Commission of the European Communities

nuclear science and technology

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

Report

EUR 7440 EN

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(First annual progress report (year 1980))

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FOREWORD

This is the first progress report of the European Community's programme (1979-1983) of research on the decommissioning of nuclear power plants. It shows the status of the programme on 31 December 1980.

The Council of the European Communities has adopted the programme in March 1979*, 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 No. 1 : Long-term integrity of buildings and systems ;
- Project No. 2 : Decontamination for decommissioning purposes ;
- Project No. 3 : Dismantling techniques ;
- Project No. 4 : Treatment of specific waste materials : steel, concrete and graphite ;
- Project No. 5 : Large transport containers for radioactive waste produced in the dismantling of nuclear power plants ;
- Project No. 6 : Estimation of the quantities of radioactive wastes arising from decommissioning of nuclear power plants in the Community ;
- Project No. 7 : Influence of nuclear power plant design features on decommissioning.

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* O J No. L 83, 3.4.1979, p. 19

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 (indirect action programme). 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*.

After the programme was adopted by the Council on 27 March 1979, the Advisory Committee and the programme staff of the Commission had first to be appointed, since this is the first Community programme in the field of the decommissioning of nuclear power plants ; this was achieved by the end of 1979.

Subsequently, the work to be undertaken has been defined in detail and research proposals have been called for and carefully examined and selected, project by project. The Commission staff in charge of the programme during 1980 were : B. Huber**, K.H. Schaller, R. Bisci, K. Pflugrad.

This first progress report covers therefore the period of putting the Council decision into action and of initiating the research. It describes the nature and scope of the work being carried out under the various contracts and shows the orientation of the research projects ; first results will be reported in the next progress report.

B. Huber
Programme Head

S. Orlowski
Head, Division
"Fuel cycle, decommissioning of nuclear installations, and conventional and nuclear power systems"

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* See Annex

** Part-time

1. PROJECT NO. 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
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 maintanance. 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

1.1.2.1 Review of C.E.G.B.'s plants

A review of the state of the contractor's nuclear power station buildings and plant, to identify the mode and extent of degradation of the various materials as they exist, will be carried out. This review includes a systematic regular sampling of concrete, steel and protective material taken from :

- shields, tanks, building and services ;
- structure steelwork including plant supports, lifting devices and personal access ways ;
- superstructures, roofing, cladding and weather protection.

Measurements will be made to establish loss of strength or function by corrosion, erosion, work hardening and other relevant environmental factors.

1.1.2.2. Extrapolation and interpretation of results

The test results will be used to predict the consequence of degradation for up to 100 years subsequent to the normal service life.

1.2. Long term integrity of buildings and systems

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

Work period : January 1981 - December 1983

1.2.1. Objective and Scope

The aim of this study is to improve the knowledge of the ageing 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. The synthesis will present itself in the form of a methodology document. It will group, for a given type of reactor and for the constituent parts of this reactor:

- the parameters which are the cause of aging and their influence on the conservation of the barriers and on the maintaining in condition of the equipment kept in service ;
- the means employed to mitigate or control the consequences of aging.

The study, as defined above, will have as basic structure the constituent parts of a pressurized water reactor. The particular parts of other reactor types will be the subject of complementary lateral structures which will be studied, if necessary, in regard of their specific features.

1.2.2. Work Programme

1.2.2.1. Study of plant aging and maintenance in a satisfactory condition

This study will use information from the Commissariat à l'Energie Atomique and Electricité de France as regards nuclear plants and from the Department of Public Works and construction firms as regards buildings, civil engineering structures and steel structures.

a) Catalogue of information on plant aging

The following information will be gathered :

- Buildings and civil engineering structures :

  Information with due regard to climatic, atmospheric and geological (earthquake) conditions on deformations and stresses, leak-tightness, corrosion resistance and maintenance of amenity.

- Reactors and associated systems :

  Information on wet and dry corrosion resistance, leak-tightness, seal hardening and wear of moving parts.

- Ancillary power station installations and equipment :

  Information on changes over time in the condition of installations and equipment such as :
  - electricity supply and electrical equipment ;
  - radiation protection installations and monitoring and control equipment ;
  - ventilation and gas filtration equipment ;
  - hoisting and handling devices ;
  - water purification and waste water treatment plants and effluent collection and treatment facilities.

b) Study of specific aspects

The following aspects will be studied :
- changes in radioactivity in the plant ;
- the conditions for maintaining the buildings, civil engineering structures, reactors and associated systems in a safe state ;
- the conditions for maintaining the ancillary installations and equipment operable.

c) Utilization of the results obtained

The results obtained will be utilized in the following manner :

- curves defining the changes in materials and equipment with time will be plotted ;
- monitoring, maintenance and operating procedures will be drawn up ;
- the operations required in order to attenuate aging effects will be defined ;
- the financial and radiological consequences (in particular in terms of personnel exposure) of the operations referred to in the two preceding subsections will be estimated as a function of time.

1.2.2.2. Assessment of the possible life of the plant and its components with due regard to the costs

The assessment will be made for Stages 1 and 2 of decommissioning as defined by the International Atomic Energy Agency.

a) Stage 1 - Maintenance of the plant in a satisfactory condition

A list of equipment to be maintained in service or in working order will be drawn up and the resources required for operating and maintaining it will be assessed. The monitoring, supervision and technical inspection operations required for ensuring that the other equipment is maintained in a safe condition will be studied. The periods of time over which both categories of equipment can be maintained in a satisfactory condition at the lowest cost will be determined.

b) Stage 2 - Long-term containment

This part of the research will determine the modifications to be made to the plant in order to decommission it to Stage 2 and the monitoring, supervision and inspection operations required for the equipment remaining on site or in service. The maximum containment period will be assessed.

1.2.2.3. Assessment of the consequences of deferred dismantling

The following consequences of deferred dismantling will be assessed :

- technical consequences ;
- additional cost incurred in maintaining the plant in a safe condition

- concerning the radiological impact the radiological parameters will be specified which are to be included in the basic data necessary to the safety study made at the moment of the final close-down of a given reactor, taking account of the site concerned.

1.2.2.4. Proposals for further work

Measurements and tests that could subsequently be carried out in order to complete the knowledge of the aging process will be proposed.

2. PROJECT NO. 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;
- mechanical techniques (high-pressure water lance, erosion by cavitation);
- decontamination of concrete walls by flame-spraying.

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, will also be investigated.

2.1 Delegation of an expert to the USNRC* in relation to the cleanup of the TMI-2 plant

Contractor: Comitato Nazionale per l'Energia Nucleare, Rome, Italy
Work period: October 1980 - April 1981

2.1.1 Objective and Scope

The decommissioning of a nuclear power plant, in which a major loss of coolant accident has occurred, would need decontamination procedures different from those required in the case of a power station which has operated under normal conditions, because:

- the degree and the radioelement composition of the contamination are different;
- the activity may be spread over places where access is difficult and which are normally inactive;
- large quantities of radioactive liquids and gases have to be treated.

*United States Nuclear Regulatory Commission

Reactor N° 2 of the Three Miles Island station (United States) offers valuable information on the contamination of a nuclear power station which has experienced 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

The contractor sends an experienced engineer to the USNRC to participate in the study and analysis of the proposals and work programme put forward by the owner of the TMI-2 plant concerning the cleanup. The areas to be covered and to be reported on more specifically are:

- measurements of the activity in the plant and surrounding area;
- the radiological impact of cleanup operations on workers and the general public;
- the methods and procedures for cleanup;
- the treatment, transport and storage or disposal of radioactive waste;
- the standards and proposed standards applying to the cleanup and the radioactive waste management.

2.2. Decontamination of concrete surfaces by flame-spraying

Contractor: Salzgitter A.G., Salzgitter, Germany
Work Period: October 1980 - June 1982

2.2.1 Objective and Scope

When nuclear power stations which have been closed down are being dismantled, the contaminated layers of the building must be removed. This involves the treatment of large areas of concrete walls that may have been coated with decontaminable paint.

Flame-spraying has been employed for years for cleaning concrete surfaces. Complete removal of old coatings, sediments, oil stains and already detached concrete is possible. This technique involves the use of an oxyacetylene flame, the heat of which causes the concrete layers to peel off and burns up old paint layers and coatings. However, no application of flame-spraying for the decontamination of concrete surfaces is known at the present time. The advantage of this technique lies in the simple and light structure of the equipment.

The purpose of the research is to investigate the efficiency and limitations of the flame-spraying for decontamination by testing the technique on non-contaminated and contaminated concrete surfaces with and without decontaminable paint. The investigation should give information on fire hazard, aerosol formation and filtering, radiological protection of the personnel, feasibility of directly exhausting the combustion products to prevent recontamination and magnitude of the decontamination factor.

2.2.2 Work Programme

2.2.2.1 Conversion and preparation of the test station

The experiments will be carried out in an existing test station, which consists of a work cabin with a ventilation system providing a high rate of air-exchange. Air cleaning will be performed in an existing filter station, which allows to determine the optimum filter combination. The burner, mounted on a travelling welding-machine, is adjustable for its angle. Gas-supply and a flame-proof device for exhaustion of the combustion products complete the installation.

2.2.2.2 Tests on non-contaminated surfaces

Concrete specimens produced on specifications for reactor buildings with and without decontaminable paint will be used in experiments carried out to determine the performance data and the optimum parameters relating to the burner adjustment and choice of filters. The following parameters will be systematically varied in order to identify the optimal conditions:

- the distance between the burner and the concrete surface;
- the angle between the burner and the surface;
- the rate of advance.

Horizontal floors, wall, slanting surfaces and ceilings will be tested with various combustion gases and the amount of material removed will be determined.

For characteristic and optimum combinations of the experimental parameters, the most suitable filter combinations will be employed and the following measurements will be performed in addition to that of surface layer removal:

- aerosol and dust sampling with subsequent particle-size analysis;
- determination of the filter efficiency;
- particle-size analysis of the surface-layer material removed;
- determination of the surface properties (roughness).

2.2.2.3 Tests on contaminated surfaces

Specimens will be artificially contaminated so that penetration depth and distribution typical of reactor conditions after many years of operation are simulated. The samples with decontaminable paint will be contaminated by spraying with a solution of activated cobalt steel in dilute nitric acid, dried and then decontaminated. Mechanical stress will be applied, accompanied by wear and tear. The stressing of the floor by placing sharp edged objects on it will be simulated by a well-defined impact test involving specific angles of impact.

The specimens without decontaminable paint will only be sprayed with a solution containing radioactive cobalt.

The main measurements will be:

- measurement of the surface activities after each experimental operation;
- breakdown of the activity removed (air sampling for aerosol and dust analysis, precipitation on walls, filters, exhaust system, etc.);
- determination of the filter efficiency.

2.2.2.4 Evaluation

The main purposes of the evaluation is to assess surface-layer removal efficiency in relation to the operating costs. The performance will be compared on the basis of the cost per unit volume of the concrete removed, the type of removal and the hazard to the personnel.

The results of the work on contaminated specimens will be extrapolated to the dismantling of a typical nuclear power station as regards contamination of the environment, filters, the exhaust system, radiological protection measures, radiation dose to workers, etc.

2.3 Erosion of metal surfaces by cavitation at very high velocity

Contractor: Alsthom-Atlantique Neyrtec, Grenoble, France
Work period: October 1980 - September 1981.

2.3.1 Objective and Scope

Cavitation is produced by localized steam formation in a liquid as a result of high flow rates.

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

2.3.2.1 Hydraulic design of the cavitating element

The study of the design will be carried out on an existing cavitation tunnel under lower-speed conditions than those employed in the case of the prototype equipment.

The tests cover:

- the creation of a cavitation generating a very high number of aggressive cavities;
- the production, downstream of the cavitating configuration, of a high pressure gradient enabling the collapse of these cavities and, consequently, the impact speeds to be accelerated.

In order to perfect the design fast-cinematography techniques and existing high frequency acoustic sensors will be employed.

2.3.2.2 Design and testing of a prototype cavitating element

A cavitating element, suitable for cleaning pipework, will be designed and constructed. The implementation of the system requires the adaptation of a specific circuit involving, in particular, a high-pressure tank partly filled with pressurized water under an air cushion.

This equipment will be employed for carrying out systematic high-speed (90 m/s approx.) erosion tests on various materials, mainly austentic stainless steel, used as cladding in the water-reactor pipework. The erosive power of the system will be determined by weighing and by microscope analysis.

2.3.3.3 Assessment of the process

On the basis of the experimental results obtained, the feasibility of the process for application in the decontamination of nuclear power plants, will be assessed.

If the results are favourable, the hydraulic dimensioning of full-scale equipment suitable for applications in pipework and on flat surfaces of reactor dimensions, will be done.

2.4 Composition of contamination layers and efficiency of decontamination

Contractor: Kernkraftwerk Lingen GmbH, Lingen, Germany

Work period: January 1981 - June 1983

2.4.1 Objective and Scope

A primary prerequisite for the development of decontamination methods, including very aggressive methods, is knowledge of the structure of the contamination layers; in this connection, special regard must be paid to the thickness and distribution of the layers in the loops and the penetration depth of the individual radionuclides.

The purpose of this research is the characterization of, on the one hand, the contamination layers formed in long-term reactor operation and, on the other hand, of the residual contamination after the application of aggressive chemical techniques. These investigations will be 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

2.4.2.1 Sample preparation

Various pipe sections will be taken from the primary circuit of the Lingen reactor. The sections will be divided into segments about 15 cm long and bisected. From these segments samples suitable for the various examination methods will be prepared.

2.4.2.2 Examination methods

The following examination methods will be employed as suitable:

- light-microscope examination, scanning-electron-microscope examination and energy-dispersive X-ray microanalysis of material surfaces and polished sections;
- secondary-ion mass spectrometry;
- gamma spectrometry for radioactivity determination;
- chemical analysis.

2.4.2.3 Determination of the depth profile of radioactive concentration

A known electrolytical technique of layer-by-layer removal of the base material will be employed. This technique will be optimized in preliminary tests using non-active specimens, in order to achieve the removal of layers of as uniform a thickness as possible.

On active samples the following work will be performed:

- removal of the upper ferrous-oxide deposition layer by alkaline oxidation with subsequent organic acid post-treatment;
- removal of the Cr_2O_3 layer using a strongly alkaline oxidizing solution;
- layer-by-layer removal of the base material using the beforementioned electrolytical technique and determination of the depth profile of radioisotope concentration; the number of removal operations will depend on the penetration depth of the radioisotopes.

2.4.2.4 Decontamination experiments with highly aggressive chemical solutions

Preliminary tests will be performed using non-active specimens of the same material quality as the contaminated samples. The specimens will be treated with aggressive chemical solutions of various concentration and the rate of material removal, the surface roughness and the tendency to pitting corrosion will be determined. The thickness of the base material layer removed must at least equal the penetration depth of the radioisotopes.

The active samples will be decontaminated using the optimum chemical solution obtained previously. The decontamination efficiency and the extent of recontamination of the sample surface by the solution will be determined.

2.4.2.5 Evaluation

The measurement results will be supplemented by comprehensive data on the water chemistry at the sampling points during the period of operation and on other relevant operating parameters such as pressure, temperature and exposure time. From these values, it will be tried to derive the penetration depth and composition of radioisotopes to be expected in a boiling water reactor of the 1 200 MWe type after the full period of operation. The results of the decontamination experiments will be evaluated in such a way that they can possibly be extrapolated to a 1 200 MWe boiling water reactor. The suitability of various techniques of complete decontamination (for the release or re-use of the material) and the amount of secondary waste arisings will be assessed.

2.5 Vigorous decontamination tests of steel samples in a special test loop

Contractor: Ente Nazionale per l'Energia Elettrica, Rome, Italy
Work period: October 1980 - September 1983

2.5.1 Objective and Scope

The experience of decontaminating reactor circuits and components using a circulating chemical solution is so far limited to mild and moderately aggressive agents. For decommissioning purposes, the use of vigorous chemical solutions, resulting in more efficient decontamination, would be advantageous.

The aim of this research is to identify, develop and assess such a vigorous decontamination technique. The work involves the design and construction of a special test loop and experiments using contaminated samples from reactors, in particular from the Garigliano boiling water reactor. The decontamination efficiency and other important aspects such as treatment of the spent decontaminant, reliability, operational radiation exposure and costs will be investigated. Eventually, a system for in-situ decontamination of a reactor component will be designed.

2.5.2 Work Programme

2.5.2.1 Sample characterization

Samples for decontamination will be taken from representative components that are in the process of decommissioning after having served for about 15 years in the contractor's first-generation nuclear power plants. Initially, samples will be taken from components of the Garigliano reactor which have been contaminated by primary fluid. Particular reference will be made to AISI 304 stainless steel. In addition, carbon steel and certain alloys of interest, such as Monel, will be treated. The operating conditions of the sample materials will be identified (on the basis of the existing documentation) and classified.

The surface contamination will be measured both on the entire components and on the samples. Particular attention will be paid to determining whether alpha emitters are present.

Accurate measurements of alpha and gamma activity will be performed to assess the decontamination efficiency. For this purpose, mobile detectors will be employed for measurements on solid and on liquid samples. Special use will be made of X-ray and gamma spectrometers with crystals of various geometry. The measuring system comprises also an amplifier and a 4096-channel analyser. The data will be recorded and processed by means of computer codes.

2.5.2.2 Selection of chemical solutions for the decontamination experiments

Experimental data on the use of highly aggressive solutions for decontamination will be collected and a number of solutions to be tried out will be selected.

Tests under static conditions on non-active specimens will be performed in order to identify the chemical, chemico-physical and electro-chemical parameters which determine the decontamination efficiency.

2.5.2.3 Design and implementation of a test loop for vigorous decontamination

A test loop will be designed, constructed and commissioned, comprising the following components, among others:

- a test section which allows easy introduction and removal of samples and which can be connected to and disconnected from the loop with a minimum of operations involving high radiation dose rates;
- devices for the variation of the flow-rate and the temperature of the aggressive solution;
- a system of monitoring the activity and the chemical and chemico-physical properties of the solution.

2.5.2.4 Decontamination tests

In preliminary static tests the fixed experimental conditions (temperature, chemical and chemico-physical properties) will be checked.

Dynamic tests will be performed in order to determine the influence and optimum of the flow-rate of solution, the duration of treatment, the sample geometry and any other parameters brought to light by the experiments.

The following measurements and evaluations will be carried out:

- Measurement of decontamination factors as a function of the variables mentioned above, with due regard to the specific activity and colume of the liquid radwaste;
- measurement of the radiation dose absorbed by the workers and estimated correlation with the quantity of material treated;
- analysis of the transport of activity in the cycle with a view to using the results for in-situ decontamination of the component;
- estimate of the specific cost of the decontamination treatment and of the liquid radwaste treatment.

2.5.2.5 Design of a system for in-situ decontamination

Taking into account the experimental results, a complete system for the in-situ decontamination of the regenerating exchanger of the clean-up circuit of the Garigliano reactor will be designed.

2.6 Development of economic decontamination procedures

Contractor: Kernkraftwerk RWE-Bayernwerk GmbH, Gundremmingen, Germany
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 boiling water reactors. The work involves:

- supplementary contamination measurements, in addition to existing ones, in the KRB-A nuclear power plant (237 MWe boiling water reactor 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 boiling water reactor.

2.6.2 Work Programme

2.6.2.1 Primary circuit systems to be differentiated

The decontamination methods must be adapted to the chemical structure of the contamination which, in the case of the components of the primary circuit of a boiling water reactor, is conditioned by the structure of the oxide layers, which are the main carriers of contamination. The following four areas are accordingly differentiated, in respect of which, after analysis of the contamination, treatment methods will be developed (with the exception of system (a) for which decontamination methods have already been tried out):

(a) Primary water system (high level of contamination, deposition on austenitic surfaces at high temperature);

(b) primary steam system (contamination level lower than in (a) by a factor of 100-1000, deposition on ferritic surfaces at high temperature);

(c) condensate system (similar to (b), but lower temperatures);

(d) feedwater system (similar to (b), but higher temperatures with lower activity levels due to condensate purification).

2.6.2.2 Investigation of the contamination in the primary circuit

An overall analysis of the contamination is to be carried out for the systems (a) to (d) defined under 2.6.2.1, involving:

- determination of the state of the plant by means of local dose rate measurements in pipework and components (about 400 measuring points);
- determination of the loosely adhering contamination by means of wipe tests at characteristic points selected in line with the local dose rate measurements (about 80 tests);
- determination of the overall contamination at easily accessible representative points on the basis of scrape tests (about 20 tests).

A detailed analysis will be performed on samples obtained in the before-mentioned investigations and on segments of pipes or components, involving:

- nuclide determination through analysis of the gamma radiation,
- measurement of the beta activity;
- measurement of the long-lived soft beta emitters (about 8 scrape samples);
- chemical analysis of the metal oxides;
- crystallographic determinations of the ferrous-oxide structure by radiographic examination and Mössbauer spectroscopy (about 8 samples);
- determination of the surface roughness of the base material with polished cross-sections (about 8 samples).

2.6.2.3 Development of optimum decontamination methods

The decontamination efficiency as a function of base material removal will be measured on several samples, using an extremely aggressive pickling agent (HNO_3, H_2F_2).

A soft decontamination method for the systems (b), (c) and (d) defined under 2.6.2.1 will be developed on the following principles:

- objective: greatest possible decontamination effect with the lowest possible volume of secondary waste.

- intensity: limited, since the material must not be damaged.
- purpose: preparation of components which, after being overhauled and inspected, are to be re-used in other nuclear power plants.

Reference specimens will be prepared by the methods normally used at the KRB-reactor (soft decontamination, two-stage process with inhibitor, mechanical method involving glass jets).

An aggressive decontamination method for the systems (b), (c) and (d) defined under 2.6.2.1 will be developed on the following principles:

- objective: greatest decontamination effect with smallest volume of waste.
- intensity: no limitation, since damaging the material by means of corrosion is acceptable.
- purpose: preparation for disassembly/conditioning for subsequent disposal.

The following techniques will be tested:

- two-stage process without inhibitor;
- conventional pickling;
- application of pickling pastes;
- electrochemical techniques.

2.6.2.4 Demonstration experiment

The optimum technique developed will be tried out in a large component of the primary-steam, condensate or feedwater system of the KRB-A.

2.6.2.5 Measurement of the contamination of the KRB-A reactor building

The contamination of the inner surfaces of the building, and particularly of the floor, will be investigated to determine the level, distribution and penetration depth and to identify the nuclides.

2.6.2.6 Evaluation

The measurement results will be related as far as possible to the known operating data. An attempt will be made to determine by extrapolation the penetration depth and accretion of the radioisotopes to be expected in a typical boiling water reactor of 1200 MWe after the expiry of its planned operating lifetime.

The results of the decontamination experiments will be evaluated, in respect of both the KRB-A plant and a 1200 MWe boiling water reactor, in order to reveal

- the most suitable technique for complete decontamination (for release or re-use), the arisings and conditioning possibilities of the secondary waste being important factors here;
- whether decontamination of the components studied is advantageous with reference to the radiation dose to workers and the public, if the dismantling, conditioning and disposal take place after a decay time of about 30 years.

2.7 Development of gel-based decontaminants

Contractor: Commissariat à l'Energie Atomique, CEN Saclay, France
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 are exposed to radiation;
- a reduction in the contamination of a component, thereby enabling it to be decontaminated more thoroughly after it has been conveyed to the decontamination bays;
- the decontamination of the vertical walls;
- a reduction in the quantity of radioactive effluents resulting from decontamination.

2.7.2 Work Programme

2.7.2.1 Production of gels

The following range of gels will be produced : silica, glycerophtalic and glycerophosphoric derivates, thixotropic products.

2.7.2.2 Gel/decontaminant compatibility tests

The gels produced will be tested to assess their compatibility with the various types of product currently used for decontamination purposes :

- grease-removing solutions ;
- decontaminating solutions ;
- pickling solutions (nitric acid, sulphuric acid, caustic soda).

2.7.2.3 Decontamination tests

Using real or artificial contaminated samples, tests will be carried out involving the decontamination of materials regularly encountered in nuclear installations (austentic stainless steel, carbon steel, aluminium alloys, plastics).

2.7.2.4 Evaluation of the experimental results

The decontamination factors will be determined and a comparison made with a usual method of applying decontaminating solutions. The production and treatment of secondary waste and the radiation exposure of personnel will be taken into account. The approximate costs of the processes per surface unit will be assessed. Details will be given of the effects of the processes as regards the treatment of radioactive effluents by evaporation and incorporation in bitumen.

2.8 Metal decontamination by chemical and electrochemical methods and by water lance

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

2.8.2.1 Decontamination based on chemical and electrochemical techniques

Decontamination tests will be performed using various baths (acid, alkaline, oxidizing, oxidizing-alkaline baths, etc.) and specimens made from the principal steel grades (austenitic stainless steel) used in nuclear engineering, as well as samples taken from contaminated components of nuclear power plants. The work will involve the following stages :

- The aggressivity of the various baths will be estimated by measuring the density of the corrosion current on a sample placed under decontamination conditions.

- Using the same baths, samples of various steel grades will be soaked at various temperatures and measurements taken to determine their chemical erosion.

- The condition of the specimen surface after soaking will be examined (photomicroscope, surface-finish tester, etc.), in order to determine the regularity of the erosion and, where applicable, instances of pitting.

- For the various steel grades and baths, current-voltage curves will be plotted to show the electrochemical action.
- Measurements will be taken to determine the decontamination factor vs chemical and electrochemical erosion.
- The purification of the effluents will be studied, taking into account the toxicity of the effluents and the quantity of solid waste produced will be determined.
- By aggregating the results obtained, the optimum scouring conditions and formulae for the decontamination baths will be defined.
- A soaking tank in which the components to be treated (maximum surface 0.5 m^2) can be placed either in the cathode or in the anode position will be made.
- Decontamination tests will be carried out on pipework of graphite-gas and light-water reactors, to show the effects of chemical action alone and of chemical and electrochemical action combined .

2.8.2.2 Decontamination based on water lance

Decontamination tests using high-pressure water lance will be conducted at a CEN Cadarache decontamination plant which comprises the following facilities :

- an equipment dismantling-shearing area ;
- a high-pressure generator (700 bars ; jet flow 55 l/min) ;
- a "shooting chamber" where the operator aims from inside at the equipment to be decontaminated ;
- a "shooting cell" where the operator aims from outside at the equipment secured to a steerable turntable ;
- a post-contamination drying and monitoring zone ;
- a unit for treating the resultant effluents.

The effectiveness of spraying at pressures ranging from 100 to 700 bars will be determined on contaminated components of variable geometries, i.e. :

- flat components, frame parts, etc. ;
- external pipework surfaces ;
- complex-geometry components ;
- internal pipework surface.

Industrial water will be used in these spraying tests and, depending on the degree of difficulty encountered in decontaminating the materials, salts of low or slow solubility (boric anhydride, alkali sulphate, etc.), and inert abrasives of the type silica, ferrous oxides, etc. (if possible, capable of being recycled) will be added. The effluents formed are chemically simple and can be easily treated in crushed-resin trimming filters. If necessary, an additional decontamination on ion-exchanger resins will be carried out.

The following parameters will be checked after spraying :

- residual activity of the materials ;
- quantity of effluent produced vs the weight and surface of the equipment sprayed ;
- quantity of solid waste produced to purify the liquid effluents.

2.8.2.3 Chemical treatment and water lance in combination

The abrasive action of the water lance can be supplemented by the erosion produced by an aggressive chemical product. Chemical treatment and water lance will be employed in combination, where one of the methods on its own would not achieve adequate decontamination or where the two methods combined would facilitate the decontamination operation. Conversely, preparation of the surface by means of the water lance may enhance the action of the chemical baths, thus reducing the concentration of reagents required.

2.8.2.4 Evaluation of the results

The methods thus optimized will be evaluated with a view to their application to the dismantling of a nuclear power plant. In particular, the following aspects will be assessed :

- limitations of the application (dimensions of the components to be treated, access to the components, need for liquid-processing units, etc.) ;
- unit costs (e.g., per m2 of surface treated), including the cost of effluent treatment ;
- radiological consequences (radiation exposure of workers, possible release of radioactive products into the environment).

2.9 Economic assessment and decontamination for unrestricted release

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

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 optimization 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

2.9.2.1 Existing regulations

Legal provisions in the Member States of the European Community possessing nuclear power stations on the limit values below which unrestricted release of waste arising from the decommissioning of nuclear power stations is permissible will be compiled. The maximum and minimum requirements will be taken as reference for the following studies.

2.9.2.2. Identification of components which it seems advisable to decontaminate

A criterion catalogue will be prepared with a view to selecting components and structural parts in respect of which conditioning and storage is particularly expensive, but which could be decontaminated with subsequent unrestricted release. The components and structural parts selected on the basis of the criterion catalogue will be listed and the decontamination requirements in each case will be presented.

2.9.2.3 Comparison of "unrestricted release" and "conditioning and storage" in respect of the components selected

Data on costs will be compiled and determined for the alternatives "unrestricted release" and "conditioning and storage" in respect of the components selected, due account being taken of secondary waste. Representative model cases relating to the costs of decontamination, conditioning and storage will be defined and used as a basis, rational assumptions being adopted for the storage costs.

2.9.2.4 Evaluation and recommendations

The dependence of the results on uncertainties in the assumptions will be estimated. In particular, alterations in the legal provisions relating to unrestricted release and re-use will be applied as parameters.

On the basis of the survey, areas in which further development work seems to be necessary will be identified.

3. PROJECT NO. 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 a 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 ;
- 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

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

3.1.2.1. Preparatory Work

The test stand will be constructed in a caisson which will be connected to a mobile ventilation unit.

The following types of test specimen will be prepared or procured :

- solid concrete bodies without and with steel reinforcing ;
- solid heavy-aggregate concrete bodies without and with steel reinforcing ;
- solid carbon-steel parts ;
- solid nodular cast-iron parts ;
- solid high-grade steel parts ;
- scrap carbon-steel components (e.g. tube-bundle evaporators) ;
- scrap high-grade steel components.

The surfaces of the test specimens will be coated to simulate contamination.

3.1.2.2. Testing the dismantling techniques

The following dismantling techniques will be analysed as to their suitability for use on the test specimens listed in the preceding paragraph and tried out where applicable :

- Autogenous cutting technique : The material is first brought to its melting point with an oxyacetylene flame and then burnt by adding more oxygen. This technique is suitable for the dismantling of carbon-steel parts.

- Oxygen-lance technique : Here the energy required for cutting is produced by burning a steel tube with oxygen. The tube is first heated at its end and then oxygen is passed through it. Extra energy is generated either by burning steel powder which has been added or by steel rods introduced into the tube. Because of the high temperatures attainable at the cutting point, high-grade steel and concrete can be cut as well as carbon steel.

- Sawing technique : Here slowly rotating circular saws with large-diameter blades or band saws are employed. The wear and tear on the saw blades and the feasibility of repairing them are particularly important points to be considered.

- Abrasive-wheel technique : The abrasive-wheel techniques known so far - for example, with angle grinders - will be investigated. A study will be made of the use of various abrasive cut-off wheels to increase the cutting depth. The tool life will be given particular consideration.

During the trials, the following characteristics will be determined in particular :

- attainable cutting speed and cutting depth ;
- width of cut ;
- gas and power consumption ;
- reactive forces ;
- quantity and characteristics of the swarf and dust produced.

During the trials, the amount of air exhausted will be continuously monitored. Gas samples will be taken from the exhaust air at regular intervals and analysed. After the trials, the filters in the ventilation unit will be examined to establish the extent of deposition and the amount of residues from the surface coatings of the test specimens.

3.1.2.3. Evaluation of the techniques

The techniques tested and other dismantling techniques will be analysed and evaluated in accordance with the following criteria :

- cutting speed ;
- penetration depth in the material to be cut ;
- suitability for remote control ;
- aerosol formation ;
- secondary waste production ;
- disturbance of the cutting process by corrosion and/or protective layers on the surfaces of the components ;
- radiation dose to the personnel to be expected ;
- contamination of the work area to be expected ;
- environmental impact to be expected ;
- costs.

This evaluation will be the basis for selecting the most suitable techniques for the various dismantling operations.

3.2. Plasma techniques for cutting mineral and metal materials

Contractor : Ansaldo Meccanica Nucleare, Genoa, Italy
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 licensee for Boiling Water Reactor stations, refer to the latter type of plant.

3.2.2. Work Programme

3.2.2.1. Study of the different techniques

The different plasma-arc cutting techniques (thermal lance, plasma torch with or without heat recovery, etc.) will be studied through bibliographical research and contacts made with manufacturers of this type of equipment.

3.2.2.2. Evaluation of the performance of the different techniques

The performance of the different cutting techniques will be evaluated in relation to the various types of ionized gas that can be used, to the various types of structure to which the technique can be applied (thin, medium and thick metal structures, concrete and reinforced-concrete structures, etc.) and to operation in air and under water.

3.2.2.3. Experiments

Laboratory experiments will be carried out in order to validate the theoretical results obtained from the study and evaluations described in the two preceding sections.

These experiments will be defined in detail only after activities in 3.2.2.1. and in 3.2.2.2. have been completed ; the samples used must be representative of components of varying thicknesses in carbon steel and stainless steel and of concrete and reinforced-concrete structures. The contractor will endeavour to obtain some of these samples from ENEL in order to be able to use materials which come from nuclear power stations and are sufficiently contaminated to enable the radioactivity dispersed in the air or water to be measured. The variation in the cutting rate will be determined in relation to characteristics such as the following:

- the type of material ;
- the thickness (the maximum thickness that can be cut will be determined) ;
- the type of gas used.

3.2.2.4. Research into radiation protection and safety considerations

Radiation protection and safety considerations involved in cutting contaminated structures using plasma equipment will be studied with special reference to :

- production of contaminated dusts and/or aerosols and problems involved in catching the latter by means of suitable ventilation systems ;
- production of contaminants in suspension or dissolved in water in the case of underwater cutting ;
- operating reliability, need for direct handling of the structures to be dismantled and foreseeable radiation exposure of the operators;
- impact on the size and organization of the site.

The study will include experimental tests performed at the CNEN test center in Casaccia (Rome) by means of the 30 kW NERTAJET plasma torch available there.

3.2.2.5. Preparation of specifications and recommendations

Specifications and recommendations will be drawn up both for the design and construction of plasma-arc cutting equipment and for the use of the latter in the various types of operation that can be envisaged during the dismantling of a nuclear power station.

3.3. Diamond tipped saws for cutting concrete structures

Contractor : Central Electricity Generating Board, Barnwood, United Kingdom

Work Period : April 1980 - December 1983.

3.3.1. Objective and Scope

The objective of the present 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

3.3.2.1. Development

Tests will be carried out by diamond saw to improve the technical performance. The experimental work will measure cutting depths, cutting speed, quantities of dust or slurry produced, reliability and costs. Problems of remote handling and control will be examined.

3.3.2.2. Application

The method will be applied to cutting remotely blocks approximately 1 metre cube in size from the inner face of a biological shield or a pre-stressed concrete pressure vessel.

3.4. Plasma-oxygen cutting of steel pressure vessels

Contractor : Salzgitter AG, Salzgitter, Germany

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 thick sections of low-alloy steel cladded with stainless steel, occurring in the pressure vessel of light water reactors.

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

3.4.2. Work Programme

3.4.2.1. Cutting Tests

Preliminary cutting tests will be carried out on normal structural steels with austenitic steel plate welded onto them. For the further tests, specimens of up to 600 mm thickness will be prepared from various pressure vessel steels with austenitic steel cladding.

The cutting tests will comprise :

- optimization of the angles and the distance between the plasma and oxygen torches ;
- optimization of the angles and the distance between the combination torch and the work-piece ;

- determination of torch performance data as a function of the cladding thickness and the work-piece thickness.

During cutting tests, dust and aerosol samples will be taken and will undergo the following analyses :

- gravimetric determination of the mass concentration ;
- particle-size determination using electron scanning microscope (particles not greater than 1 micron) and Royco counter (particles not smaller than 1 micron).

3.4.2.2. Testing, optimization and assessment of filter systems

The efficiency and the dust retention capacity of individual filters and of filter combinations will be determined.

Minimum-cost filter combinations will be selected and tested. The radioactivity of the dust and the surface dose rate of the filter systems will be estimated.

3.4.2.3. Evaluation of the test results

A concept for the dismantling of reactor pressure vessels will be developed on the basis of the test results. The exposure of the personnel to radiation, the environmental impact, the secondary waste arisings and the cost will be estimated.

3.5. Dismantling of concrete structures and metal components using laser beam

Contractor : FIAT TTG SpA, Torino

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

3.5.2.1. Experimental facility

The laser equipment to be used in this research is installed at Fiat's Orbassano Research Centre and has a continuous beam intensity of 15 kW. Its main components are a laser source, a control panel and two independent work stations that can be shifted to several positions in order to move the workpiece under the laser beam. A minicomputer assists the machine in coordinating both the adjustment of the reflectors directing the laser beam and the movement of the workpiece.

Additional apparatus will be designed and constructed to enable the existing laser facility to operate under the desired conditions. Suitable suction and filtration systems will also be set up for decontaminating or containing the gases and dusts released from the facility.

3.5.2.2. Samples

Tests will be performed on the samples listed on the following page.

The samples will be suitably prepared for the various experimental conditions in order to reflect the physico-chemical characteristics existing in nuclear power plant structures.

3.5.2.3. Tests

The tests that will be carried out on the samples described previously should determine the maximum thicknesses of steel, concrete and reinforced concrete, possibly clad with steel plate, that can be drilled and/or cut. The test variables will be the following :

- material quality and thickness (see section 3.5.2.2.) ;
- cutting and drilling rate ;
- hole diameter ;
- auxiliary gas used during the operation ;
- laser intensity and type of optics ;
- possibly, remote control and monitoring systems.

LIST OF SAMPLES (see section 3.5.2.2.)

Material Type	Origin	Shape
1. Concrete	Biological shielding, structures	200 mm cube (1)
2. Reinforced concrete	Biological shielding, structures	200 mm cube with reinforcing bars up to Ø 38 mm
3. Prestressed concrete	Structures	See under 2
4. Composite	Vessels, pools	200 mm cube with sheet cladding (1)
5. Ferritic steel	Steam generator shell	100 x 200 mm and 200 x 200 mm plate, up to 100 mm thick (2)
6. Ferritic steel	GRC primary circuit piping	100 x 200 mm and 200 x 200 mm plate, 38 mm thick
7. Stainless steel	Tanks	See under 6.
8. Stainless steel	PWR primary circuit piping	100 x 200 mm and 200 x 200 mm plate, up to 50 mm thick (2)
9. Nickel alloy	PWR steam generator tube bundle	Individual tubes and 200 x 200 mm tube bundles

(1) The dimensions may be modified in the light of the results of the first tests.

(2) Should difficulties arise in cutting samples of these thicknesses, the maximum thickness that can be cut will be determined.

3.5.2.4. The use of lasers in dismantling nuclear power plants

Conditions for the use of lasers in dismantling nuclear power stations will be drawn up.

The containment and monitoring of the products arising from the cutting operations (aerosols, gases and dross), which, in the dismantling of nuclear power plants, could be radioactive, will be studied.
A system for arresting contaminated smoke, fumes and aerosols and for containing dusts produced will be designed and possibly constructed.

A preliminary design study of a dismantling cell and equipment will be made.

Specifications and recommendations will be drawn up for the design and construction of laser cutting equipment and for its use in the various dismantling operations. The advantages deriving from the use of laser techniques will be highlighted and the implications thereof will be studied from a radiation protection and health physics standpoint.

4. PROJECT No. 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 smelting of radioactive steel scrap including investigation of the possibility of decontaminating the melt ;
- development and assessment of techniques for coating metal and concrete parts, in order to immobilize the radioactivity ;
- comparative assessment of various modes of treatment and disposal of radioactive graphite.

4.1. Assessment of management modes for graphite waste

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

Work period : October 1980 - December 1981.

4.1.1. Objective and Scope

Large quantities of radioactive graphite will arise as waste during the decommissioning of gas-graphite reactors. Preliminary estimates have been made of the arisings and the radionuclide composition of the activated graphite and its impurities. A number of potential waste management and disposal schemes have been identified but no comparison has been made of their relative merits, in terms of feasibility and radiological impact.

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.

4.1.2. Work Programme

4.1.2.1. Evaluation of the arisings and radionuclide composition of activated graphite from decommissioning

The existing measurements on the chemical composition (in particular impurity levels) of un-irradiated graphite will be reviewed, together with existing predictions and measurements of the radionuclide composition of irradiated graphite.

Theoretical estimates of the radionuclide composition of irradiated graphite will be made from the chemical composition and will be compared with existing measurements.

Additional measurements on the radionuclide content of samples of irradiated graphite from Advanced Gas-cooled Reactors and Magnox reactors will be made to obtain an improved understanding of the range of variation from one graphite sample to another and to obtain a more comprehensive specification of the radionuclides present permitting extrapolation to the end of reactor life.

The quantity of graphite for disposal and its radionuclide composition will be specified as a result of this review and the additional measurements and theoretical estimates.

4.1.2.2. Specification of waste management and disposal options

Consideration will be given to the following waste management schemes :

- incineration
- disposal on the ocean floor
- disposal in geologic formations.

A variety of waste forms (in terms of packaging and encapsulation) will be specified for disposal on the ocean floor and in geologic formations. When considering incineration due account will be taken of the radiological impact of the various additional waste streams generated.

4.1.2.3. Assessment of the radiological impact of each waste management option

The radiological impact of each management scheme will be evaluated using existing models which describe the transfer of radionuclides to man for activity discharged to atmosphere, or disposed of to the ocean floor or into a geologic formation.

One of the parameters of particular importance in this assessment will be the rate of leaching of radionuclides by both groundwater and seawater. A limited experimental programme will be undertaken to determine the likely magnitude of the leach rate in various conditions. This programme consists of :

- measurements of leach rates of samples of graphite, as an input to modelling studies which will include an analysis of the sensitivity of the radiological impact to leach rates ;
- assessment of the effects of hydraulic pressure on the permeability and mechanical properties of graphite.

4.1.2.4. Overall assessment

An overall assessment will be made of the relative merits of the management options identified for graphite wastes, taking into account the quantities of graphite and of activity involved, and the technological and radiological aspects of each option. Needs for research in support of preferred options will be identified.

4.2. Immobilization of contamination on metals by coating with thermosetting resins

Contractor : Ecopol S.A., Paris, France
Work Period : January 1981 - March 1982

4.2.1. 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 immobilizing the contamination both for components to be placed directly into an ultimate repository and for components to be cut up and further conditioned. The work involves tests on non-active and on active specimens.

4.2.2. Work Programme

4.2.2.1. Review of existing information

Qualitative and quantitative data on paints and coatings with a thermosetting resin base will be compiled and reviewed with reference to constituents, application techniques and coating properties (resistance to impacts, to high and low temperatures, to corrosion, etc.).

4.2.2.2. Preparation of the experimental facility and of test specimens.

The experimental cell and its ancillary equipment, comprising the devices for preparation, storage, coating, ventilation, handling, etc. will be designed and assembled. Metal test specimens for coating will be prepared.

4.2.2.3. Tests under non-active conditions

The coating technique will be optimized using non-contaminated steel specimens of different shapes and varying the coating thickness and material (pure and composite material). Selected coated specimens will be subjected to mechanical impacts, temperature variations and immersion in solutions that are representative of the different possible ultimate repository environments. The data necessary for remote handling of the coating technique will be collected.

4.2.2.4. Tests under active conditions

The effectiveness of the coating will be determined by wipe and leach testing of contaminated parts without coating and with coatings of various thicknesses and materials.

4.3. Cobalt removal from steel by a smelting process of the Electro-Slag Refining type

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

Work Period : June 1981 - February 1983

4.3.1. Objective and Scope

The aim of this research is to assess by means of theoretical studies and tests under non-active conditions the feasibility of extracting cobalt and, possibly, nickel from steel (austenitic steel in particular) by a process of the Electro-Slag Refining type (this is a metal refining process where the metal is smelted by internal electric heating). This process makes it possible to work with a small amount of slag and gives rise to little pollution by fumes. Since the conventional application of the process is for the extraction of easily oxidizable elements, the extraction of cobalt and nickel, more noble than iron, requires the development of new types of slag ; the use of phosphides or sulphides appears to be promising.

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.

4.3.2. Work Programme

4.3.2.1. Specification of reference conditions

On the basis of existing documents and discussions with specialists, qualitative and quantitative data will be compiled, making special reference to Pressurized Water Reactors. The data will be used to prepare the following specifications :

- waste characteristics, i.e. geometry, weight, steel grade, activation and contamination ;
- value of the recovered steel ;
- conditions of reception of the steel waste from dismantling (on the site of the nuclear power plant or of a central facility).

4.3.2.2. Process development

Existing thermodynamic and physico-chemical data will be collected, reviewed and used to define the most promising processes. The main items to be studied are :

- solubility of phosphides (or sulphides) in calcium fluoride ;
- solubility and activity of phsophides (or sulphides) in the steel ;
- relative levels of stability of iron, cobalt and nickel phosphides (or sulphides) ;

- electric conductivity of solutions of phosphides (or sulphides) in calcium fluoride.

Laboratory tests will be performed to confirm and specify the main points in the preparation and use of the saline or metal mixtures to be employed and to determine the partitioning and extraction coefficients.

Having due regard to the results obtained, the most appropriate process will be selected. This process will be tested on a significant scale (heat of several tens of kilograms) under non-active conditions.

4.3.2.3. Industrial-scale design and assessment

The following items will be prepared :

- general design of an industrial-scale unit (including : equipment for melting and extraction, waste preparation, furnace feeding and handling operations ; radiation protection systems ; ancillary equipment ; consumable material) ;
- estimate of investment and operating costs ;
- estimate of operational radiation doses ;
- estimate of the time required to realize the industrial-scale unit.

4.4. Coating of materials to protect against corrosion, fix contamination and avoid powder formation

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

Work period : July 1981 - December 1982

4.4.1. 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.

4.4.2. Work Programme

4.4.2.1. Bibliographic research

In order to identify suitable candidate materials for coating concrete and metal parts, appropriate literature will be studied and contacts made with suppliers of coating materials.

4.4.2.2. Coating tests

The selected coatings will be applied in one or several layers

- on ferrous pieces of various shapes and corrosion states,
- on concrete pieces of various shapes and surface conditions (smooth and fractured surfaces).

The specimens so obtained will be submitted to

- impact and wear tests,
- corrosion tests corresponding to various environmental conditions.

The radiation resistance of the selected coatings will be tested.

4.4.2.3. Conceptual design of coating equipment

The modifications of commercially available equipment will be studied, which are necessary to apply the coatings to contaminated waste material from the dismantling of nuclear power plants.

5. PROJECT No. 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.

Proposals for performing the system study outlined hereunder have been received and contracts are being negotiated.

System study aimed at defining the types of large containers needed

A system study has to be made which is aimed at defining the types of large transport and/or disposal containers needed, depending on the characteristics of the waste, such as radiation level, previous conditioning, etc.

A system of containers suitable for any reactor site taking into account the limitations due to the normal means of transport by railway will be the main subject of the study. Besides, transport by boat will be considered ; in this regard the list of nuclear power plants situated within the Community and which are accessible by boat will be established.

The loading procedure of the shipping casks will equally be studied, taking into consideration, should the occasion arise, particular limitations in the case where the evacuation of waste precedes the dismantling of the reactor containment. The existing ruling concerning the transport of radioactive materials will be taken into consideration.

6. PROJECT No. 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 for the research programme in progress should contribute information on this subject. While awaiting this information, Project No. 6 will be limited in a first phase to the studies defined hereafter, for which a call for research proposals has been issued in November 1980.

6.1. Study of the activated masses of the biological shield

The study covers the estimation of one or more typical reactors existing in the European Community concerning the quantities of material of the biological shield which are activated in such a manner that unrestricted release would not be allowed after a period of relatively strong decrease in activity. The end of this period will be defined as the moment when the activity of a radionuclide with a half-life of more than ten years becomes dominant. Existing measurements or measurements to be made on power plants possibly assisted by computations should enable obtaining detailed flux mapping necessary for the activation computations. The activation calculations will be oriented towards long-lived radionuclides such as C-14 (from N-14), Cl-36, Ca-41, Ni-63, Nb-94, Sm-151, Eu-152 and Eu-154. In order to obtain the input necessary for these computations, analyses of concrete samples from different Community countries will be performed, the objective being the estimation of the variation of trace elements. Measurements on activated concrete samples could confirm the presence of certain trace elements.

The limits for unrestricted release to be considered are :

- reference values for this study of 37, 3.7 and 0.37 Bq/g on specific activity ;
- any other optional value or limitation which could be applied in the proponent's country.

6.2. Study of some technical factors related to the timing of dismantling of reactors after final shutdown

The method of dismantling a reactor will, even after a long decay time, be determined or strongly influenced by the gamma radiation of the activated structures. Consequently, the contents of gamma emitters present in the reactor vessel and its internals are to be studied. Only the most important radionuclides will be considered, with special attention to be paid to Co-60 and Nb-94.

The work includes :

- identification of the radionuclides to be taken into account ;
- measurements of content in relevant parent nuclides, even at trace level, in stainless steel samples typical for the inner surface of the reactor ;
- measurements of the relevant radionuclides on activated steel samples ;
- computation of the activities and the radiation for a plant service life of 30 years ;
- determination of the protections necessary or of the allowable intervention time for workers dealing with the dismantling, taking as a parametre the decay time after final shut-down (between 2 and 100 years).

7. PROJECT No. 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. In order to perform this task effectively, while safeguarding industrial information, the participation of plant designers would be welcome.

A call for proposals of research on the following subjects has been issued in November 1980.

7.1. Catalogue of design features facilitating decommissioning

On the basis of existing studies a catalogue of design features facilitating decommissioning should be worked out for use by nuclear power plant designers. The catalogue should have an annex in which the advantages and disadvantages of each design feature are shown, in particular effects on costs and radiation exposure, taking into account the construction, operation and decommissioning of the plants. The following aspects are in particular to be considered :

- dismantling of the plants : design features facilitating the access, dismantling and handling of radioactive components and systems ;
- long-term storage of the plants (for radioactivity decay) : design features facilitating the conservation of safe conditions and of equipment required for the ultimate dismantling ;
- radioactive waste volumes : design features facilitating the separation of radioactive and non-radioactive materials ; design features reducing the arisings of wastes with long-term hazard potential.

7.2. Prolongation of the useful plant life

Prolongation of the useful plant life could reduce the number of plants required and hence the number of plants to be decommissioned. This would lighten the overall burden of decommissioning, since the burden of decommissioning one plant - in terms of costs, radiation exposure and waste arisings - appears not to be very sensitive to the plant operation time.

A study should be made on the possibilities of designing plants for longer life. The study should include the following items :

- a review of the processes limiting the design life of the critical plant components and systems and of the considerations underlying current plant design life ;
- assessment of possibilities of designing the critical components and systems for longer operation time ;
- assessment of replacement of critical components and systems, including plant design changes required, if any, to enable replacement ;
- cost-effectiveness estimate of the most promising concept for prolongation of plant life.

7.3. Reduction of the cobalt content of reactor materials

Cobalt is the principal parent element for the production, by neutron activation, of cobalt-60. The strong gamma emitter cobalt-60 will in most cases be the dominant source of radiation exposure in decommissioning.

A study is to be made of the possibilities to reduce the cobalt content of the materials which are used within the neutron field or subject to corrosion by the reactor coolant. The study should include the following items :

- review of current material specifications for cobalt impurities (upper limits) and of data on the effective cobalt content (as determined by analysis) ;

- review of the possible influence of the cobalt impurity content on the relevant properties ;
- evaluation of the influence of the cobalt impurity content on the radiation exposure of workers in operation and decommissioning of the plants, considering prompt and delayed dismantling ;
- evaluation of the possibilities to reduce the cobalt impurity content by using low cobalt ores, taking into account the availability of the ores and the influence on costs ;
- evaluation of the possibilities to reduce the cobalt impurity content by special treatments removing cobalt from the raw materials (assessment of the techniques including cost estimates) ;
- cost-effectiveness assessment (cost versus man-rem) of reducing the cobalt impurity content ;
- review on the use of cobalt alloys in primary circuits and on the possibilities of their substitution.

7.4. Concepts minimizing the activation of concrete

Most existing reactor types have a biological shield made of steel-reinforced concrete, part of which is activated during the reactor operation. These shields pose problems of dismantling and of separating the activated parts from the inactive ones.

Design concepts should be worked out, which avoid the activation of the shield concrete by the choice of materials, the use of removable panels, etc. The various possible concepts, verified by neutron flux and activation calculations, should be compared by an evaluation of the additional investment costs over current designs.

7.5. Protection of concrete against contamination

The contamination of concrete structures results in large decommissioning waste volumes.

A study should be made on concrete surface coatings which assure efficient and durable protection against contamination. Coatings which can be easily stripped off in decommissioning would be preferable. The study should include the following items :

- review of the experience with protective coatings of walls and floors in nuclear and other facilities (mechanical resistance, aging, etc.) ;
- inventory of the concrete surfaces in a typical nuclear power plant, for which protective coating would be appropriate ; classification in dry and periodically or incidentally submerged surfaces; determination of the mechanical loadings acting regularly or ocassionally on the surfaces ;
- study of the most suitable economic coatings applied as liquids and as foils ; supporting tests simulating loadings and environmental conditions ;
- cost-effectiveness estimate of the most suitable solutions.

7.6. Improved documentation system

The dismantling of a nuclear power plant will normally be delayed by a radioactive decay period which may extend up to several decades. A good documentation of the plant is, therefore, required for dismantling.

An improved classification and up-to-date documentation system should be worked out, which contains a high amount of information within a small volume. The system, which may use microphotographs and videotapes, should contain all construction drawings and relevant records taken during plant construction and operation. Stereoscopic photographs may solve the problem of producing drawings of the real configuration of the piping systems, including supports, and of the reinforcement bars. The chosen system should be assessed with regard to :

- additional cost over current systems ;
- expected savings of costs and radiation exposure in plant operation and decommissioning.

8. IDENTIFICATION OF GUIDING PRINCIPLES

Guiding principles need to be progressively evolved in order to plan the research and development actions efficiently ; conversely, the results of the actions may influence the shaping of the guiding principles. In view of this interdependence the programme includes as Part B provision for the progressive evolvement of guiding principles, namely :

- certain guiding principles in the design and operation of nuclear power plants with a view to simplifying their subsequent decommissioning ;

- guiding principles in the decommissioning of nuclear power plants, which could form the initial elements of a Community policy in this field.

The first step will be to assemble relevant material for guiding principles prepared in the Member States.

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ANNEX

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