviour of activation product nuclides not previously modelled.

4.1.3.4. Leaching tests

In support of the radiological assessment, CEGB has carried out 100-day leaching tests on samples of irradiated graphite, including tests using seawater at the pressure and temperature of the deep ocean bed, and using simulated natural underground water.

Large-scale blocks of unirradiated graphite and small-scale blocks of irradiated graphite maintained their structural integrity after being subjected to the hydrostatic pressures likely to prevail at a deep ocean disposal site.

The leach rates of the following radionuclides from samples of irradiated graphite were measured: ^{14}C, ^{3}H, ^{134}Cs, ^{133}Ba, and ^{60}Co. With the exception of ^{60}Co, all showed comparable leaching behaviour in the four test environments considered.

For the radiologically significant nuclides ^{14}C and ^{3}H, the maximum observed incremental leach rates did not exceed 10^{-4} and 10^{-5} cm/day respectively, and fell during the course of the tests tending towards apparent equilibrium values. The loss in activity from the graphite samples during the 100-day test corresponds to 0.08% for ^{14}C and 0.3% for ^{3}H.

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

4.2.2.1. Laboratory, pilot plant and full-scale melts

The types of scrap material available to the contractor in quantities sufficient to perform this programme are as follows:

- carbon and stainless steel grades originating from heat exchangers of several CEGB Magnox reactors;
- carbon and stainless steel grades originating from contaminated reactor components probably coming from Windscale (AGR), Winfrith (SGHWR, Dragon HTR) and Calder Hall (Magnox);
- further it would be tried to include in this programme scrap representative of LWRs.

For each material type, the origin, operating and storage period and conditions, pre-treatment and alloy composition will be identified.

The activity partition trial melting will be done at various scales through small induction melts to full-scale melts. Slag and other practices being used will be chosen with regard to material type and product and to the slag seeking nature of any verified radioelements. The envisaged laboratory and pilot plant melting concerns the treatment of 5 to 1000 kg of contaminated/activated steel (carbon and stainless grades) by up to 20 melts mainly during 1982. The envisaged laboratory, pilot plant and full-scale melting concerns the treatment of amounts up to about 10 t of contaminated steel by a similar number of melts during 1983.

In all melting tests, samples of metal, slag and emission fume and dust will be taken. Metal samples will be cut from various parts of the product, to check the homogeneity of radionuclide distribution. Furnace bath samples will also be taken. Slag samples will be taken at several times during the steel melting/refining. Emission samples will be taken continuously during the process. The various samples will be measured for gamma and beta radioactivity contents. The contamination of the test equipment (e.g. furnace refractory) will be determined.

Balances will be established showing the material and radioactivity transferred to the ingot, the slag, the off-gas and test equipment. The balances will differentiate between the most relevant radionuclides found in the scrap material (e.g. cobalt-60, nickel-63, niobium-94) and those found on the scrap material (e.g. cesium-134/137, strontium-90).

4.2.2.2. Radiological safety aspects

The radiation exposure of the workers involved in the melting trials, especially those at larger scale, will be evaluated (individual and collective doses). Likewise, the radioactive emissions to the environment will be assessed using appropriate measurements. The contribution of the most relevant radionuclides to these exposures and emissions will be differentiated in order to enable predictions for varying radionuclide compositions in the scrap material.

4.2.2.3. Assessment of cost/benefit aspects

On the basis of the experimental results obtained, the best techniques for pre-treatment, handling, melting etc. at industrial scale will be specified, considering alternatively the recycling of the treated metal and its disposal as radioactive waste. Taking these techniques as a reference and making appropriate assumptions where necessary, the following aspects will be estimated:

- cost of the treatment (including the consequential costs, e.g. for slag disposal);
- salvage value or cost of disposing of the metal as waste, respectively;
- collective occupational radiation exposure;
- radioactive emissions to the environment.

These estimates will be compared with reference data for disposing of the scrap material without the melting treatment.

4.3. Immobilisation 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 varying thicknesses and to evaluate their effectiveness in immobilising 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 nonradioactive and on active specimens.

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

4.3.3. Progress and Results

Data on paints and coatings with a thermosetting resin base and relevant coating properties by commercial application techniques have been compiled and reviewed. The thermosetting paint process (epoxy or polyester) has been selected and applied by electrostatic projection.

An electrostatic projection device based on commercial equipment has been assembled in a 1.5 m^3 ventilated plastic cell. Preliminary tests have been carried out to optimise the application techniques with selected epoxy paints and commercial projection spray guns.

4.3.3.1. Tests with nonradioactive samples

These tests have been carried out in order to determine the quantity of paint lost to the filters. The results are given in Table 3.

Table 3. Tests with nonradioactive samples

Ventilation	Sample	Measured paint deposit (g)			Paint loss
		sample	receptacle	filter	(%)
high ventilation ($1500\ m^3/h$)	lid (60cm dia.)	230	741	19.25	2.0
	pipes and grids	160	121	15.15	5.1
low ventilation ($1000\ m^3/h$)	lid (60cm dia.)	180	748	8.10	0.9

4.3.3.2. Tests with contaminated samples

Sample plates of stainless steel and surface-oxidised mild steel have been contaminated with radioactive solutions and then coated with one or two layers of expoxy paint of 25-30 micron thickness. Gamma counting was performed on smears taken before and after the coating. The smear counts were found to be reduced by a factor of the order of 200 after one coat and of 10,000 after two coats.

4.3.3.3. Preparation of thick coatings

In order to determine the maximum achievable coating thicknesses, nonradioactive metal plates were coated by electrostatic powder projection. For each coating layer, the paint was applied during 20 s. Before the first coating, between successive coatings and after the last coating, the samples were heated for 10 min to 200°C. The total coating thickness obtained with one, two and three coats was 0.7, 1.4 and 1.8 mm, respectively.

4.4. Cobalt Removal from Steel by a Melting Process of the Electro-Slag Refining Type

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

Contract N°: DE-D-004-F Work Period: January 1981 - April 1983

4.4.1. Objective and Scope

The aim of this research is to assess by means of theoretical studies and tests under non-active 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 made it possible to recycle that metal.

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

4.4.3. Progress and Results

The feasibility of removing cobalt from stainless steel by a process of the type Electro-Slag-Refining has been studied. In this metal purification process, an electric current flowing between a consumable electrode, consisting of the metal to be purified, and a cooled baseplate is used to melt a slag bath. When the temperature of the slag bath exceeds the melting point of the metal, metal droplets fall through the slag, thereby being purified, and solidify on the baseplate. In order to transfer cobalt from the metal to the slag, an element is required which forms with cobalt stabler compounds than with the other steel constituents, and these compounds must be soluble in the calcium fluoride slag. Phosphor and sulphur have been considered as promising candidate elements for this purpose.

A literature research on thermodynamic data of phosphides and sulphides of the relevant elements has been made. Using these data, formation energies have been calculated which show that the stability of phosphides and sulphides increases in the order: nickel, iron, cobalt. Since the sulphide reactions involve low energies, phosphides have been chosen for experimental investigations.

The stability order indicated above has been experimentally confirmed for phosphides. Solubility tests (1520°C, 20 min) showed, however, very low solubilities of cobalt phosphides in calcium fluoride slags. The maximum cobalt content obtained in the slag was 0.8 wt%. After melting (1600 and 1700°C, 20 min) of cobalt doped (0.5 wt%) stainless steel in the presence of calcium fluoride slag, only a very small fraction (less than 0.4%) of the cobalt was transferred to the slag.

In the light of these negative results, work on the process studied so far has been stopped and investigation of an alternative separation process has been proposed. This process involves the introduction of an oxygenous gas flow into a steel/slag melt and is 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 used for the production of stainless steel.

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

4.5.2.1. Analysis of existing knowledge

Both a literature survey and a consideration of existing methods of dust control will be made. Following this survey an appraisal of suitable alternatives to sodium silicate will be prepared. The following is a list of possible candidate materials:

- alkyl silicates which hydrolyse with moisture to form a silicic acid gel;
- starch or cellulose ethers which are simple film formers, convenient but weak;
- casein which forms insoluble products with calcium;
- water soluble urea formaldehyde resin solutions which are used for textile treatment, or phenol-formaldehyde resin solutions which are used in soil consolidation and both of which become insoluble with heat or catalyst;
- natural and synthetic rubber lattices, e.g. styrene-butadiene;
- bitumen in solution or emulsion form;
- epoxy or urethane, single or two pack systems, both in solvent and emulsion form.

4.5.2.2. Laboratory evaluation of materials

Materials will be considered for evaluation in the following order, which is based on estimated cost and convenience of use:

- sodium silicate;
- other water dispersed materials;
- single pack organic solvent dispersed materials;
- two pack solvent dispersed materials.

Coating processes: Experiments will be made to determine the best method of insolubilising sodium silicate (in the case of most of the other suggested materials the film forming process is more obvious). Experiments will be made to impregnate and thereby fix layers of dust on small pieces of concrete for convenient laboratory handling and to impregnate dust in bulk in order to form it into blocks.

Mechanical resistance of the coatings: Studies then will be made of the stability of the dust layers and of the strength of the impregnated dust blocks. Methods of test will be developed but it is suggested that in the case of dust-coated concrete lumps, these would be subjected to air blast and to tumbling in a barrel; for impregnated blocks of dust, measurements may be made of impact strength and again air blast and barrel tumbling. In both cases, the material removed will be analysed for calcium as a measure of the concrete dust removed.

Weathering resistance of the coatings: The resistance of test specimens to simulated weather using alternate cycles of high and low humidity at different temperatures (freezing included) will be determined.

Radiation resistance of the coatings: Studies will be made of the effects of radiation on the coatings but it is probable that sufficient data already exists to make experimental work unnecessary.

Wetting by coating solutions: Studies will be made of methods to secure complete wetting and penetration and consideration given to pre-impregnation treatment if necessary.

Compatibility of the coating with embedment materials: Consideration will be given to the physical and chemical reactions of the impregnating material with that material, probably cement, to be used for encapsulating the rubble. It is probable that sufficient knowledge exists to predict any direct chemical interaction, but consideration must be given to the possibility that degradation products, derived from any material used by the action of radiation, may be detrimental to the encapsulating material.

4.5.2.3. Full-scale aspects

Studies will be made to translate the results of the laboratory work to the practical situation of demolition of an actual structure and consideration will be given to the influence of the main types of demolition (i.e. impact, explosive, saw cutting) on the nature of the rubble and dust produced.

Studies will be made of the methods by which dust can be impregnated and the dusty rubble can be coated without handling (remote application).

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 - June 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

Due to internal re-organisation problems the research will only be started in January 1982.

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 - August 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 (Pressurised Water Reactors, Boiling Water Reactors and Gas-cooled Reactors).

5.1.2. Work Programme

The first phase of the work involves the collection of data on the waste produced from a series of reference power plants and a survey of existing containers.

This will provide the entry into an iterative sequence, the output of which will be the definition of possible types of package worthy of more detailed investigation. The output from this second phase will be:

- conceptual designs of large transport and/or disposal containers;
- conceptual transport systems and procedures;

- estimates of the cost and radiological detriment associated with the various procedures for transport.

5.1.2.1. Collection of data and activity analysis

Data will be collected with regard to:

- regulations and legislation;
- range and suitability of existing containers;
- system constraints (in particular: constraints due to normal transport facilities and possibly due to loading within the reactor buildings).

The data will be evaluated on the waste arising from dismantling of reference power plants chosen to represent the major types. Data on the activity of the major reactor components will be collected or calculated to obtain the shielding requirements of the containers. Prompt and delayed dismantling will be considered as regards radioactivity levels and dose rates.

5.1.2.2. Definition of possible conceptual designs

The data from 5.1.2.1. will be subjected to detailed scrutiny in order to evolve possible conceptual designs of large transport and/or disposal containers. These designs will take account of dimensions and weight of reactor components. They will also be examined in relation to engineering feasibility and cost.

5.1.2.3. Definition of feasible conceptual transport procedures

Feasible procedures for transporting the defined containers to sea or land, disposal or storage will be established. The procedures to be identified will be subjected particularly to the following points:

- Means of transportation in normal conditions:
  road, rail, inland waterway, sea;
- Regulations:
  requirements of IAEA Safety Series 6;
  competent authority regulations and approvals;
  road, rail, port, waterway and maritime regulations;
  requirements of the London Convention of 1972, concerning sea dumping;

requirements for qualification programme, quality assurance and notional test sequence.

5.1.2.4. Assessment of cost and radiological detriment of the procedures

The radiological detriment associated with the various recommended procedures for transport and disposal will be assessed to identify both the total radiological detriment and the important operations from a radiological standpoint. The radiological detriment will then be converted into financial terms so that the overall economics of the procedures can be assessed.

5.1.3. Progress and Results

5.1.3.1. Collection of data on Gas-cooled Reactors (UKAEA)

The purpose of the first phase of the study has been to introduce and list the main types of nuclear reactor in the European Community, select the reference plants for further study, classify the large gas-cooled reactor plants relative to the reference and indicate the methodology to be used in assessing the radioactive inventories anticipated during decommissioning of those reactors which are relevant to this study. Although work by the UKAEA and its sub-contractor Pollution Prevention (Consultants) Ltd has been concentrated on gas-cooled reactors in the United Kingdom, contact has been made with Electricité de France to obtain similar data on French gas-cooled reactors. Two reference stations have been chosen as representatives of the 50 gas-cooled reactors in the European Community, i.e.:

- UK Magnox reactor with steel pressure vessel;
- UK Advanced Gas-cooled Reactor with prestressed concrete pressure vessel.

Besides, data have been collected on the Trojan 1175 MWe PWR because of the availability of a very detailed decommissioning study based on this plant, made by Battelle Pacific Northwest Laboratory.

5.1.3.2. Collection of data on Light Water Reactors (Transnubel)

The work started with the collection of data required of the four reference power stations, i.e.: German KWU PWR, French EdF/Framatome PWR, Belgian PWR and German BWR.

Meetings were organised with the Belgian operators and Transnuklear who had already been involved in previous studies on the optimisation of large transport containers. Detailed data on the composition of the alloys employed (cobalt and trace element content) were generally not available. Belgonucléaire S.A. was also involved in the study of documents on the decommissioning of PWR and BWR plants and related transport problems (American and German optimisation studies, reviews and analyses of current regulations).

A first survey of the documentation on existing containers available from Belgonucléaire, CEN-Mol, SEMO (Tihange 1 power plant), CEA-France, as well as within the group Transnucléaire-Transnuklear-Transnubel, has been performed.

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

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

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

An additional call for proposals concerning the trace elements in the concrete of biological shields was issued in October 1981.

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 - June 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 are related to those important for the mechanical and thermal behaviour of the material and, correspondingly, precision is not high enough for detailled computations of activation. Consequently, it is necessary to include pessimistic assumptions into activation calculations, leading to overestimation 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, linked to elaborate neutron flux and activation computations with known operational characteristics of the reactor, will allow a better definition of the interface between activated and non-active concrete.

6.1.2. Work Programme

6.1.2.1. Specifications

In preparation of the measurements, detailled specifications on the locations for sampling and on the nuclides to be determined will be established.

6.1.2.2. Sampling

By horizontal core drilling at various levels of the reactor's biological shield, samples will be taken to a depth as high as technologically possible.

6.1.2.3. Radiological analysis of the concrete samples

From the cores, disks will be cut, which are carefully identified as to their original position in the shield. Each disk will be analysed for beta and gamma emitting isotopes with special attention to those with long half-life. For identified radioisotopes, the specific activity will be determined.

6.1.2.4. Chemical analysis of samples

Part of the samples will undergo chemical analysis with special attention to elements, including trace elements, relevant for neutron activation.

6.1.2.5. Determination of neutron fluxes

The existing calculations and measurements of neutron fluxes will be checked and, if necessary, updated. After having performed the chemical and radiological measurements, a computation of fluences will allow a cross-check based on the measured activation.

6.1.2.6. Evaluation of the results

By combining measurements and computations, it will be possible to define, for each location inside the biological shield, the time to reach a limit value which forbids unrestricted release of the material. The validity of data to be introduced in computer programmes will be checked and, possibly, isotopes will be determined which have not been adequately taken into account in computations.

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 - June 1982

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

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 - December 1982

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; in order to reduce uncertainties, measurements to values below the ppm-level are required.

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

6.3.2.1. Preliminary research

As a preliminary step, all existing data on the steel of the vessel and the internals and all information, calculations as well as measurements, on neutron fluxes and fluences will be assembled.

6.3.2.2 Analysis of samples

The available samples, which have been irradiated with the vessel, will be analysed carefully to determine the specific gamma activity and the cobalt and niobium content. The measured activation will be correlated to the physical location of the samples in the vessel.

6.3.2.3. Activation calculation

Taking into account the measured material composition and given fluxes and operation time, a detailled activation analysis by means of a computer code will be performed. This includes comparison of computed and measured activation for the sample locations, determination of the activation of all parts of the vessel and computation of the decay of activity.

6.3.2.4. Evaluation of the dismantling techniques

The methods for dismantling and for protection of workers which will be needed during interventions will be evaluated taking decay times from 2 to 100 years as a parameter.

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. For this reason and with regard to disposal of the activated parts, better knowledge of radionuclides present in the vessels and internals of reactors after the end of commercial operation is needed.

The aim of the research is to carry out trace element assessment of low alloy steels (LWR vessel) and stainless steels (LWR vessel lining and internals), with particular reference to gamma sources important at long decay times.

6.4.2. Work Programme

The most important trace elements will be analysed. Particular attention will be paid to the trace elements Nb, Co, Eu, Sm, Ho and Ag and to the more major constituents Mo and Ni. Both activation analysis and atomic absorption spectrometry will be carried out to ensure confidence in results obtained and to ensure results even if concentrations are at ppm or sub-ppm levels. The aim will be to study the variability of trace element levels and not simply to establish upper limits.

6.4.2.1. Selection of steels

The reference materials for the study will be low alloy steels conforming to A553B - that is current vessel steel for a light water reactor - and 18/8/1 stainless steel typical of vessel lining material. However, those reactors which will be due for decommissioning first, have vessels not conforming to A553B and this should be reflected in the selection of samples.

The analysis of 35 samples of low alloy and 15 of stainless steel is envisaged. Provision of samples could be arranged in consultation with interested parties in Community member countries.

6.4.2.2. Atomic Absorption Spectrometry

Atomic absorption spectrometry will be carried out by the Imperial College Analytical Services Laboratory in the Metallurgy Department. The flame method will be used for Ni, Mo and Co, while the electrothermal atomisation method which is capable of sub-ppm level will be used for measurements Eu, Sm and Ag.

6.4.2.3. Activation analysis

Activation analysis will be carried out at the University of London Reactor Centre. It is envisaged that irradiations of samples in the form of coupons of a few grams weight will be carried out in the Core Tube Facility and counting with Ge-(Li) detectors be carried out after an appropriate decay period. Cobalt wire will be used as an irradiation monitor. Radiochemical separation of cobalt will be carried out for selected samples if necessary to increase the sensitivity to other activities. In the case of such selected samples a systematic search would be made for all long-lived activities. Supporting activation analysis work would also be carried out in the Nuclear Technology Laboratories of the Chemical Engineering Department.

6.4.2.4. Activation calculations

Calculation of long-lived gamma activities would be carried out by the Nuclear Power Section. The build-up and decay codes ORIGEN will be used with two activation libraries due to Kee and to Jarvis of AERE Harwell. The data of the latter libraries are divided into 100 energy groups and include some 1,010 reactions for isotopes from hydrogen to bismuth. Cross sections have been obtained from various US, UK, French and Australian data files and supplemented by theoretical work. For selected sample compositions, long-lived gamma activities will be assessed for representative LWR structure fluxes, spectra and irradiation times.

6.4.2.5. Evaluation of results

The results obtained will be used to compute, for a typical 1200 MWe LWR having been operated during its full service-life, the evolution of gamma activity at typical locations as a function of decay time. Attention will be drawn to trace elements which should be strictly limited in the steels employed for vessels and internals.

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 minimising the activation of concrete;

e) protection of concrete against contamination;

f) improved documentation system.

Concerning items a, d and f, which are not yet adequately covered by contracts, additional research proposals were called for in October 1982.

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 minimising 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

In a first phase, the task is to produce a preliminary decommissioning proposal identifying design features to facilitate decommissioning which will include:

- access proposals, installed features, removable shielding;
- health physics provisions (that is change rooms and outline ventilation requirements);
- control of isotopic composition of materials used in reactor construction to reduce long-term radioactivity.

During a second phase and following finalisation of the material specification, an interim report will be produced which will include an estimate of the inventory of induced radioactivity in the fixed reactor structure and components at ten years and one hundred years after the reactor has been shut down. More detail will be available to supplement the preliminary report.

At the end of a third phase a final report will be issued covering the total decommissioning proposal as agreed with the South of Scotland Electricity Board, the Central Electricity Generating Board and the Licensing Authority. This will include total provision for decommissioning including classification and application of radioactive inventory, estimate of anticipated radioactive dose, handling routes and the factors influencing the selection of the decommissioning alternatives.

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 design of steel and concrete structures of nuclear power plants has not hitherto taken account of the demolition of the structures which is required after the end of their service life. Thus the treatment of existing structures will require the use of difficult and expensive processes which could, in part, have been avoided if suitable facilities had been built in during the original design. Little previous study has been made of design provisions which could be made for this purpose.

Concrete structures have high durability and it is possible that refurbishing of reactor systems or processing plants with new and/or up-dated components could be a viable alternative to new construction. Refurbishing has been considered so far only in relation to the routine renewal of plant and there does not appear to have been any significant study of the possibilities of extensive refurbishment or the structural features required to facilitate this.

The main purpose of the research is to study the designs of pre-stressed concrete pressure vessels, containment, shielding and general structures within the contaminated areas of nuclear power stations (gas-cooled and light water reactors) to identify features which inhibit demolition and refurbishing.

The study would assess the identified problem area, categorise features to be incorporated in future designs to ease access, dismantling, demolition, refurbishing and renewal of the plant, and where appropriate, the up-rating of its design study.

7.2.2. Work Programme

7.2.2.1. Literature study

A literature study will be made to identify the principal forms of existing vessels, containments, shielding and conventional structures used in nuclear industry and the main problems posed by refurbishing, renewal, dismantling or demolition.

7.2.2.2. Definition of favourable design features

Design features which would aid the above activities including the adoption of forms of construction comprising discrete separable elements, the incorporation of break planes, planned provisions for explosives jacking, etc. will be defined.

7.2.2.3. Measures facilitating the removal of active material

An examination of measures to facilitate the removal of active material including surface treatment, choice of surface materials, barriers, break-away surfaces, inter-face membranes, etc. is effected.

7.2.2.4. Assessment of effects on operation performances and costs

For typical forms of structure, the effects of the features defined before on design criteria, functional requirements, operational safety, reliable durability and other pertinent matters will be assessed.

7.2.2.5. Modifications in design

A study of modifications in design which would be necessary to embody provisions for refurbishing, or for renewal of deteriorated structural components or the dismantling or demolition of typical structures will be effected.

7.2.2.6. Identification and cost-benefit analysis of the most promising features

An identification of concepts, policies and practical design embodiments which offer most promise and consideration of the cost benefit of these will be made.

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 - October 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

7.3.2.1. Materials to be tested

Samples will be prepared of the following materials (trade marks):

- reference material: Stellite 6;
- substitutes deposited by welding: Pantamax 25/0 Mo-G, Alloy E3, Tribaloy T700, Nitronic 60;
- substitute deposited by detonation: LC-1C;
- substitutes used as base materials: Nitronic 60; 17-4 pH.

7.3.2.2. Tests and measurements

The tests will be performed in an existing installation enabling simultaneous testing of eight samples at pressures up to 15 bars, temperatures up to 150°C and flow velocities up to 100 m/s. The main part of the installation is a motor driven rotor into which the coolant enters at the center, flowing outward. The samples are fixed on the casing, very near to the outer diameter of the rotor. They are electrically isolated against the casing in order to enable electrochemical measurements.

The tests will be performed at 150°C and flow velocities of 40 and 80 m/s with borated water as used as primary coolant in pressurised water reactors.

The following sample characteristics will be measured before and after the tests:

- macro-hardness and, possibly, micro-hardness;
- weight (precision: 0.1 mg);
- surface profile.

Metallographic examinations and measurements of mechanical properites have already been performed.

7.3.2.3. Evaluation

The results obtained will be extrapolated to the operating conditions and time of a light water reactor. A systematic correlation analysis of the erosion-corrosion behaviour and material characteristics will be performed.

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 - September 1982

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-ray 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 non-experimental study will be 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 to define the present upper limits, and of data on the effective cobalt and niobium content existant in nuclear grades of steel;
- evaluation of the possibility to reduce the cobalt and niobium impurity content by using low cobalt ores and feedstocks, taking into account the availability of the ores and the influence on costs;
- evaluation of the possibilities to reduce the cobalt and niobium impurity content by special treatments (if appropriate) from the raw materials (assessment of techniques including cost estimates);
- a 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 will be carried out, considering thermal reactor and fast reactor fluxes.

7.4.2. Work Programme

7.4.2.1. Quantitative study of the origin of cobalt and niobium content

As a first step, the origin of the cobalt and niobium content in typical nuclear quality steels and alloys will be studied quantitatively. This will include a paper study of the ore and the final metal production. From this, the areas where most of the cobalt and niobium arises will be pinpointed and remedial action will be suggested.

7.4.2.2. Non-experimental study of the process parameters

The experiments will be identified, which are needed to prove that the theoretical assessments are on a firm foundation.

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 - September 1982

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 coating 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

7.5.2.1. Collection and critical analysis of existing knowledge

The collection of knowledge on the behaviour of protective coatings on walls and floors of nuclear reactors and other installations, e.g. hot cells, will be the first step. Characteristical values like mechanical resistance, aging etc. will be assembled. The existing experimental data (from national and international research centres), the conclusions of the ad hoc working group UNICEN and the experience with coatings on walls and floors of nuclear installations will be carefully analysed.

7.5.2.2. Inventory of concrete surfaces

Taking the Caorso nuclear power plant as a reference, an inventory of concrete surfaces for which protective coatings would be useful, will be drawn up. These surfaces will be categorised following criteria such as mechanical load, corrosive agents coming in contact with the surfaces etc.

7.5.2.3. Experiments

Existing coatings will be studied with special attention to those which are easily removable. Tests at room temperature will be made, applying the coatings by various procedures to concrete surfaces of various geometries and surface conditions. On concrete samples coated with one or several layers, tests on shock resistance and wear will be performed.

7.5.2.4. Economic evaluation

An evaluation of the economical aspects by cost-benefit analysis will be made for the most promising coating.

7.6. Characterisation 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 a number of 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-time protection against the penetraction of contamination into the base material. An ideal coating should be:

- easy to apply and inexpensive;
- tight, adherent, resistant to chemical attack and to aging;
- easily to be decontaminated 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 aim of this study.

7.6.2. Work Programme

7.6.2.1. Review of existing experience and selection of the most promising proctection systems

Existing product and protection systems for concrete surfaces will be examined, taking into account data from producers and from users (Commissariat à l'Energie Atomique, Electricité de France and other organisations). Existing information on the following aspects will be assembled:

- method of application;
- adherence to surface;
- conformity with safety rules concerning toxicity, explosivity, inflammability etc.;
- tightness;
- resistance to chemically aggressive agents and behaviour at elevated temeratures;
- decontamination capabilities;
- possibilities of removal from the base material;
- cost per surface unit.

An inventory will be drawn up of the concrete surfaces of a nuclear power plant for which a protective coating would be useful. This inventory will be used as a basis to determine the mechanical loads and other conditions, like humidity, and to define a limited number of conditions and environments to be applied in the further investigations.

7.6.2.2. Experiments

The selected coatings will undergo a series of investigations, in particular:

- analysis of the composition of the coating;
- tests of the possible methods of application and influence on costs per surface unit;
- measurement of adherence on concrete;
- porosity measurements by means of special instrumentation;
- tests of shock and abrasion resistance;

- corrosion resistance to various chemical solutions in neutral, acid or basic environment;
- tests of resistance to acid mist and to accelerated aging;
- decontamination measurements;
- measurement of the influence of temperature;
- tests for removing the coating.

7.6.2.3. Evaluation of the results

The measurements of the characteristics of the selected products and the knowledge of the loading of chemical or physical origin in the nuclear power plant will be used to draw up specifications for coatings. For each product will be indicated in particular:

- the limits of application (temperature, type of surface, mechanical loads or shocks, ets.)
- the cost per surface unit including cost for removal of the coating and, if necessary, conditioning as radioactive waste.

8. IDENTIFICATION OF GUIDING PRINCIPLES

Available material in the Member States that could serve as a basis for guiding principles in the field of decommissioning has been assembled. It appears that the existing systems of authorisation and control, together with the radiological protection standards in force, already make it possible to decommission nuclear power plants on a case-by-case basis. However, no specific technical regulations on decommissioning exist. A basic uncertainty as regards final decommissioning, i.e. release of sites for re-use, lies in the absence of criteria for distinguishing non-radioactive material from radioactive material.

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2. Council Decision of 27 March 1979 adopting a research programme concerning the decommissioning of nuclear power plants (79/344/Euratom). OJ N° L 83, 3.4.1979, p. 19.

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