SAFETY ASSESSMENT FOR DECOMMISSIONING

Annex I, Part B

Safety Assessment for Decommissioning of a Research Reactor

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CONTENTS

FOREWORD

The purpose of Annex I is to provide a demonstration of the application of the DeSa safety assessment methodology described in the main report. For that purpose three examples of facilities to be decommissioned were selected by the DeSa project participants for evaluation. The chosen test cases are broadly representative of ongoing or completed decommissioning projects.

The test cases selected for evaluation were:

- A nuclear power plant (NPP);
- A research reactor; and
- A nuclear laboratory.

The facilities were selected because they represented a range of differing types of facility and because the operating organizations had committed to provide all necessary technical information to allow safety assessments to be conducted.

Once the safety assessments for the decommissioning of NPP, research reactor and the nuclear laboratory had been developed, each test case report was reviewed by the Regulatory Review Working Group and the Graded Approach Working Group to provide a simulation of a regulatory review and to demonstrate that the regulatory review procedure developed for DeSa (see Annex III) and the recommendations on the graded approach (see Annex II) are robust.

In Part B of Annex I the DeSa safety assessment methodology is applied to a research reactor that was shut down in 2001 for immediate dismantling. The research reactor was a homogeneous liquid fuelled and moderated reactor with a low thermal power. Its enriched uranium fuel was in the form of uranyl sulphate. The reactor had been shut down and its fuel removed before the commencement of decommissioning.

The purpose of the safety assessment is to support the decommissioning plan and the licence application for decommissioning. The safety assessment also aimed:

- To confirm the safety of workers and public during the planned decommissioning activities;
- To identify the requisite safety control measures; and
- To act as a basis for seeking regulatory approval to proceed.

1. INTRODUCTION

1.1. BACKGROUND

Evaluation and demonstration of safety is an essential component of the successful planning, performance and completion of decommissioning of facilities using radioactive material. This has been highlighted by the international safety standards on decommissioning of nuclear power plants, research reactors medical and nuclear research facilities, etc. Recognising the need for exchange of information and experience and consolidation of the best experience and lessons learned in these areas, the International Atomic Energy Agency (IAEA) launched in 2004 the international project on "Evaluation and Demonstration of Safety for Decommissioning of Facilities Using Radioactive Material (DeSa)". This project aimed to develop a harmonized methodology for evaluation and demonstration of safety during decommissioning and to develop safety assessments for selected facilities by applying this methodology. The application of the methodology developed as part of the project was to be tested in several test cases.

The DeSa test cases aim to provide practical illustration of the application of the methodology (presented in the main report), to illustrate the need and application of a graded approach due to the complexity and hazards of the facilities.

The test cases resulted in safety assessments for a selected number of facilities with different complexities and hazards (e.g. nuclear power plant, research reactor and a nuclear laboratory) following the individual steps of the methodology. By developing these test cases, practical issues related to the use of the methodology were identified, such as the criteria for selection and justification of scenarios and models, definition of types of uncertainties and approaches for their treatment. Decisions on the importance of input data required, the use of generic vs. site specific data, as well as the depth for safety assessment necessary for demonstration of safety for decommissioning of various nuclear facilities with different hazard potential will also be addressed in the project. The formulation of the test cases was based on the information provided by volunteer facilities. With respect to the intended demonstration of the application of the DeSa safety assessment methodology the volunteer facilities became simplified due to the time and resource constraints of the DeSa project. This resulted to some extent to limitation of the volume of information documented in Annex I of this report, which is needed to understand the facility and approach adopted for the safety assessment.

1.2. SCOPE

Part B of Annex I presents the safety assessment for decommissioning of a small research reactor (referred to also as the Research Reactor Test Case). For this test case it is assumed that the chosen decommissioning strategy is immediate dismantling with a view to remove the reactor building from regulatory control.

1.3. OBJECTIVES

The objective of Part B of Annex I is to illustrate the application of the DeSa safety assessment methodology developed as part of the DeSa project to a relatively small size nuclear facility (a research reactor) by applying the graded approach discussed in the main report and Annex III. The safety assessment and its results are not supplemented by additional documentation (e.g. facility description) which may be required by the national regulatory system to be submitted to the Regulatory Body. Therefore some sections within Part B, especially on the facility description, are introduced to provide information to allow understanding of the safety assessment.

1.4. STRUCTURE

The structure of the report is based on the methodology developed by the DeSa project. Section 2 of Part B describes the assessment framework starting with its context, scope, timeframes, final end state up to safety assessment approach and management measures.

Section 3 contains a description of the safety relevant structure, systems and components of the research reactor during decommissioning and the decommissioning strategy, including description of the dismantling activities, as well as of the radioactive inventory of the nuclear facility and details on environmental aspects. It has to be noted, that in a full scale decommissioning project and depending on the national requirements, the description of the full nuclear facility may be part of the safety assessment documentation or may be part of documents forming a decommissioning plan. For this test case a description of the research reactor is part of the appendix.

The identification of radiological hazards is covered by Section 4. Section 5 covers the detailed analysis of both normal and accident scenarios. According to the DeSa methodology Section 6 will provide details of the engineering analysis conducted. In Section 7 the results of the safety assessment will be evaluated and the limits and controls needed to ensure safety are explained.

Section 8 contains an explanation on the graded approach applied to the research reactor and serves to justify the approach applied.

Finally, Section 9 provides information to demonstrate, that the results of the safety assessment are reliable and thus confidence is built in the performed safety assessment. Again, this section might usually be supported by additional documents e.g. on the management system in place for the decommissioning project. Thus, to some extent in Part B of Annex I information is presented which usually will not be covered by the document on the safety assessment, but will be part of the overall decommissioning plan [1, 2 and 3]

Section 10 summarizes the results of the safety assessment and the lessons learned from applying the DeSa methodology. Especially this Section 10 reflects the recommendations resulting from the review processes of the Graded Approach Working Group and by the Regulatory Review Working Group with regard to the DeSa safety assessment methodology.

A set of seven appendices supplement Part B of Annex I.

It has to be noted, that depending on the national regulatory system and the requirements of the Regulatory Body either the content of Section 4 might not be submitted to the Regulatory Body, or for a less complex nuclear facility with low radioactive inventory as this research reactor, the content of Section 4 might be regarded sufficient for the assessment of safety to be performed. This may mean that no detailed analysis might be required to be submitted to the Regulatory Body. Nevertheless, for the illustration purpose of this test case safety assessment documentation both sections will be provided.

2. ASSESSMENT FRAMEWORK

2.1. CONTEXT OF SAFETY ASSESSMENT

The safety assessment forms part of the documentation presented by the operator for approval by the Regulatory Body prior to initiating decommissioning of the research reactor. The reactor is located on the premises of a research centre with several nuclear facilities and facilities using radioactive material (e. g. research reactors, radioactive waste treatment facility, see Section 3.1). The basis for the safety assessment for decommissioning of the facility was set up on the basis of radiological characterization which provides sufficient information about the distribution of activity for decommissioning planning purposes. Prior to starting the decommissioning activities of the research reactor the fuel solution and the radioactive material, resulting from operation, were removed.

No effects on the safety during decommissioning resulting from interdependencies between the research reactor under decommissioning and other nuclear facilities which are present in the research centre are taken into account in the safety assessment described below. Due to their technical characteristics and distance from the research reactor no significant influence on the safety during decommissioning does exist (see Section 3.1).

In addition, no treatment of radioactive material or waste, including the clearance of radioactive material, is subject to this safety assessment, as radioactive material will be transferred to a separate radioactive waste treatment facility at the research centre site. The transport of radioactive material or waste is taken into account in the safety assessment only as far as the handling is inside the reactor hall.

2.2. SCOPE OF THE SAFETY ASSESSMENT

The safety assessment covers all decommissioning activities that start after tapping of the fuel and removal of the operational radioactive waste and terminate with the release of the remaining former reactor hall for unrestricted use. As the controlled area during operation was limited to the reactor hall itself, the scope of the safety assessment is limited to any activity inside the reactor hall. Measurements to ensure that the area surrounding the reactor hall is free of radioactive contamination from operation of the research reactor or from decommissioning activities are subject to a separate survey project.

Due to the distance to any further nuclear facility, facility using radioactive material and conventional facility at the research centre (see Section 3.1) no impact on the safety of the research reactor is envisaged. Aspects on safety of these facilities, including the radioactive waste treatment facility, are not subject to this safety assessment but to dedicated safety assessments related to their individual licences.

Material to be cleared and radioactive waste handling are taken into account only to the extent that they are relevant to safety during the decommissioning. Both material to be cleared and radioactive waste, will be transferred to the radioactive waste treatment facility at the research centre. Safety of the transport and related consequences are taken into account as far as it takes place inside the reactor hall, all aspects outside the reactor hall are subject to the safety assessment which is supporting the operational licence of the radioactive waste treatment facility (including on-site handling).

2.3. OBJECTIVES OF THE SAFETY ASSESSMENT

The safety assessment aims to demonstrate that the decommissioning activities can be carried out without any undue exposure of the worker, the public or the environment or any undue release of radioactive material to the environment. The main focus is placed on assessing radiological hazards. Nevertheless, non-radiological hazards are screened with regard to their potential impact on the radiological consequences. Furthermore, the safety assessment demonstrates to the Regulatory Body that the decommissioning activities can be performed with personnel doses well within the limits and constraints, which are based on the as low as reasonably achievable (ALARA) principle.

2.4. TIMEFRAMES

The decommissioning of the research reactor is scheduled to be carried out within a time frame of 18 months (including time for clearance of the former reactor hall after completion of the dismantling). The decommissioning activities are separated into the following four work packages (see Appendix I):

(a) Dismantling of systems (WP1);

(b) Demolition of the active parts (reactor block) (WP2);

(c) Demolition of the non-active parts and clearance (WP3); and

(d) Final activities, including final survey and documentation (WP4).

A graphical representation of the duration and interdependence of the work packages of the decommissioning activities is shown in Figure 1. Further details can be obtained from Appendix I.

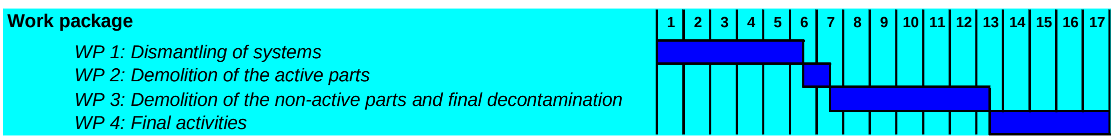

FIG. 1. Schedule of the work packages of the decommissioning activities (scale in months).

2.5. END POINTS OF THE DECOMMISSIONING STAGES

The decommissioning activities are planned to be carried out as part of one licence to be granted by the Regulatory Body taking into consideration this safety assessment, i.e. the decommissioning project will be performed in one decommissioning stage. The end point of this single decommissioning stage is the end state of the decommissioning project. It is envisaged that after the research reactor and its auxiliary systems are dismantled, the reactor hall will be decontaminated and released for unrestricted use from regulatory control. For that, the wooden floor in the reactor hall is planned to be partially removed and the foundation to be checked for possible contamination and compliance with relevant clearance levels will be demonstrated.

Note, that depending on the national regulatory system the release of the reactor hall from regulatory control might be subject to a separate licence.

2.6. REQUIREMENTS AND CRITERIA

For the purposes of the Research Reactor Test Case the radiation protection standards laid down in IAEA Safety Reports Series No. 115 [4] is used. Further criteria are derived from other international recommendations as follows [5 and 6]:

— A dose criterion of 0.3 mSv/a for evaluating doses to the public from release of radioactive material from the site during normal decommissioning activities is used;

— The assessment of doses to workers is carried out using a dose constraint of 2 mSv/y which is a tenth of the average annual dose limit for exposure (20 mSv per year and 100 mSv over 5 consecutive years) and corresponds to dose constraints normally used for evaluating similar situations [4];

— A dose criterion of 50 mSv for workers and of 5 mSv for members of the public are used for evaluating incident and accident scenarios during decommissioning [4];

— An equivalent dose limit of 500 mSv/a for the hand of a worker during normal decommissioning conditions is used [4]; and

— With regard to the release of the reactor hall any remaining activation or contamination have to be below those clearance levels as laid down in [5 and 6] to meet the 10 µSv concept (De Minimis Concept).

Note, as the clearance of radioactive material is performed outside the decommissioning project by the radioactive waste treatment facility, located at the research centre, no criteria on clearance of that material is considered in this safety assessment. For the internal handling of the radioactive waste and material to be cleared, which are collected during decommissioning in waste containers the operator voluntarily proposes to apply criteria relevant for waste packages and overpacks to the waste drums according to [7] (see Section 3.6.1).

2.7. ASSESSMENT OUTPUTS

The safety assessment analyses the effective doses to workers and to members of the public, both during normal decommissioning conditions and during incidents or accidents. The results of the assessment are compared with the relevant criteria set out in Section 2.6. If the assessment results are compliant with the criteria, the decommissioning activities can be carried out as planned. Otherwise, appropriate modifications are necessary resulting in a re-assessment of safety on base of the modifications performed. Note that, depending on the national legal and regulatory system and requirements of the Regulatory Body a licence might be granted before any decommissioning activities can be started.

In addition, conventional hazards are identified, but no quantitative analysis is carried out.

2.8. SAFETY ASSESSMENT APPROACH

The safety assessment for the research reactor is based on the safety assessment methodology proposed by the DeSa project and documented in the main report. A graded approach is also applied in this test case, which takes into account the low level of contamination and potential radiological risk for workers, the public and the environment. Figure 2 gives an overview on the relevant steps within the DeSa safety assessment methodology applied in this assessment.

In the hazard identification, as described in Section 4, the “Check list” and “What-if technique?” approaches are applied; these approaches are explained in detail in Volume I of this report.

Further on, the detailed analysis as described in Section 5 is based on a preliminary hazard assessment and screening, in which conservative scenarios and calculations without in detailed site specific models are used to define the relevant scenarios.

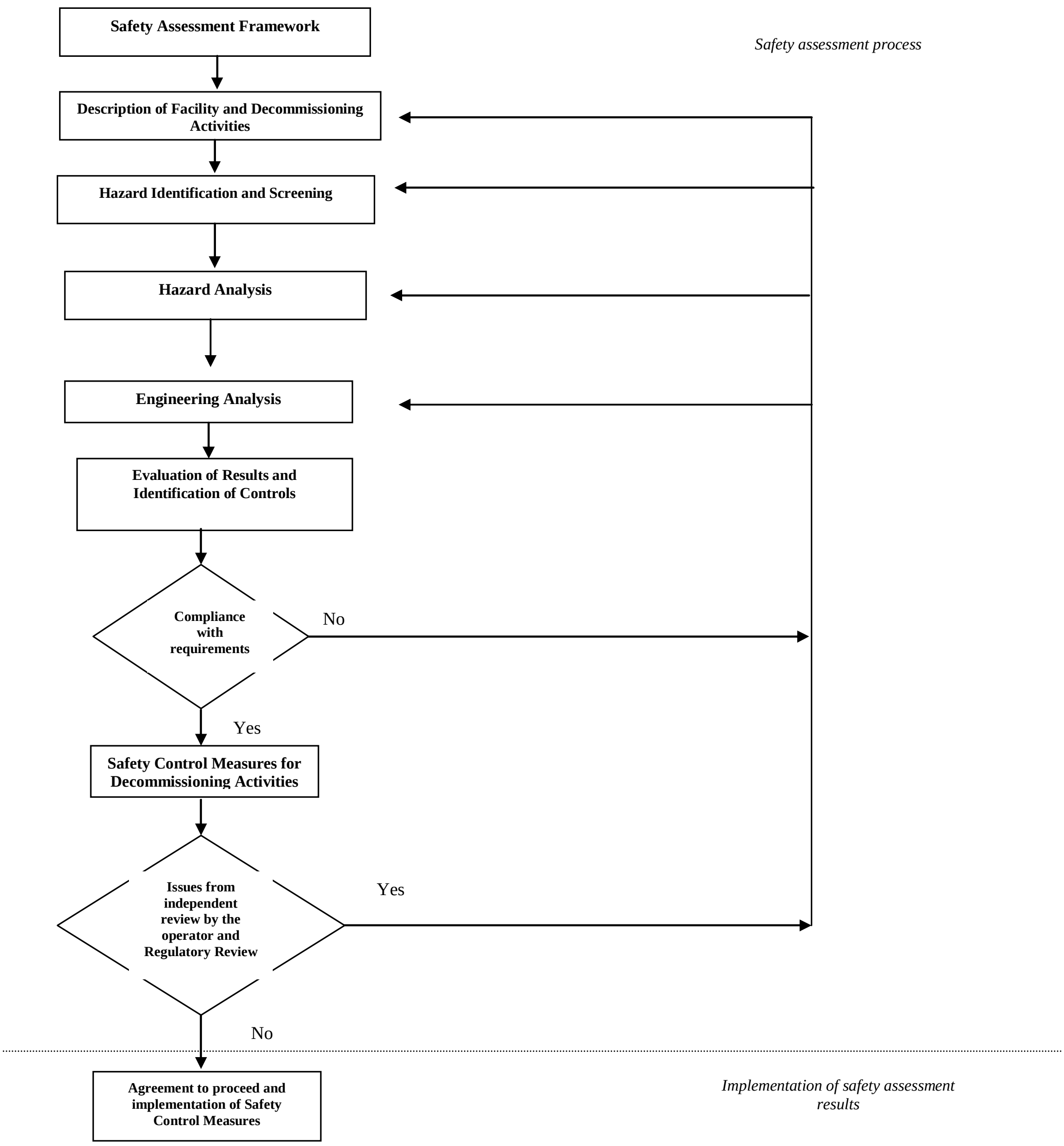

FIG. 2. Applied DeSa safety assessment methodology [3].

2.9. EXISTING SAFETY ASSESSMENT

The main source of risk studied in the safety assessment during the operation of the research reactor was a leakage of the fuel solution. Since this fuel solution is no longer present at the research reactor at start of the decommissioning (see Section 2.1), the operational safety assessment does not provide significant safety relevant results which can be used within the safety assessment for the decommissioning.

In real safety assessment for licensing purposes, detailed description of the climate, earthquake or the hydrological situation could be obtained from the operational safety assessment (if being part of that) with benefit for this safety assessment. Due to the low radioactive inventory and low complexity of the research reactor resulting in minor consequences in case of an accident such detailed information are not taken into account as the safety assessment is based on conservative assumptions (see also Section

2.10).

2.10. SAFETY MANAGEMENT MEASURES

In a real decommissioning project the operator's safety management system will be described as part of the final decommissioning plan and will be subject to the Regulatory Body's final decommissioning plan review to assess whether the system is appropriate to ensure safety.

For the purpose of demonstration of the DeSa safety assessment methodology described in the main report and in Annex III of the main report, it is assumed that the operator of the research reactor has in place an effective safety management system. This safety management system ensures that all work is carried out in accordance with the regulatory framework of the country, the operator's policies and procedures and that staff and contractors involved are appropriately qualified and experienced for the work to be performed.

3. DESCRIPTION OF THE FACILITY AND DECOMMISSIONING ACTIVITIES

The research reactor is a thermal homogeneous liquid fuel research reactor (see Figure 3). Originally, the reactor was built to generate an output of 5 W. It was commissioning in August 1957. In the spring of 1959, the output was increased to 2 kW following the installation of cooling systems and improvement of the shielding and the reactor was subjected to a test run at 2.3 kW. At an output of 2 kW, the maximum thermal flux in the reactor was approximately 6 $l0^{10}$ n/(cm^2.s). The reactor used 19.9 % enriched ^{235}U as in the form of uranyl sulphate dissolved in light water. When the reactor was started, 0.984 kg of ^{235}U and 15.5 l of solution volume were added. The research reactor was shut down in 2001.

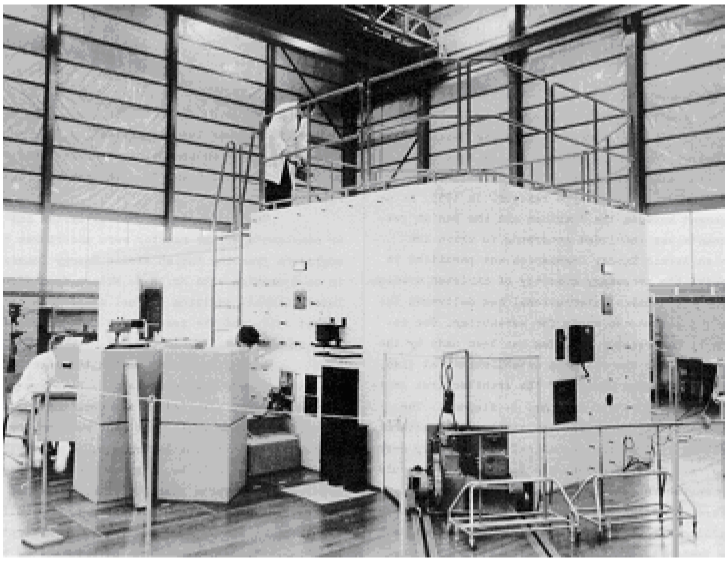

FIG. 3. Research reactor.

Note that depending on the national requirements a more detailed description of the research reactor to be decommissioned might be needed in the safety assessment report. This information will be also part of the final decommissioning plan.

This safety assessment report was developed as a standalone report that can be read without additional documents. A more detailed description is presented in Appendix II.

3.1. SITE DESCRIPTION AND LOCAL INFRASTRUCTURE

The reactor is a nuclear facility at a multi-facility nuclear research centre which is located at a peninsula (see Figure 4). About 1300 people have their daily work at the research centre. The site facilities consist of conventional research facilities, of a radioactive waste treatment facility, a hot cell complex and of two research reactors currently under operation. The nuclear facilities are in a distance of at least 500 m from the research reactor. The conventional research facilities are in a distance of at least 200 m. With regard to the decommissioning of the research reactor the area outside the research reactor is regarded as public area with restricted access.

East of the centre itself (right hand side of Figure 4) there is a group of 16 single family houses, housing a total of about 40 persons, and a hostel with 13 single rooms for guests.

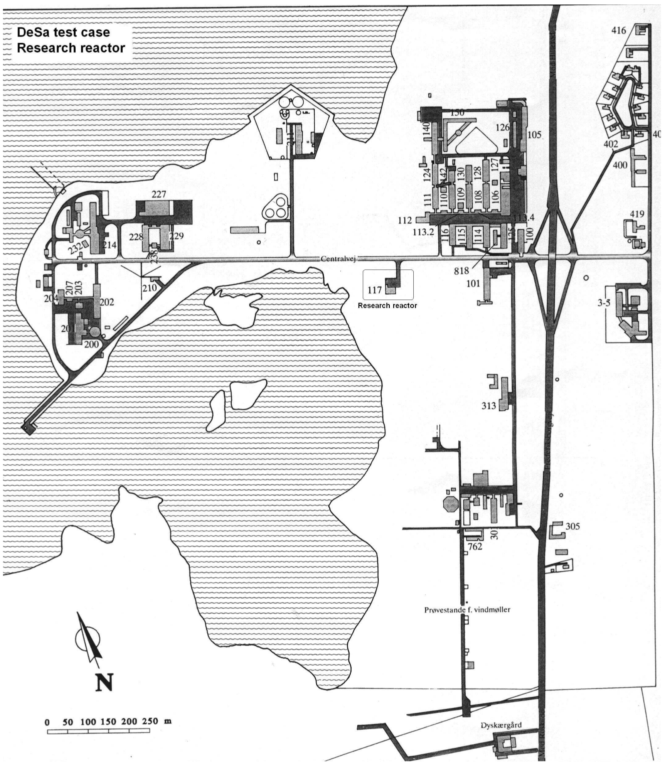

FIG. 4. Map of the research centre.

Most of the immediate surroundings is farmland and – to the west – a fiord. About 2 km south of the site there is a settlement with about 200 single family houses. The population within a distance of 10 km from the site is about 65 000, with about 50 000 concentrated in a city 6 km to the south of the site. The population between 10 and 15 km away from the site is about 105 000.

No further details of the hydrological and the hydrological situation are provided as no dispersion of radionuclides via water pathways during the decommissioning is expected to occur. No details on the weather situation are provided. Instead, conservative assumptions of a standard weather situation with westerly winds are used.

3.2. SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

After completion of the research reactor operation (see Section 3.4) – but still under the terms and conditions of the operating licence – the fuel solution was tapped and the primary circuit has been flushed. As a consequence safety functions was to be ensured during the conduct of the decommissioning activities were identified:

- Limitation of the exposure of workers and the public; and
- Containment of radioactive material and avoidance of discharge, i.e. uncontrolled and unplanned release of radioactive material.

Safety related structures, systems and components (SSCs) needed also during decommissioning of the research reactor are:

(a) The existing ventilation system;

(b) The existing fire detection system;

(c) The enhanced radiological monitoring system of the reactor hall; and

(d) A new storage cell to temporarily store radioactive waste in the reactor hall.

Existing ventilation system

Figure 5 provides an overview on the layout of the ventilation system of the research reactor. It is assumed that the system is in compliance with the relevant national requirements and is also subject to inspections by the Regulatory Body. It is fit for purpose as it has been designed to meet the requirements resulting from the normal operation of the research reactor and related incident and accident conditions.

Depending on the sequence of the decommissioning activities the ventilation system will be adapted to ensure airflow through the stack of the reactor hall. As long as the biological shielding is not subject to decommissioning airflow through the floor will be established and the radiation monitoring from operation of the reactor will be used. Any required modification of the ventilation system is subject to agreement by the Regulatory Body.

In addition to the existing ventilation system a mobile system, consisting of a tent and a filtering system, is planned to be used when the biological shielding will be dismantled. The system will establish an adjusted air flow from the reactor hall to the tent. The filtered air will be then released into the reactor hall again. The filter system is also assumed to be in compliance with the relevant national requirements and is subject to adequate work procedures to ensure reliable and correct functioning.

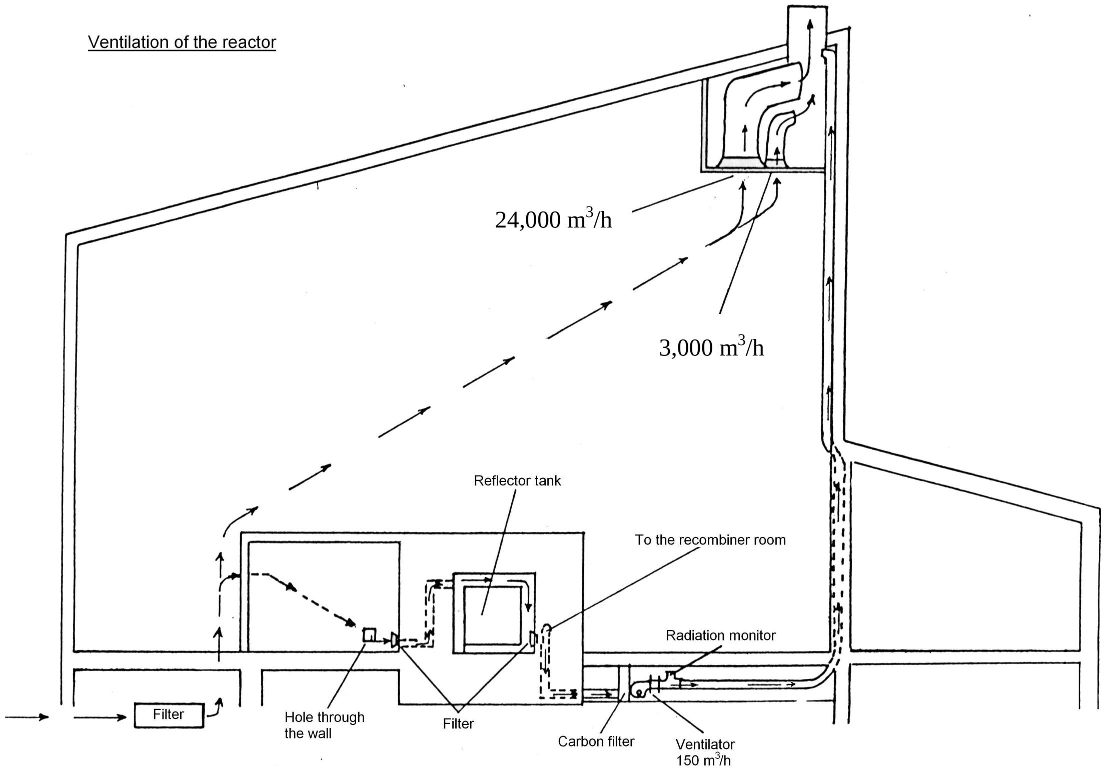

FIG. 5. Block diagram of the ventilation system of the reactor hall.

Existing fire detection system

The existing fire detection system of the reactor hall will be used to detect potential fire in the reactor hall. As during the operation of the research reactor, it is planned to remain connected to an alert system in the reactor hall and to the central guard house of the research center. This is to ensure that the personnel in the reactor hall are alerted immediately and that fire brigades are activated as soon as a fire is detected.

In addition, in case of any decommissioning work to be performed which might result in the ignition of combustible material appropriate work procedures and work control procedures and portable fire protection equipment (e.g. fire extinguishers) are foreseen to avoid a fire or to immediately mitigate consequences.

Enhanced radiological environmental monitoring system

In addition to the radiological monitoring system in the floor ducts of the ventilation system, a new is planned to be installed at the stack of the reactor hall. The system will consist of a β/γ online monitoring system and of a $\alpha/\beta/\gamma$ sampling system, which will analyse the balance of discharged radioactive material from decommissioning activities or immediately after an incident. The detection limits of the β/γ online monitoring system will ensure that any release of radioactive material from the facility and the site due to an incident can be measured and controlled. The analysis of the samples will be carried out by a radionuclide laboratory which is certified according to related national requirements and standards. The total stack system will be subject to an inspection and maintenance programme, which is under control of the safety management system.

The stack system is compliant with the related national requirements to detect a release of radioactive material and is subject to periodic inspection by the Regulatory Body. Any modifications of the system are subject to an agreement by the Regulatory Body.

Depending on the national requirements, in addition to the static radiological environmental monitoring system at the stack, a mobile $\alpha/\beta/\gamma$ radiological monitoring system is intended to be used to monitor a potential release of radioactive material into the reactor hall during decommissioning activity. The probes will be assembled by radiation protection personnel and according to written instructions for those decommissioning activities with potential release of radioactive material.

The monitoring system is assumed to be compliant with related national requirements and standards.

New storage capacity for radioactive waste

A temporarily storage capacity (a cell) for radioactive waste in the reactor hall is planned to be developed. This cell will be used before the radioactive waste can be handled to the radioactive waste treatment facility on the site. A shielded storage cell will be built at the southern end of the reactor hall (see Figure 3.4.). The cell walls will be of 60 cm thickness and made of concrete bricks to ensure a sufficient shielding of the working places taken due account of the dose rate limitations of the radioactive waste drums. To avoid accidental access to the cell the entrance will be closed by concrete bricks so that access will be possible only by overcoming that obstacle.

The design of the storage cell will be based on related national conventional and radiological requirements to ensure structural integrity and radiological protection. Any modification of the cell will be subject to agreement by the Regulatory Body.

Existing reactor hall crane

During decommissioning of the existing reactor a hall crane will is planned to be used to lift heavy load and contaminated material. Depending on the national requirements the crane might be regarded as a safety related system. In the Research Reactor Test Case and taking into account the results of the safety evaluations (see Section 4.3) the crane is regarded not to be a safety related system but important equipment which will be subject to inspection and maintenance programmes according to convention safety to avoid drop of any material.

3.3. RADIOACTIVE INVENTORY

After tapping of the fuel solution and partial decontamination of the primary circuit, characterization of the research reactors was performed to determine the radioactive inventory of the research reactor prior decommissioning. The results are summarized below and details also presented in Appendix III and Appendix IV.

3.3.1. Activation and contamination of different components of the reactor

The reactor contains a very limited amount of radioactivity because of:

- The small output of the reactor;
- Very limited period of operation;

— Tapping of the fuel solution; and

— The partial flushing of the primary circuit).

In total the activity content in the reactor is a few GBq β/γ activity in the remaining core solution and hardly any α activity.

Table 1 provides a more detailed overview of the different parts of the reactor and of their radioactive inventory. Details on estimated activations, contamination and dose rates at the research reactor are presented in Appendix III.

TABLE 1. PARAMETERS OF REACTOR COMPONENTS

Component	Dimensions*	Mass [kg]	Activity [Bq]	Remark
Reactor tank and pipe connecting the tank to the recombiner	Tank: d = 32 cm Pipe: d = 6 cm, l = 100 cm	Tank: 5.5 Pipe: 2	2 x 10^9 (^{137}Cs) 4 x 10^6 (^{90}Sr) < 5 x 10^7 (α) 3 x 10^8 (^{60}Co)	contamination contamination contamination activation
Recombiner	h = 50 cm d = 25 cm	35	1 x 10^9 (^{137}Cs) 10^6 Bq of ^{90}Sr	contamination
Control rods	l = 125 cm, w = 10 cm t = 1.3 cm	15	5 x 10^7 (^{60}Co)	activation
Graphite reflector	h = 130 cm d = 150 cm	4 141	1 x 10^8 (^{152}Eu)	activation
Reflector tank	h_i = 130 cm d_i = 150 cm t = 6 mm	525	1 x 10^8 (^{60}Co)	activation
Reflector tank lid	d = 160 cm t = 6 mm	200	1 x 10^7 (^{60}Co)	activation
Steel plates on inner side of concrete shield	w = 360 cm (total) h = 162 cm t = 13.5 mm	700	1 x 10^7 (^{60}Co)	activation
Concrete shield (heavy concrete)		20 000 (fraction above clearance limit)	1 x 10^7 (^{60}Co)	activation
Beam plugs	10×10 cm across l: 60 cm	896	5 x 10^4 (^{60}Co, ^{152}Eu)	activation
Experimental plugs	10×10 cm across l: 60 cm (active part)	61	5 x 10^6 (^{60}Co, ^{152}Eu)	activation

*) l = length; d = diameter; h = height; w = width; t = thickness

During decommissioning of older facilities using radioactive material, it is important to know whether other hazardous materials, such as asbestos, were used during construction of the facility. With regard to the research reactor used in this test case, asbestos has not been observed, so far. Nevertheless, as a measure of precaution this issue will be considered and examined during the detailed planning of each decommissioning activity.

The main conclusions of the reactor characterization, as well as a subsequent measurement of the radiation level through the core vessel, are outlined in the following Sections.

(a) Reactor tank and the pipe connecting the tank to the recombiner

During the characterization of the research reactor it was not possible to measure the activity of the reactor tank and the tube connecting the tank to the recombiner. This activity originates partly from the neutron activation of the stainless steel of the reactor tank and the connecting tube, partly from fission products that has been deposited on the walls of the two components. The Department of Applied Health Physics of the research centre had measured the γ-dose rate through the "Glory Hole" (V2) with TL-dosimeters, but the radiation from a Ra-Be neutron source in the graphite reflector (0.1 Ci) dominated the radiation around the reactor tank. Subsequently, the Ra-Be neutron source was removed and a new measurement was made. The report of this measurement is enclosed as Appendix D. Based on this measurement, the remaining activity of γ-emitters in the core vessel is estimated to be about 2×10^9 Bq of ^{137}Cs.

(b) Recombiner

For determination of the radiation from the recombiner, the concrete block above the recombiner was removed and the γ-dose rate was measured in three distances above the recombiner and two below it. Then the γ-spectrum was measured with the spectrometer hanging above the recombiner in the hook of the crane. The measurements resulted in an assessment of the γ-activity in the recombiner of approximately 1×10^9 Bq of ^{137}Cs (with the estimated uncertainty of a factor 2).

(c) Control rods

The control rods were made of boron with a stainless steel casing. Based on measurements on a regulating rod and a safety rod, the total γ-activity of the four control rods is estimated to be about 5×10^7 Bq of ^{60}Co with about 75% of the total activity concentrated in the regulating rods, which was close to the core all the time of the reactor operation.

(d) Graphite reflector

From the measured activities of graphite stringers (including samples) and from the graphite volume of the reflector (about 2 400 dm^3) it is estimated that γ-activity of graphite is about 1×10^8 Bq of ^{152}Eu. No evaluation of the ^{14}C-activity, present in the graphite reflector was carried out.

The graphite stringers do not contain any jacket. They will be dismantled step by step using a remote suction system. Then they will be measured with respect to radiological and geometrical properties and posted into (6 to 9) cubic aluminium containers, closely located to the biological shielding during handling of the graphite stringers. (Note: For illustration purposes for this test case Figure 6 shows a picture from the later dismantling of the graphite stringers, where the cubic aluminium containers and the storage cell can be seen, see Fig. 22).

FIG. 6. Cubic aluminium containers for storage of the graphite bricks and the storage cell during dismantling.

(e) Reflector tank

The mass of the reflector tank is estimated, on the basis of it dimensions, to be about 500 kg. Taking this into account and the measured activity of a disk cut out of the reflector tank, it is estimated that the activity of the reflector tank is about 1 x 10^8 Bq of ^{60}Co.

(f) Reflector tank lid

As there were no drawings available of the reflector tank lid, its' mass of about 200 kg was estimated based on the available data. From the measured activity of the material sample of the reflector tank lid (see Table 27) the total activity of the lid is estimated to be about 1 x 10^7 Bq of ^{60}Co.

(g) Steel plates on the inner side of the concrete shield

The four steel plates situated on the inner side of the concrete shield between the firstly cast and the four concrete corner pillars are estimated to have a total mass of about 700 kg. From the measured activities of the two material samples of the plates (see Table 26) the total activity is estimated to be about 1 x 10^7 Bq of ^{60}Co.

(h) Concrete shield

From the measurements on the cores of the four drillings in the concrete shield, significant activity was only detected at the inner end of the core from the drilling down through the central concrete block above the reactor. This piece had a length of about 6 cm and an activity of 5 290 Bq. Assuming that the activity is the same everywhere in the lowest 6 cm of the top shield, a conservative assumption, the total estimated activity of the top shield is about 1 x 10^7 Bq of ^{60}Co. In the cores of the horizontal drillings the activity due to activation, even on their inner end, was determined to be almost at background level.

(i) Beam plugs

Measurements of the inner beam plugs showed that the major part of the activity is located in the innermost 5 cm. From the measurements, the total activity of the beam plugs is estimated to be about 5 x 10^4 Bq of ^{60}Co and a significantly lower activity of ^{152}Eu.

The fact that no activity was measured on the innermost end of the horizontal borehole cores does not agree with the fact that dose rates above the background level were measured at the inner 15 cm of the inner beam plugs and that activity above background level was detected on these plugs. However, in this context it needs to be noted that the steel plate on the inner side of the concrete shield is thicker (1.35 cm) than the steel plate at the inner end of the inner beam plugs (0.35 cm) and that the concrete in the beam plugs does not seem to be the same as that of the concrete shield.

(j) Experimental plugs

The research reactor was provided with a number of experimental plugs that were not part of the initially supplied plugs. Due to the materials used in the plugs their activity is larger than that of the standard plugs and stringers. The total activity of these plugs is estimated to be about 5 x 10^6 Bq of ^{60}Co and a significantly lower activity of ^{152}Eu.

(k) Radioactive inventory in the reactor

From the activity of the radionuclides listed above it can be seen that the total γ-activity of the reactor can be estimated to be above 1 x 10^9 Bq and that the major part of this radioactive activity is located in the recombiner and the core vessel. The dominant radionuclides are ^{137}Cs in the recombiner and the core vessel, ^{152}Eu in the graphite reflector and ^{60}Co in the rest of the reactor. From a volume point of view the most important reactor components that contribute to the waste generation from the decommissioning activities are the graphite reflector and the concrete shield.

3.3.2 Dose rates

Appendix III and Appendix D provide information on the dose rates of several components to be handled during the decommissioning of the research reactor. In addition, the detailed work plan for dismantling of the core vessel (see Appendix I) provides further information on the dose rates at the working place.

While the dose rates of the beam plugs and graphite stringers are not higher than a few µSv/h (except for one beam plug, for which the dose rate is 45 µSv at the inner part), the dose rate of the control rods at the inner part are reaching values up to 100 µSv/h.

It was determined that the dose rate in 50 cm distance of the recombiner is about 300 µSv and a simplified calculation results in a maximum dose rate at the surface of about 1.5 mSv/h (see Section 5.1.1). The dose rate of the reactor vessel is difficult to predict as during the characterization the graphite stringers and the reactor tank, which encase the reactor vessel, did not allow an easy measurement of the dose rate originating from the core vessel. An average value at the surface of 7 – 8 mSv/h was estimated. As the dismantling activities will be carried out from a distance the maximum dose rate at the working place is estimated to be up to 1 mSv/h.

3.4. OPERATIONAL HISTORY

3.4.1 Overview of the reactor's operation

The reactor was commissioned in August 1957 at a thermal power of 5 W. In spring 1959 the thermal power was increased to 2 kW until the reactor was shut down in 2001.

For the first approximately 20 years, the research reactor was used as a neutron source for following purpose:

(a) Neutron activation

When a material is exposed to neutrons, these neutrons can be absorbed by the nucleus of the material. If the isotopes thus formed are radioactive, the composition of the material and thus any impurities can be determined on the basis of the energy of the γ-radiation emitted during decay.

(b) Neutron radiography

In neutron radiography, neutrons are used to photograph objects that are to be subjected to non-destructive testing (e.g. X-ray).

For the last 20 years approximately, the primary objective of the research reactor was to serve for education of upper-secondary school students and students from the Technical University. The related teaching activities consisted of reactor experiments, reactor physics experiments (including neutron activation analysis) and neutron radiography. Approximately 600-700 upper-secondary school students and 6-12 university students have been educated per year.

Consequently, the reactor was operated at different outputs up to 2 kW for the first 20 years, while the output has been between 0 and 300 W in the subsequent period. For calculating the activity induced into structural materials and the radioactive inventory, the simplified output history shown in Figure 7. was used. The total generated energy is about 0.5 MWd.

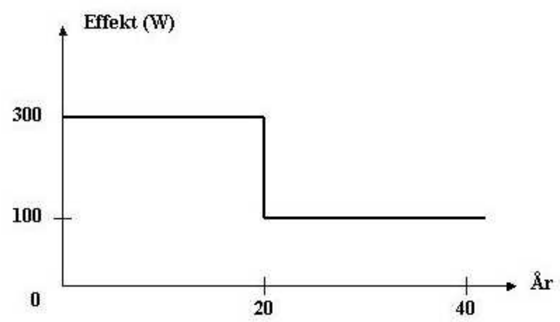

FIG. 7. Simplified output history ("Effekt" = Power, "År" = Year).

The only major modification that had been made to the research reactor was the installation of the cooling system in 1959. This change was made with a view to increasing output from 5 W to 2 kW. The modifications required to make this change were included in the original supply.

During the operational period of the research reactor some experimental setups were built at the reactor and dismantled after completion of the experiments, i.e.:

— *The Expo experiment*

This trial was an Exponential experiment with the aim to characterize moderator materials in terms of reactor physics (i.e. determination of diffusion-lengths and buckling). The research reactor acted as neutron source for a set-up placed on top of the reflector tank. The set-up consisted of a graphite block on which a tank filled with heavy water had been placed (see Figure 8). The equipment was dismantled and removed in 1962.

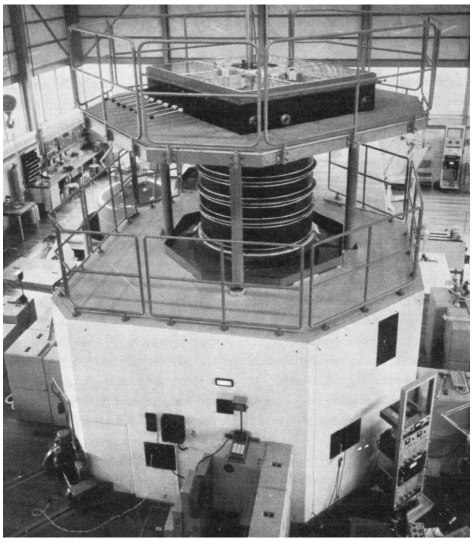

FIG. 8. Exponential set-up at the reactor.

— *The Pile Oscillator experiment*

This experiment was a reactor physics experiment to precisely determine the absorption cross-section of different materials and in particular materials with very low cross-sections. The reactor flux was made to oscillate using carefully controlled movements of a sample of the materials to be examined. Compensation for the oscillations was effected by means of a separate control rod, and the amplitudes measured formed the basis for a calculation of the cross-section. The experiment ended in 1970 and the set-up was partially dismantled in 1992.

3.4.2 Accidents and irregularities

No significant accidents were registered during the operation of the research reactor. Nevertheless, in the reactor hall there are smaller contaminated areas as a result of spills of core solution samples.

Furthermore, in 1972 an incident occurred in the reactor hall during work with a so-called “coated particle” from another reactor. The particle, which had a diameter of only few millimetres, was to be crushed in a mortar in preparation for further irradiation in the reactor, but fragments flew out of the mortar and some of them ended on the floor. There is reason to suppose that no remaining contamination from this incident will be found at the research reactor; however it cannot be completely excluded.[1]

3.4.3. Pre-decommissioning activities

During planning and preparation of the decommissioning in the post-operational phase some activities were performed to promote and facilitate the subsequent decommissioning. These activities were subject to the existing operation licence and performed in accordance to its terms and conditions.

(a) Tapping of fuel solution

The research reactor contained approximately 15.5 l of fuel in the form of about 5 kg uranyl sulphate dissolved in doubly demineralized water. The 5 kg of uranium comprised of 1 kg of ^{235}U and 4 kg of ^{238}U. Following more than 40 years' operation, the liquid also contained fission products and small quantities of transuranic elements.

— *Tapping*

The reactor fuel was tapped off according to operational procedures.

— *Storage of fuel*

The fuel solution was collected in four criticality safe stainless steel bottles and stored at a separate storage facility that is located at the research centre site.

Further details on the tapping of the fuel solution is presented in Appendix V.

(b) Decontamination of the primary system

After the fuel was tapped, the core system with the exception of the recombiner became partially decontaminated by flushing with demineralized water. The rinse water was collected in a 200 l stainless steel drum and sent to the radioactive waste treatment facility at the research centre for further processing.

Due to tapping of the fuel solution and partial decontamination of the primary system the radioactive inventory was reduced from an order of 100 GBq β/γ activity and 5 GBq α activity to a few GBq β/γ activity in the core solution and hardly any α activity.

(c) Removal of the Ra-Be source

The Radium-Beryllium source, which was used as a neutron source during the start-up of the research reactor, was removed prior to decommissioning. The source was located in a hollow in the graphite 220 cm into one of the experimental channels. The source activity was 3.7×10^9 Bq (0.1 Ci). The source was extracted by means of an electromagnet mounted, 2.5 m long rod and placed in a lead-

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[1] Note: during conduct of the decommissioning of the research reactor, which was used within this test case, some contamination was observed from spilled radioactive liquid during normal operation. This incident was not reported and thus not recognized in the documents on the operating history. Thus, as a "lesson learned" in real decommissioning safety assessments is that the assessment of the normal operation needs to be performed and periodically updated.

shielded container that was transferred to a storage facility for intermediate level radioactive waste at the research centre. The radiation level of 365 mSv/h was measured at a distance of about 2 cm from the source. Only insignificant personnel doses were registered as a result of this operation (max. 1 μSv).

3.5. DECOMMISSIONING ACTIVITIES AND TECHNIQUES

3.5.1 Decommissioning activities

During dismantling of the research reactor, it will be important for all connected supply services, such as electricity, water and sewers, to be disconnected correctly in accordance with applicable regulations. Since the reactor building does not have active drains, it will be also important to provide adequate cover to ensure that no active substances enter the ordinary drainage system. Therefore, as far as possible decontamination activities will be performed in external facilities. If decontamination activities might be necessary during dismantling activities no solvents or liquids are planned to be used in order to avoid spreading of radioactive material. Note that the core solution was trapped and decontamination of the primary circuit was performed after shut down of the research reactor but still under the operating licence (see Section 3.4.3).

(a) Emptying of cooling systems

Before the research reactor is dismantled, the secondary cooling system will be disconnected from the domestic water supply and emptied. Information on the cooling systems can be taken from Fig. 9 and Fig. 10.

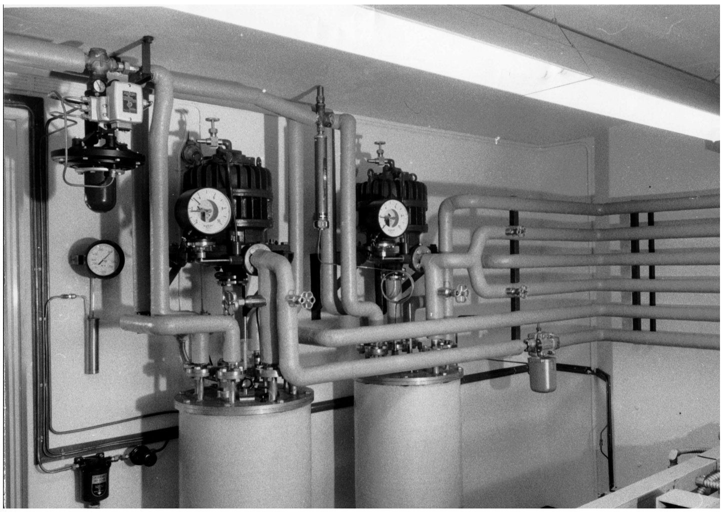


FIG. 9. Cooling systems.

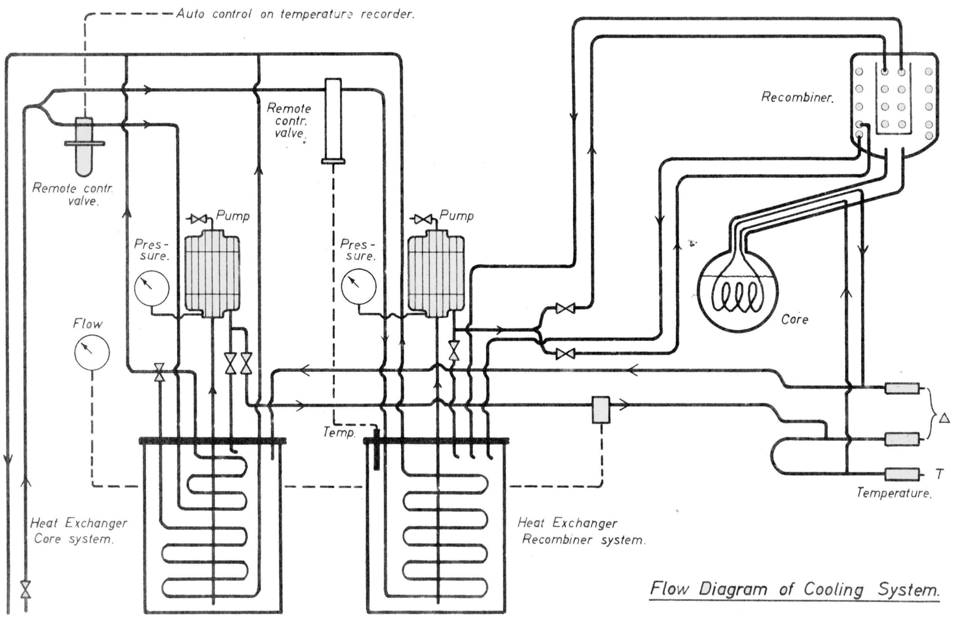
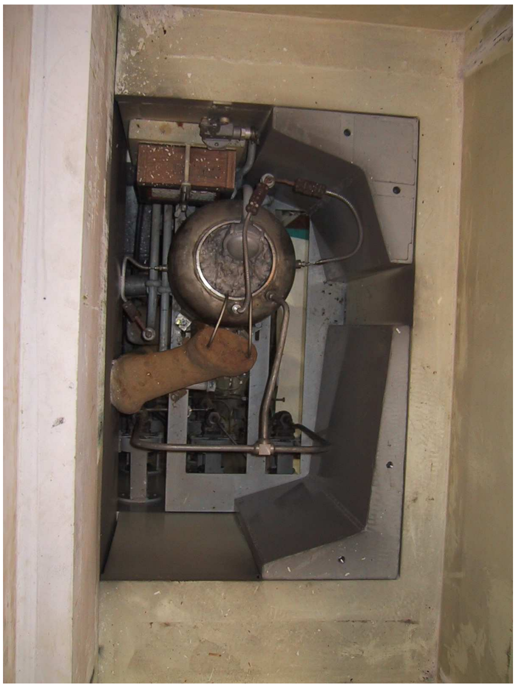


FIG. 10. Diagram of cooling systems.

(b) Exposing and removing the recombiner

The recombiner will be exposed by lifting away the concrete blocks at the top of the reactor block, using the hall crane. After removal of the concrete blocks, the radiation level are envisaged to be so high as to prevent employees from remaining in the area for brief periods of time.

Since the recombiner is contaminated on the inside with ^{137}Cs, it will be removed in one piece and transferred to a waste container. The recombiner has an external diameter of 270 mm, a height including connection pipes of approximately 500 mm, and weight of approximately 30 kg. In the bottom, it is connected to the core vessel by a flanged joint in the 2⅜" (60 mm) pipe (see Fig. 11) Because of the radiation of the recombiner the worker is planned to act from a distance. Especially, the recombiner will be lifted out using the crane of the reactor hall.



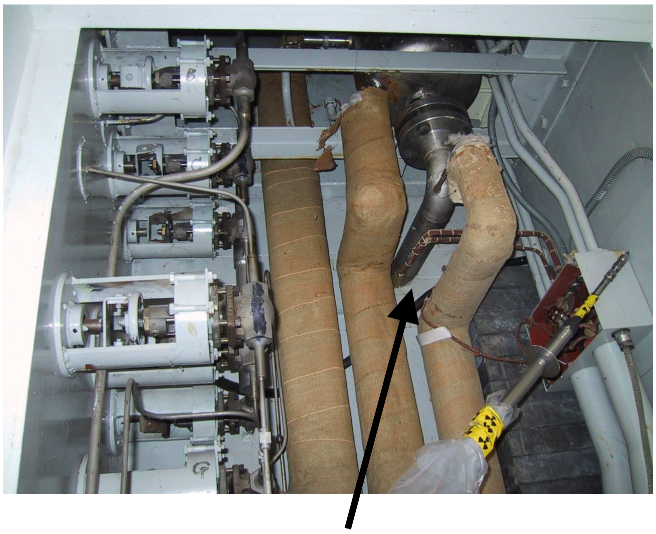


FIG. 11. Recombiner (left: top view; right: bottom view).

Removal activities are planned to be divided into the following processes:

— Removal of electricity cables to the heating element and signal wiring to the pressure transducer and thermal elements.

— Manual cutting of piping to the cooling spiral without use of a remote system but using a hydraulic scissors.

— Disconnecting the pipe to the recombiner. Closure of the connection pipe at the top of the recombiner by squeezing and cutting it using hydraulic cutting tool. The pipe connection to the core vessel will be opened at the flange at the bottom of the recombiner. Then cover plates will be mounted on the two flanges. To insert the cover plates, the recombiner will be lifted with help of the reactor hall crane, which will be connected to the recombiner using special connectors.

— The recombiner will be lifted up with the crane and transferred to a waste container (with a concrete shielding) for handling to the radioactive waste treatment facility.

(c) Dismantling of control and safety rods

The control and safety rods will be withdrawn horizontally in the control rod housing via the control rod lead-ins (see Figure 12).

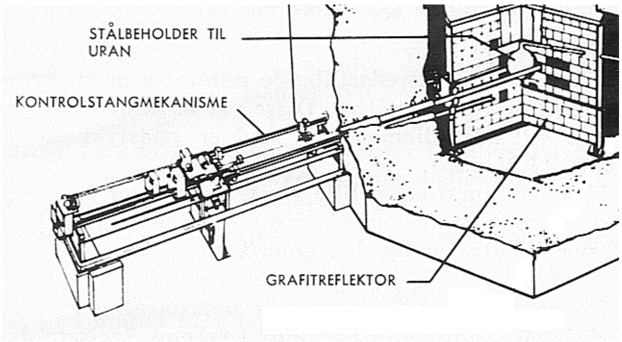


FIG. 12. Control and safety rods.

On the basis of the results from the calculation and measurements of the activity of the control rods, it is regarded possible to handle the absorber element during removal by means of the connection rod. After removal, the absorber element will be placed behind a lead wall in the storage cell so as to be separated from the connection rod.

Dismantling is planned to comprise of the following activities (see Fig. 13 and Fig. 14):

— Removal of signal wiring, electricity cables and worm gear;

— Removal of drop weights and shock absorbers on the safety rods and counting mechanisms on the control rods;

— Withdrawal of the rods and separation of these in the joints between connecting rods and absorber elements; and

— Removal of cover plates on the reactor block and withdrawal of the absorber.

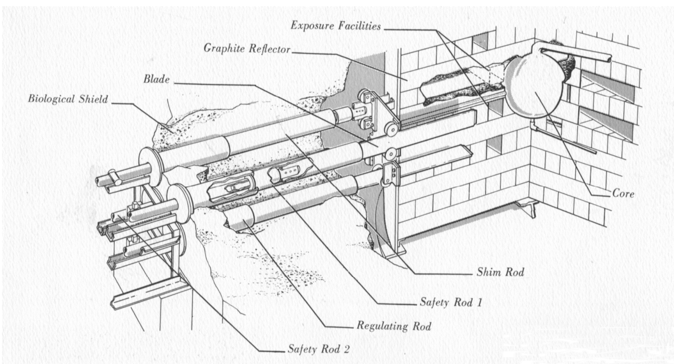


FIG. 13. Control rods – absorber element.

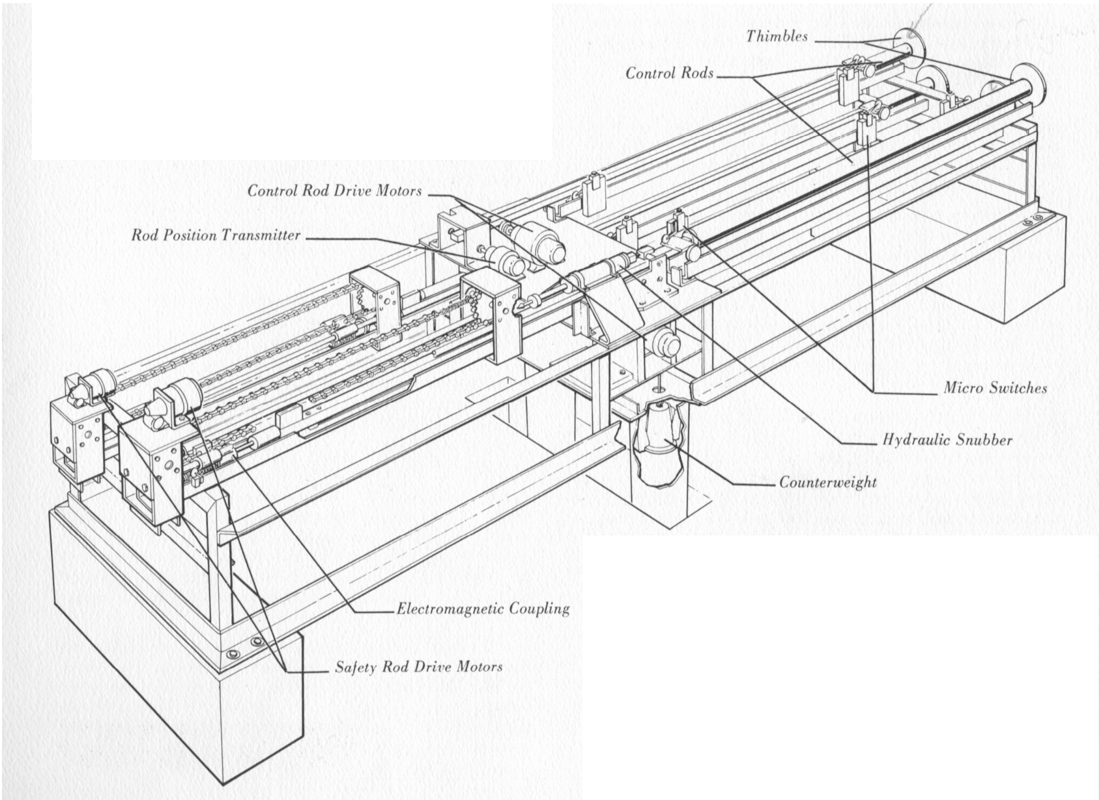


FIG. 14. Control rod drive.

(d) Dismantling of graphite, core vessel and reflector tank

The radiation level from the core vessel will require (see results of the TLD-measurements around the core vessel in Appendix IV) removal of the vessel and subsequent loading into a shielded waste drum. A related detailed work plan for the dismantling activities was developed and is provided in Appendix I.

Before the core vessel can be removed, the reflector lid will be removed. As it is not contaminated but only activated, it will be transferred immediately to the radioactive waste treatment facility at the centre after monitoring for contamination. The graphite reflector will become accessible and will be removed partially to allow free access to the core vessel. Depending on the radiation level, this will be carried out remotely by using suction packs.

Furthermore, the connection pipe to the recombiner, the drainage pipe in the bottom and the “Glory Hole” pipe will be disconnected. For this the drainage pipe will be squeezed and cut using a hydraulic scissors, while it will be examined whether the pipe through the “Glory Hole” can be withdrawn from the outside.

The core vessel has a volume of 17 l and a weight of approximately 6 kg. The reflector tank contains about 2 m^3 (about 4 100 kg) of graphite (see Figs 15 and 16).

FIG. 15. Core vessel.

FIG. 16. Reflector tank with reflector graphite.

Furthermore, during dismantling it will be assessed whether removal of the reflector tank and possibly taking it apart requires special measures to protect against workers' radiation and dust inhalation. If so, personal protective equipment will be worn to avoid any intake of possible dust from the graphite stringers.

(e) Dismantling of the other parts of the core system

The other parts of the core system, consisting of the filling system, the draining tank and the fission gas system, are considered only to be slightly contaminated and thus will be taken apart using normal health physical control measures.

(f) Dismantling of the cooling system

The piping and accessories outside the core vessel belonging to the cooling system are considered not to be contaminated and thus will be taken apart without any special measures (see also b) above).

(g) Demolition of the concrete shield

— *Cleaning*

After removal of the reflector tank the inside of the concrete shield will be cleaned by means of vacuuming and cleaning with wet cleaning tissues (i. e. no liquids are intended to be used), if deemed to be necessary.

— *Detailed characterization of activity content*

When the inside of the concrete shield becomes accessible a detailed characterization of its activity will be carried out to enable the demolition in a way that radioactive material (due to activation or contamination) above the clearance limits can be segregated. This radioactive material could be placed in a repository, while the remainder can be cleared. The characterization of the material from the concrete shield is important for minimizing the amount of decommissioning waste. The foundation of the block (4.4 x 4.4 x 2.6 m) contains approximately 100 t of heavy concrete and approximately 50 t of normal concrete. It is important to note that the clearance of material is intended to be carried out in the radioactive waste treatment facility at the research centre site.

— *Detailed plan for demolition*

Based on the characterization of the concrete shield, a detailed plan for demolition of the reactor block is planned to be developed.

Depending on the distribution of radioactive activity, different methods will be applied, ranging from grinding or chipping off activated surfaces to dry wire-cutting the concrete so as to separate radioactive from non-radioactive material.

In preparation of the cutting, a tent will be installed covering the reactor block to avoid any distribution of concrete dust through the reactor hall. In addition, the workers will wear appropriate personal protective equipment as air masks, overalls etc. to avoid any intake of radioactive contaminated dust.

A high performance suction and filtering system will be used to extract the dust generated at the cutting line of the dry wire-cutting system. In addition, a local ventilation system will be operated to filter the air inside the tent.

After completion of the cutting the inner surface of the tent will be cleaned by means of vacuuming and wiping with wet towels.

(h) Final decontamination and clearance of the reactor building

The reactor building is planned to be used for other purposes after completion of decommissioning. For that reason, the reactor hall will be finally decontaminated to ensure release from regulatory control and future use without restrictions. Therefore, possible contaminated parts of the floor, the related systems and the components of the reactor will be removed using conventional techniques if the related survey programme indicates contamination above the clearance levels [5]. Details on the clearance procedure intended to be applied in the Research Reactor Test Case can be found in Section 3.6.2.

Final measurements will be carried out after decontamination to document compliance with the clearance values.

3.5.2 Decommissioning techniques

During decommissioning of the research reactor only conventional techniques will be used which are routinely used in decommissioning projects world wide and thus they are proven to be fit for purpose. They will be compliant with regulatory requirements on tools and technique to be applied in decommissioning projects. For cutting of the systems and components, and especially of pipes, saws and hydraulic scissors will be used. No thermal cutting techniques are intended to be applied. The biological shielding will be dismantled using dry wire cutting with a high performance suction system at the cutting line to minimize potential release of dust to the work place.

Local handling of systems, components and other radioactive material (including waste) inside the reactor hall (from the reactor to waste drums or to the local storage cell) will be performed using the crane of the reactor hall. The maximum capacity of the crane is 2 Mg and is sufficient with regard to:

— The maximum mass of the individual waste piece (no component is expected to exceed 600 Mg as the biological shielding will be cut into pieces of no more than 500 Mg); and

— Waste drums (including additional shielding).

The crane will be subject to inspections by the Regulatory Body or by another competent authority (depending on the national regulatory system) in order to ensure compliance with the relevant national regulations on conventional safety of cranes.

3.5.3. Protective measures

Following the draining of the core solution, only moderate quantities of radioactive activity remained in the core vessel (mainly ^{60}Co) and in the recombiner (mainly ^{137}Cs), representing the main part of the radioactive inventory. The work procedures and protective measures will take this into account (see Section 7). Due to their radiological characteristics all other components will be handled unshielded, without exposing the decommissioning staff to any significant doses. Depending on the decommissioning activity to be performed appropriate personal protective equipment is envisaged for the workers. This will include:

— Direct reading and passive dose meter to determine the effect dose;

— Finger dose meters in those cases, in which the safety assessment during work planning indicated a potential exposure of the hand;

— Overalls, including gloves;

— Helmets; and

— In case of generation of dust or aerosols, breathing masks.

When external contractors will carry out decommissioning activities in the controlled area, for which the use of dose meters and medical examination of these employees are required by national regulations, the person in charge of the decommissioning work will not allow work to start until a personal radiation passport has been received from the external employee. The decommissioning project manager will ensure that:

(a) The external employee receives instructions and specific training with regard to the work in question;

(b) The external employee is given a personal direct reading dose meter and – if necessary – a finger dose meter, and

(c) The signed radiation passport is returned to the external employee’s company with information given to the employee and the company regarding the size of the dose measured.

The person in charge for radiation protection for the safety of workers during the implementation of decommissioning activities needs to ensure that dose limits are not exceeded.

The air will be monitored continuously during decommissioning. This will include control of the content of particular α/β activity by means of an air monitor (CAM), while surfaces must be monitored for contamination with α/β activity by means of wipe samples.

3.6. WASTE MANAGEMENT

The waste generated during decommissioning activities will be subject to a dedicated conventional and radioactive waste management. Clearance also will be applied in case of radioactive material. It has to be noted, that the clearance measurements for the radioactive material are not in the context of the Research Reactor Test Case, but that related measures to ensure effective minimization of radioactive waste will be implemented during decommissioning activities.

3.6.1 Radioactive waste management

The estimated total volume of the decommissioning waste expected to be generated from the reactor (mainly of concrete from shielding) is 60 m³. Most of it will be cleared for unrestricted use and will be disposed of as conventional waste after clearance. Consequently, it is expected that only very few concrete containers and ISO containers will be required for the collection, storage and handling of radioactive waste from the research reactor.

In total, the radioactive waste will mainly consist of heavy concrete (3.6 g/cm³), lead from shielding, metal and graphite. It will contain only small quantities of long lived radionuclides from material around the core. The main activity is estimated to as a result of ^{60}Co, ^{137}Cs and $^{152+154}Eu$, as presented in Table 3.1.

Before dismantling begins, an assessment will be made of whether the material generated will be treated as radioactive waste or non-radioactive waste. This assessment is necessary so as to avoid separating the waste more than once during the operation and also will minimize the doses for the workers involved.

Material generated during decommissioning, which potentially can be cleared, will be collected in grid boxes and will be cleared at the radioactive waste treatment facility.

The management of radioactive or potentially radioactive waste at the research reactor during will be based on the following basic safety criteria:

(a) Workers’ doses from decommissioning activities will be minimized in accordance with the ALARA principle (As Low As Reasonably Achievable).

(b) After dismantling, the radioactive waste will be placed in approved waste containers. While the waste container for the core vessel will be placed in the biological shielding (and thus will be craned after loading) all other waste containers will be placed on the top of the biological shielding for loading. When a container is completely filled it will be kept at a distance from the work site itself, or duly shielded, so that it does not contribute unnecessarily to the personnel doses. The radioactive waste will be temporarily stored in the reactor hall before handled to the radioactive waste treatment facility in a shielded storage cell at the southern end of the reactor hall (see Figure 3.4). When a waste container is filled, the contained waste will be recorded in files in accordance with the quality management system and national requirements and a description must be given of the content of each waste container.

(c) Containers temporarily stored in the storage cell will be transported from the reactor hall to the radioactive waste treatment facility as soon as the capacity of the storage cell and the progress of the decommissioning activities allow. The filled waste containers will be placed in a buffer hall at the radioactive waste treatment facility of the research centre site for further processing.

(d) The waste drums will be transported to the radioactive waste treatment facility according to the international standards [7] which are incorporated into the transport interface instructions of the research centre. Therefore the waste containers are envisaged to fulfil the requirements on:

— Contamination below 4 Bq/cm^2 for β/γ radionuclies;

— Contamination below 0.4 Bq/cm^2 for α radionuclides; and

— Dose rates below 5 µSv/h at the surface of the waste drum or of the concrete shielding of the waste drum (packages and overpacks of category I-WHITE).

As in the research reactor lead, different purities has been used for shielding purposes this lead will be re-used inside the waste drums as temporary shielding if needed to meet these conditions for transportation. The fulfilment of these criteria is planned to be checked by the radiation protection personal prior each transportation.

3.6.2 Clearance of the reactor hall

Clearance measurements for the building will be carried out after all radioactive contamination and activations – if any- have been removed to demonstrate that the relevant clearance values will be met. The measurements will be performed as a combination of contamination measurements with hand-held instruments and spectrometric measurements with Ge-detectors or NaI-detectors. The detectors will be compliant with the relevant national standards and requirements and will be subject to regular inspection and maintenance programmes and daily functional tests to ensure that the results are reliable and correct.

The rooms and structures in the building will be categorized into three classes, Class 1, Class 2 and non-classified, according to the likelihood of finding any radioactive contamination or activation. The categorization will be subject to agreement by the Regulatory Body:

(a) The surfaces in the set of rooms and structures with the highest likelihood of being contaminated will be classified as Class 1 and will be measured to coverage of 100%;

(b) Those in the middle group, Class 2, will generally be measured to coverage of 10-50%; and

(c) A few random measurements will be carried out on the non-classified surfaces that had no or very little contact with radioactive materials.

The reactor hall, the counting laboratory in the basement, the locker room and toilet at the ground floor are classified as Class 1. The stairway, the rooms for heat control in the eastern part of the basement and the dark room are classified partly as Class 1 and partly as Class 2. All remaining rooms, including the control room and the offices are classified as Class 2. There were no non-classified rooms in the building.

Ge-detectors are intended to be used in larger rooms as one or two measurements can measure the surface-contamination in the whole room. Ge-detectors can also measure the γ-emitting radionuclides that may have penetrated into the floor or walls. Furthermore, gamma spectrometric measurements can determine the radionuclide composition of γ-emitters. The measurement results with the Ge-detectors are planned to be analysed by means of the ISOCS software. In the reactor hall the brick walls and the window sills will be measured with contamination monitors. The concrete wall to the west will be measured with a contamination monitor up to a height of 2 m. above the floor. The rest of the walls will then be measured with Ge-detectors. It is assumed conservatively that all the radioactive activity seen by the detector is located in one square meter that is positioned farthest away from the detector. Most likely, however, the contamination will be evenly distributed on the wall.

Contamination of the ceiling is intended to be measured with Ge-detectors. The crane will be measured with both contamination monitors and Ge-detectors. In order to distinguish between potential ^{137}Cs-contamination originating from the operation of the research reactor and the ^{137}Cs originating from fallout from the nuclear weapons tests and from the Chernobyl accident, background spectra will been taken in a number of locations similar to those surrounding the research reactor.

(a) Floor, channels and pits in the reactor hall

The floor will be measured with both Ge-detectors and contamination monitors.

(b) Concrete below the reactor

The "crater" left open after the removal of the biological shield of the reactor will be measured with a Ge-detector, combined with analysis of drill-core samples to determine the decrease of the activity concentration as a function of depth.

(c) Heating and ventilation system in the reactor hall

The ducts for the heating system are not accessible. It is assumed that if any contamination is found in the ducts, it can be detected behind the outlet grills at the southern end of the hall. Dust behind the ventilation grills will be scraped off and analysed by gamma-spectrometry in the laboratory. Similarly, smear tests will be taken from the inside of the ventilation channel.

(d) Other rooms in the building

Although most of the rooms outside the reactor hall are classified as Class 2 areas, for the purpose of the Research Reactor Test Case they will be measured to a near 100% coverage. The counting laboratory in the basement will be measured by means of both contamination monitors and spectrometers, because some contamination has been found earlier (and removed) here. All other rooms will be measured with contamination monitors only.

3.7 SUPPORTING FACILITIES

The decommissioning of the research reactor relies mainly on existing facilities at the research centre to manage radioactive material and to clear radioactive material. These facilities are subject to individual existing licences which allow treatment and storage of radioactive material or clearance of material originating from the decommissioning of the research reactor.

In addition a new facility, a storage cell, for temporary storage of radioactive material inside the reactor hall is needed, which will be set up in close distance to the biological shielding (see also Section 3.2).

3.8. END-STATE

When decommissioning is completed:

- The research reactor (including the biological shielding) will be removed completely; and
- All remaining systems (e.g. ventilation system, heating system) and the reactor hall are compliant with clearance criteria and can be release for unrestricted use.

4. HAZARD ANALYSIS

4.1. HAZARD IDENTIFICATION

Relevant hazards within this decommissioning of the research reactor are primarily due to radioactive material contained in components of the primary circuit (e.g. core vessel, recombiner, connecting pipe), and activations of the reflector vessel, the graphite and the biological shielding (if any).

In addition, conventional hazards due to the use of asbestos, cadmium and lead need consideration.

4.2 APPROACHES TO HAZARD IDENTIFICATION

4.2.1. Choice of the approach

As presented in the main report, there are various approaches to performing hazard identification (see Section 3.3.2, “Approaches to Hazard Identification”). These approaches are discussed below as well as their suitability for application to the Research Reactor Test Case:

- *Evaluation based on existing safety assessment for the research reactor*

  The safety assessments which exist for operation of the research reactor are based on scenarios which are not relevant for decommissioning. Therefore, existing safety assessments cannot be used in this context.

- *Evaluation based on past operational experience*

  Evaluation of past operational experience is of limited use in this context as no major incidents occurred during the operation of the research reactor. All existing data known from the operational period are presented in Section 3, “Description of the Facility and Identification of Controls”.

- *Use of checklists*

  The generic checklist supplied in Appendix I of the main report is used for identifying the hazards for the research reactor. Additional considerations are applied using the “What-If-Technique” described below.

— *Hazard and operability study (HAZOP)*

This approach is not well suited to the hazard identification of a nuclear facility with less complexity and low radioactive inventory like a small research reactor. Such a technique can best be applied to larger nuclear facilities, in particular those where there are interdependencies between various parts or processes where the failure or malfunction of one part may affect others.

— *What-if ? technique*

The approach of brainstorming to check whether any item not covered by the generic checklist need to be additionally considered is used to ensure completeness of the hazard identification.

4.2.2. Results of the hazard identification

Hazards are identified using the generic checklist in combination with the “What-If? technique”, taking into account additional hazards from the following areas:

— External factor influences;

— Operator error and other human factors;

— Equipment/instrumentation failure;

— Utility failures; and

— Integrity failure.

Table 1 shows the result of the hazard analysis pertinent to the research reactor and lists hazards and related initiating events of relevance for normal and accidental scenarios.

TABLE 2. IDENTIFIED HAZARDS, INITIATING EVENTS AND RELEVANT SCENARIOS

Hazards	Relevant for Workers during Planned Conditions	Relevant for Workers during Accident Conditions	Relevant for Public During Planned Work	Relevant for Public during Accident Conditions
	Yes/No	*Yes/No*	*Yes/No*	*Yes/No*
Radiological hazards				
Direct radiation sources	yes	yes	no	no
Improper removal of shielding	yes	no	no	no
Radioactive material, incl. form: (solid, liquid, gaseous)	yes	yes	yes	yes
Criticality	no	no	no	no
Contaminated liquid, material	no	yes	no	yes
Other radioactive sources (smoke detectors, lightning rods)	no	no	no	no
Fire/explosion hazards				
Oxygen	no	no	no	no
Sodium	no	no	no	no
Explosive substances	no	no	no	no
Flammable gases (e.g. oxyacetylene, propane gas), liquids, dust	see “Combustible/inflammable materials”			
Combustible / inflammable materials (for the RR: graphite; wooden floor)	no	yes	no	yes
Compressed gases	no	no	no	no
Hydrogen generation	no	no	no	no
Overheating or fire, caused by e.g. portable heaters, overload of electrical circuits, application of cutting techniques	yes	yes	no	no
Electrical hazards				
High voltages	no	no	no	no

Hazards	**Relevant for Workers during Planned Conditions**	**Relevant for Workers during Accident Conditions**	**Relevant for Public During Planned Work**	**Relevant for Public during Accident Conditions**
	Yes/No	*Yes/No*	*Yes/No*	*Yes/No*
Power overload and shortcuts, power failures	no	yes (power failure during crane operation with source exposed)	no	no
Inadequately disconnected circuits/prevention against inadvertent connection	yes	no	no	no
Non-Ionizing Radiation Hazards				
Non-Ionizing Radiation Sources, incl. lasers	no	no	no	no
Electromagnetic radiation (e.g. microwaves)	no	no	no	no
High Intensity Magnetic Fields	no	no	no	no
Chemical/toxic hazards				
Chemotoxic material	no	no	no	no
Spills	no	no	no	no
Chemicals (aggressive chemicals) Remark: no acid based batteries available at the reactor	no	no	no	no
Accidental mixing/combination of chemicals (e.g. in sewage systems, in decontamination work, etc.)	no	no	no	no
Asbestos and other hazardous materials, like lead or beryllium	yes (asbestos from insulation of pipes at control rod house; cadmium at fuel drain tank)	no (lead to be removed before introducing fire hazards e.g. from welding)	no	no
Pesticide use	no	no	no	no
Biohazards	no	no	no	no

Hazards	Relevant for Workers during Planned Conditions	Relevant for Workers during Accident Conditions	Relevant for Public During Planned Work	Relevant for Public during Accident Conditions
	Yes/No	*Yes/No*	*Yes/No*	*Yes/No*
Physical hazards				
Kinetic energy	no	no	no	no
Potential energy (springs, Wigner energy in graphite)	no (no Wigner energy produced because of low flux of fast neutrons)	no	no	no
Degraded or degrading structures, systems and components	no	no	no	no
Steam	no	no	no	no
Temperature extremes (high temperatures, hot surfaces, cryogenics)	no	no	no	no
High pressure (pressurized systems, compressed air)	no	no	no	no
Working environment hazards				
Working at heights (e.g. ladders, scaffolding, man baskets)	yes (scaffolding; top of biological shield)	no	no	no
Excavations, formation of underground cavities (subsidence) from rain, waste degradation etc.	no	no	no	no
Vehicle traffic	yes (in case of waste transports)	no	no	no
Heavy lifts, material handling, heavy equipment, manual lifting, overhead hazards, falling objects, cranes	yes (lifting, cranes)	yes	no	no
Inadequate illumination	no	no	no	no
Inadequate ventilation	no	no	no	no

Hazards	**Relevant for Workers during Planned Conditions**	**Relevant for Workers during Accident Conditions**	**Relevant for Public During Planned Work**	**Relevant for Public during Accident Conditions**
	Yes/No	*Yes/No*	*Yes/No*	*Yes/No*
Noise (high noise areas and tools)	yes (concrete demolition; hearing protection)	no	no	no
Dust	yes (concrete demolition; respiratory protection)	yes	no	no
Pinch points, sharp objects	yes (cutting operations)	no	no	no
Confined space	no	yes (in recombiner vault)	no	no
Dangerous equipment, e.g. power tools, compressed gas cylinders, welding and cutting, water jet cutting / decontamination, abrasive decontamination techniques, grinding, sawing	yes (various tools)	yes	no	no
Remote work area	no	no	no	no
Obstruction of passageways or exits	no	no	no	no
Human/organizational hazards*)				
Human error	not applicable	not applicable	not applicable	not applicable
Safety culture aspects	not applicable	not applicable	not applicable	not applicable
Assigning inadequate training for work steps	not applicable	not applicable	not applicable	not applicable
Assigning inadequate protective measures for work steps	not applicable	not applicable	not applicable	not applicable
External hazards / initiating events				
Ambient temperature extremes	no	no	no	no
Airplane crash	not to be taken into account			
Storm and adverse weather conditions	no	no	no	no

Hazards	Relevant for Workers during Planned Conditions	Relevant for Workers during Accident Conditions	Relevant for Public During Planned Work	Relevant for Public during Accident Conditions
	Yes/No	*Yes/No*	*Yes/No*	*Yes/No*
Earthquakes	not to be taken into account			
Flooding	-	no (site sufficiently high above sea level)	-	no
External explosions and fires	-	no (no other buildings in neighbourhood)	-	no
Other Hazards	no	no	no	no
Degraded / corroded barriers, ageing of materials	no	no	no	no
Unknown or unmarked materials	no	no	no	no
Spills (due to decommissioning activities)	no	yes (see contaminated liquids)	no	no
Malfunction of safety relevant systems	no	no	no	no

*) For the purpose of this test case it is assumed that the human organization and factors do not impose any hazards relevant for the safety assessment.

The positive entries in Table 2 have a different relevance for the safety assessment, ranging from situations or scenarios with minor consequences to those hazards which are limiting in the framework of the safety assessment for the research reactor. Therefore, the positive entries are discussed and put into perspective:

(a) Direct radiation sources

As described in Section 3, the radionuclide inventory mainly consists of radionuclides which contribute to external irradiation (^{60}Co, ^{137}Cs). External irradiation is the only relevant pathway for workers exposure during normal decommissioning conditions (see Sections 4.3.1 and 5.1.1). It is also relevant for accident scenarios for workers and therefore direct radiation sources are taken into account (see Sections 4.3.3 and 5.2.1). However, inhalation proves to be the more relevant exposure scenario.

(b) Improper removal of shielding

The procedures for performing the decommissioning activities have been carefully evaluated to ensure that large sources are always well shielded or that they are exposed only shortly during handling with the necessary temporary shielding. Improper removal of shielding has therefore been dealt with in the operating procedures. Improper shielding conditions will, in addition, be readily detected by the health physics personnel during routine measurements. As the overall dose rate from the parts of the research reactor is small, improper removal of shielding is not considered relevant for being included in the safety assessment for accident scenarios as a separate case. It is covered by the other cases which are being evaluated (see Sections 4.3.3 and 5.2.1).

(c) Radioactive material (solid, liquid, gaseous)/contaminated liquids

In the present context, radioactive materials with relevance to the safety assessment mainly consist of the contamination present on the inner surfaces in the form of dust or liquids. This issue has been taken into account when designing work safety measures, like prescription of wearing respiratory protection during those work steps where contamination could be mobilized and be suspended into the breathing air of personnel. Therefore, doses from inhalation are not expected to be relevant for the safety assessment of the research reactor decommissioning during normal working conditions (see Sections 4.3.1 and 5.1.1).

For the analysis of accidents, however, it is conservatively assumed that the inner contamination will accidentally be mobilized and that the worker being affected is not wearing respiratory protection equipment (see the scenario description in Section 4.2.3 and 5.2.1). In addition, the fact that some contamination which can be mobilized is also the reason for assuming exposure of members of the public via airborne pathways during normal operating conditions and accident situations as outlined in Sections 4.3.2 and 4.3.4 and Sections 5.1.2 and 5.2.2.

(d) Flammable gases (e.g. oxyacetylene, propane gas), liquids, dust

For normal operation, dust from dismantling of graphite is a hazard which is taken care of by the implementation of industrial safety measures. Graphite dust is, however, also regarded as one possible initiating event as it may be set on fire. But, with respect to the low total generated energy of 0.5 MWd within 44 years of research reactor operation the Wigner-Energy deposed by high energy neutrons can be regarded to be too low to allow a self-ignition of the graphite bricks during handling. Thus ignition of graphite dust requires some external fire source and cannot result from Wigner-Energy. Nevertheless, it is regarded as part of an initiating event that an external source of fire will exist leading to a fire causing accidental release of radioactive material as described in Sections 4.3.4 and 5.2.2.

(e) Combustible/inflammable materials

For the research reactor, only the graphite and the wooden floor fall into the category of combustible/inflammable materials. These materials do not pose any hazards during normal decommissioning conditions. However, they are expected to contribute to a fire causing accidental release of radioactive material as described in Section 5.2.2.

(f) Overheating or fire

Overheating or fire can be caused by different events, e.g. portable heaters, overload of electrical circuits and application of cutting techniques. The application of certain cutting techniques may be the initiating event for a fire causing accidental release of radioactive material as described in Section 5.2.2.

(g) Power overload and shortcuts, power failures

A power failure or a shortcut does not cause a hazard during normal operations, as the safety of operation does not depend on the continuous operation of any electrical powered device. If a crane lifting an unshielded source (activated and contaminated items) accidentally ceased operation due to a power failure so that the source remains unshielded, this could cause a hazard from external irradiation to the personnel. As, however, the personnel will leave the area immediately and return only after power has been restored and the source can be lowered into its shielded container, this scenario is mainly comparable to the external exposure of the worker during normal operation in Sections 4.3.1 and 5.1.1.

(h) Inadequately disconnected circuits/prevention against inadvertent connection

This is taken into account in the operation procedures for planned decommissioning activities. Therefore, it is not expected to form an initiating event for accidents during decommissioning.

(i) Asbestos and other hazardous materials, like lead or beryllium

Hazardous materials like asbestos from the insulation of pipes at the control rod house and cadmium at the fuel drain tank are present. Their position is well known, and the operating procedures take the presence of these materials into account. They are handled appropriately and do not pose an undue risk. The presence of these materials is not regarded as an initiating event for or forming part of any fault sequence leading to radiological consequences.

(j) Working at heights (e.g. ladders, scaffolding, man baskets)

Working at heights takes place only to a very limited extent. Scaffolds are used for working on or near the top of the biological shield. For this purpose appropriate industrial safety measures have been put into place to avoid accidents. Therefore, this item is not regarded as relevant for analysis of accidental working conditions as it bears no radiological consequences. Working at heights is also not seen as an initiating event for a fault sequence leading to radiological consequences.

(k) Material handling

Material originating from dismantling activities will be temporarily stored in the reactor hall until the material will be transported to the radioactive waste treatment facility. Related vehicle traffic of very low frequency will be taken into account during the development of the work procedures and during work control. Therefore no radiological risk during decommissioning is envisaged.

(l) Heavy lifts, material handling, heavy equipment, manual lifting, overhead hazards, falling objects, cranes

For normal operation, these hazards are duly covered by the operating procedures. For example, the crane is subject to regular inspection and maintenance programmes according to national requirement related to conventional work safety and will be operated by skilled personnel, workers are forbidden to stay beneath lifted loads etc. These hazards are therefore not explicitly addressed in the hazard analysis for normal decommissioning conditions. With respect to accident conditions, however, a drop of a waste container or a contaminated part of the research reactor which has been lifted may cause spread of contamination. Drop of loads is therefore included as an initiating event in the analysis performed in Sections 4.3.3 and 5.1.2.

(m) Noise (high noise areas and tools)

For some of the dismantling steps, tools creating high noise levels are used, for example for decontamination of concrete surfaces or during concrete demolition. For this reason protective measures during normal operation have been taken (hearing protection). Noise is not regarded as an initiating event for or forming part of any fault sequence leading to radiological consequences.

(n) Dust

Larger amounts of dust are created mainly during concrete demolition. For this reason protective measures during normal operation have been taken (respiratory protection).

(o) Pinch points and sharp objects

These may occur mainly during cutting operations. Normal industrial safety procedures are in place to prevent injuries caused by pinch points and sharp objects. Pinch points or sharp objects are not regarded as forming part of any fault sequence leading to radiological consequences.

(p) Confined space

The only confined space which is of relevance here is the recombiner vault. During normal operation, working in this area does not present any particular hazard. For the analysis of accident conditions, however, this area is chosen as the place where the highest amount of loose contamination could become mobilized, leading to doses from inhalation to a worker present in the recombiner vault. Working in this confined space is, therefore, not the cause for an accident scenario (i.e. does not form an initiating event), but contributes to such a scenario as outlined in Sections 4.3.3 and 5.2.1.

(q) Dangerous equipment, e.g. power tools, compressed gas cylinders, welding and cutting, water jet cutting/decontamination, abrasive decontamination techniques, grinding, sawing

During normal operation, such techniques are applied only by skilled personnel. It is thus ensured that they do not pose a risk to the personnel during normal operation.

(r) Airplane crash

An airplane crash is not considered as a design base accident for the research reactor.

(s) Malfunction of safety relevant systems

Safety relevant systems are operated with regard to the monitoring of air born radioactive material. Further systems, especially the ventilation system are not safety relevant. In case of the demolition of the reactor block, a high performance suction system and a local ventilation system will be used. It is envisaged that in case of any failure the dismantling activity will be stopped so that no further analysis is required. As part of the protective systems that are planned during decommissioning, personal protective equipment will be worn by workers. Failures need to be taken into account in the safety assessment, especially related to the risks due to the dust from graphite handling and concrete cutting.

4.3. PRELIMINARY HAZARD ANALYSIS AND SCREENING

The following scenarios provide upper and conservative estimates of potential doses that could arise from normal and from accidental conditions both for workers and for members of the public. This preliminary analysis serves to put the potential hazards from the research reactor into perspective and to justify the choice of the level of detail in which the detailed hazard analysis is carried out in Section 5 (graded approach). The Section 4.3.1 – 4.3.4 contain rough estimates of upper bounds on doses for workers and the public. The influence of the waste management is preliminary analysis in Section 4.3.5. The most relevant scenarios are summarized in Section 4.3.6.

4.3.1 Preliminary analysis for workers under normal conditions

With regard to the normal conditions the following estimates on the most significant contributions to the workers' exposure, taking into account the results of the hazard identification in Section 4.1:

(a) External exposure due to the course of work

— A conservative estimate is based on the detailed information on the course of work in Appendix I by summing over all external exposure, independent from the affected worker. The estimated total exposure is about 550 µSv for the preparation and removal of the core vessel, which can be regarded as the main contributor to the external exposure. Thus, the effective dose can be regarded to be low, but a more detailed analysis is appropriate as the contribution from the removal of the recombiner is not taken into account (the work break down considers only the removal of the reactor graphite, the reactor vessel and the reflector tank).

— The possible scenario that could lead to maximum equivalent dose for the hand of a worker needs to be assessed. This means that those components with the highest dose rates, i.e. the reactor core vessel and the recombiner, need to be addressed. For dismantling and removal of the reactor core vessel it is assumed that all relevant activities performed in near distance are performed at the level of the dose rate at contact. The time assumed can be obtained from Appendix I and is roughly equal to 45 min. Taking into account a dose rate of 7 to 8 mSv/h the resulting maximum dose is 6 mSv (see Appendix D). The contribution from disconnecting and removing of the recombiner with a dose rate at contact of 1.5 mSv/h is equal to 0.5 mSv as the relevant time for conduct of the work is about 20 min. In total the contribution from both components is less than 7.5 mSv. Another contribution to the workers exposure during decommissioning is the handling of the graphite stringers. On the basis of the assumed total processing time of 13 h (see Appendix I) and the maximum dose rate of 80 µSv/h the estimated contribution is about 1 mSv. Thus in total the equivalent dose for a hand of a potential worker, who is involved in all activities is less than 10 mSv, which is below the dose limit of 500 mSv. Taking into account, that most activities will be performed without contact the real dose will be much lower than the estimated value.

(b) Removal and handling of the graphite stringers

The graphite stringers are removed remotely and in addition are in a good physical shape. Thus, it is not expected that there will be generation of graphite dust which might be incorporated in the hazard analysis. In addition, the inspection and radiological measurement on activations will be performed manually but with personal protective equipment as overalls and gloves. Therefore this is only of relevance for the accident worker scenario.

During removal of the connecting pipe between recombiner and core vessel some drilling in the reactor graphite will be performed. The inhalation of graphite dust can be neglected in the Research Reactor Test Case as:

— A suction system will be used to extract any graphite dust; and

— The workers will wear personal protective equipment including dust masks.

(c) Dry cutting of the biological shielding

The dry cutting will be performed with local ventilation in a dedicated tent to avoid contamination of the reactor hall. In addition, workers will wear personal protective equipment including breathing masks or respiratory systems, so that no incorporation of dust will occur.

4.3.2. Preliminary analysis for members of the public from normal decommissioning conditions

Exposure from the normal operation of the research reactor to members of the public can only occur from gaseous discharge of radioactive material. There are no liquid discharges of radioactive material from the decommissioning work as no wet dismantling techniques will be used. The radioactive inventory of the research reactor is so small that external exposure of the personnel of the research centre, which is not involved in the decommissioning, can be neglected due to the large distance of their daily ways from the research reactor.

A preliminary analysis of doses from gaseous discharge of radioactive material consists of the following four steps.

(a) Determination of the source term

During dismantling of the research reactor, inner surfaces are exposed and building surfaces are decontaminated, leading to a mobilization of radionuclides into the research reactor’s atmosphere. A part of this activity is discharged via the stack at a height of approx. 15 m. It is a conservative assumption that 1 % of the total radioactive inventory will be discharged in this way over 1 a. An upper estimate for the total activities and for the resulting discharge rates is contained in the following Table 3.

TABLE 3. ESTIMATE OF THE DISCHARGE RATE

	^{137}Cs	**^{90}Sr**	**^{60}Co**	**^{154}Eu**
Total radioactive activity [Bq]	3×10^9	1×10^7	2×10^8	1×10^8
Discharge rate [Bq/y]	3×10^7	1×10^5	2×10^6	1×10^6
Discharge rate [Bq/s]	0.95	0.003	0.06	0.03

It has to be noted, that ^{14}C in graphite is of no concern to the public during normal operation due to low mobility.

(b) Modelling of the dispersion in the environment

A model for the atmospheric dispersion of the radionuclides from the point of discharge into the direction of the houses next to the research establishment where the research reactor is located is taken from Section 3.5 of IAEA Safety Reports Series No. 19 [8]. The ground level air concentration can be calculated as follows:

$$C_A = \frac{P_p \cdot B \cdot Q_i}{u_a} \qquad (1)$$

where:

C_A ground level air concentration at downwind distance x (Bq/m³);

P_p fraction of the time the wind blows towards the receptor of interest;

B dispersion factor with building wake correction (1/m²);

Q_i average discharge rate for radionuclide i (Bq/s); and

u_a geometric mean of the wind speed at the height of discharge representative of 1 m/s.

Taking the conservative assumption that the wind always blows into the direction of the nearest houses (P_p = 1), assuming an average wind speed of 1 m/s and using a factor B of 5 x·10^{-4} m^{-2} corresponding to the geometrical conditions, the following air concentrations at the receptor point are calculated and presented in Table 4.

TABLE 4. GROUND LEVEL CONCENTRATION IN AIR

	^{137}Cs	^{90}Sr	^{60}Co	^{154}Eu
Ground level concentration in air [Bq/m³]	4.8 x 10^{-4}	1.6 x 10^{-6}	3.2 x 10^{-5}	1.6 x 10^{-5}

The ground deposition of the radionuclides is calculated as follows:

$$\dot{d}_i = (V_d + V_w) \cdot C_A \qquad (2)$$

where:

$\dot{d}_i$ total daily average deposition rate on the ground of a given radionuclide i from both dry and wet processes, including deposition either on to impervious surfaces or on to both vegetation and soil (Bq·m^{-2}·d^{-1});

V_d dry deposition coefficient for a given radionuclide (m/d); and

V_w wet deposition coefficient for a given radionuclide (m/d).

A conservative estimate for the sum of both deposition coefficients is 1 000 m/d. Assuming that the process to last for 1 y, the ground surface concentration values are calculated and presented in Table 5.

TABLE 5. GROUND SURFACE CONCENTRATION

	^{137}Cs	^{90}Sr	^{60}Co	^{154}Eu
Ground surface concentration [Bq/m²]	174	0.6	12	6

(c) Modelling of inhalation and external exposure from ground deposits

The doses from inhalation are calculated as the result of breathing rate, exposure time, inhalation dose coefficient and radionuclide concentration in the air. Taking into account the exposure time during the entire year (8 760 h/y), an average breathing rate for adults of 0.95 m³/h and for infants (0-1 y) of 0.16 m³/h, the doses presented in Table 6 were calculated.

TABLE 6. DOSE COEFFICIENTS FOR INHALATION

	^{137}Cs	^{90}Sr	^{60}Co	^{154}Eu
Dose coefficient inhalation, adults [Sv/Bq]	3.9 x 10^{-8}	1.6 x 10^{-7}	9.2 x 10^{-8}	5.3 x 10^{-8}
Dose from inhalation, adults [Sv/y]	1.5 x 10^{-7}	2.1 x 10^{-9}	2.4 x 10^{-8}	7.0 x 10^{-9}
Dose coefficient inhalation, infants (0-1 y) [Sv/Bq]	1.1 x 10^{-7}	4.2 x 10^{-7}	3.1 x 10^{-8}	1.6 x 10^{-7}
Dose from inhalation, infants (0-1 y) [Sv/y]	7.3 x 10^{-8}	9,3. x 10^{-10}	1.4 x 10^{-9}	3.6 x 10^{-9}

The effective doses resulting from inhalation are 0.2 µSv/y for adults and 0.1 µSv/y for infants.

The doses from external irradiation from the activity deposited on the ground is calculated as the product of the ground surface concentration values and the dose coefficients for surface deposits. This results in the estimated external exposure presented in Table 7.

TABLE 7. DOSE DUE TO EXTERNAL EXPOSURE DUE TO GROUND DEPOSIT

	^{137}Cs	^{90}Sr	^{60}Co	^{154}Eu
Dose coefficient for ground deposit [(Sv/y)/(Bq/m²)]	1.8 x 10^{-8}	3.5 x 10^{-9}	7.5 x 10^{-8}	3.8 x 10^{-8}
Dose from external irradiation ground deposit [Sv/y]	3.1 x 10^{-6}	2.0 x 10^{-9}	8.7 x 10^{-7}	2.2 x 10^{-7}

The resulting effective dose from external irradiation from ground deposits is 4 µSv/y.

(d) Modelling of the secondary ingestion via radioecological pathways

For modelling secondary ingestion, only vegetable consumption is taken into account as the houses next to the research reactor only have gardens allowing production of vegetables but no cultivation of corn or rearing of cattle. The contamination of vegetation is calculated as follows:

$$C_{v,i,1} = \frac{\dot{d}_i \alpha \left[1 - \exp\left(-\lambda_{E_i^v} t_e\right)\right]}{\lambda_{E_i^v}} \qquad (3)$$

Where

$C_{v,i,1}$ is measured in Bq/kg fresh matter for vegetation consumed by humans

$\dot{d}_i$ total daily average deposition rate on the ground of a given radionuclide *i* from both dry and wet processes (Bq·m^{-2}·d^{-1}), see above α is the fraction of deposited activity intercepted by the edible portion of vegetation per unit mass (or mass interception factor, m²/kg) as the result of both wet and dry deposition processes; here 3 m²/kg

$\lambda_{E_i^v}$ is the effective rate constant for reduction of the activity concentration of radionuclide *i* from crops (d–1), where $\lambda_{E_i^v} = \lambda_i + \lambda_w$

t_e is the time period that crops are exposed to contamination during the growing season (d), here 60 d

λ_w is the rate constant for reduction of the concentration of material deposited on the research reactor's surfaces owing to processes other than radioactive decay (d^{-1}), here 0.05 d^{-1}

λ_i is the rate constant for radioactive decay of radionuclide *i* (d^{-1}).

Using the deposition rate calculated above, the ingestion dose coefficients and a conservatively high ingestion rate of 200 kg/y for vegetables and other crops grown in the own garden, the following results were obtained and presented in Table 8.

TABLE 8. INTERNAL EXPOSURE DUE TO INGESTION

	^{137}Cs	**^{90}Sr**	**^{60}Co**	**^{154}Eu**
Activity concentration in vegetation [Bq/kg]	2.7 x 10^{1}	9.0 x 10^{-2}	1.8	9.0 x 10^{-1}
Dose coefficient for ingestion [Sv/Bq]	1.3 x 10^{-8}	2.8 x 10^{-8}	3.4 x 10^{-9}	2.0 x 10^{-9}
Dose from ingestion of 200 kg/a crops and vegetables [Sv/y]	7.0 x 10^{-5}	5.1 x 10^{-7}	1.2 x 10^{-6}	3.6 x 10^{-7}

A total dose of 70 µSv/y follows from this ingestion pathway, mainly dominated by ^{137}Cs. The sum from all dose contributions is thus conservatively estimated to less than 80 µSv/y.

4.3.3 Preliminary analysis for workers under accident conditions

Several scenarios may result in the exposure of the workers under accident conditions. Mainly these scenarios are dominated by the spill of liquid contaminants from either the recombiner or from the core vessel resulting.

The recombiner is the item containing the highest amount of loose contamination 1 x 10^9 Bq of ^{137}Cs and less than 1 x 10^6Bq of ^{90}Sr. It is important to note that the core vessel is intended to be lifted without being segmented and placed in shielding drum to be stored. When opening the flanges, connecting the connection pipe with the recombiner, the worker has to be close to the component under difficult spatial conditions so that he might be exposed to the contamination falling down and needs a longer time to leave the recombiner vault. The assumptions for the scenario are the following:

— Most of the ^{137}Cs and ^{90}Sr will adhere to the surface;

— 1 % set free into the atmosphere;

- Volume into which contamination is dispersed 1 m³;
- Worker staying in that area for 1 min;
- Breathing rate 1.2 m³/h;
- Air concentration of ^{137}Cs is 1 x 10^7 Bq/m³;
- Inhaled activity is (10^7 Bq/m³ x 1.2 m³/h) / 60 1/h = $2 \cdot 10^5$ Bq;
- Inhalation dose factor is 6.7 x 10^{-9} Sv/Bq;
- Dose: ~ 1.5 mSv from ^{137}Cs; and
- An estimate for ^{90}Sr can be deduced from the results for ^{137}Cs taking into account that the activity of ^{90}Sr is 10^{-3} less than the activity of ^{137}Cs and that the inhalation dose factor (2.4 x 10^{-8} for type F) of ^{90}Sr is about a factor of 3.6 higher than of ^{137}Cs. In total a dose of about 6 µSv/y will result from the inhalation of ^{90}Sr. Thus the major contribution to the worker's exposure is expected to be from the inhalation of ^{137}Cs.

Thus, compared to the dose criteria for accident situation the screening of this scenario shows, that that hazard potential of the real working conditions is low.

Due to the different composition of the radionuclide vector (^{90}Sr, ^{137}Cs and alpha contamination) another scenario affecting the core vessel needs to be screened. It is assumed that the core vessel will drop during loading of the waste container, which is located in the former recombiner vault, after the recombiner has been removed. Different to the spill scenario discussed previously, in this scenario the worker, affected by the drop will be able to leave the vault or the area of the drop immediately (i.e. time of intake is assumed to be less than 5 s.). Assuming the drop occurs within the recombiner vault the following further assumptions are made:

- Most of the ^{137}Cs, ^{90}Sr and alpha contamination will adhere to the surface;
- 1 % set free into the atmosphere due to the drop of the core vessel;
- The volume into which the contamination is dispersed is 1 m³;
- The person staying in that area for a maximum of 5 s., if the drop occurs;
- The breathing rate is 1.2 m³/h;
- The concentration of ^{137}Cs in the air is 2 x 10^7 Bq/m³;
- The concentration of ^{90}Sr in the air is 4 x 10^4 Bq/m³. It is estimated based on the results of the recombiner scenario and the assumption that the contribution of ^{90}Sr can be neglected.
- Alpha contamination is 5 x 10^5 Bq/m³;
- The inhaled activity is:
  - For ^{137}Cs - 2 x 10^7 Bq/m³ x 1.2 m³/h x 5 / 3 600 1/h = 3.4 x 10^4 Bq; and

- For alpha contamination - 5 x 10^5 Bq/m³ x 1.2 m³/h x 5 / 3600 1/h = 8.4 x 10^2 Bq;

— The inhalation dose factor is:

- For ^{137}Cs - 6.7 x 10^{-9} Sv/Bq; and
- For alpha contamination (for ^{241}Am as reference radionuclide) - 2.7 x 10^{-5} Sv/Bq;

— The dose contribution:

- Of ^{137}Cs is 0.23 mSv; and
- Of alpha contamination is 23 mSv.

In summary, in the scenario of the dropped core vessel the most dominant contributor is the alpha contamination of the remaining inventory of the core vessel.

For an preliminary estimate the released radioactive material in both scenarios was considered to be from drops first to the floor representing a point source with a distance of 0.75 m (equivalent to a height of an average person). Taking into account the gamma coefficients for ^{137}Cs (0.092 10^{-9}mSv m^2/h) and for ^{90}Sr (^{90}Y) (0.007 10^{-9}mSv m^2/h) ([9]) and assuming the same amount of radioactive material released as for the previous calculations the resulting dose rate are:

— Spill from recombiner - 0.12 µSv/h; and

— Drop of core vessel - 0.24 µSv/h (for ^{137}Cs only as due to the gamma coefficient and the lower activity the ^{90}Sr dose can be neglected):

Taking into account that for the scenario of spill from the recombiner the duration of the exposure is not more than 1 min (60 s.) and for the drop of the core vessel is not more than 5 s. the resulting external exposures are estimated to be below 1µSv.

Accidental inhalation of graphite dust due to a failure during maintaining (unloading, loading, performing measurements) will be treated as follows: in a worst case scenario a graphite brick drops and becomes damaged dramatically resulting in the generation of a cloud of graphite dust and a fraction of the dust will be inhaled by a worker. A pessimistic assumption of the inhaled amount of dust is of a 1 cm^3 equivalent to 1.7 g graphite. Taking into account a maximum ^{152}Eu activity concentration of a brick of 115 g/cm^3 (see Table III.8 in Appendix III) the estimated inhaled activity is 200 Bq resulting in an internal exposure of less than 6 µSv (using an inhalation dose factor of 2.7 x 10^{-8} Bq/cm^3 from Ref. [4]).

Reference data about the ^{14}C concentration in the graphite was not available. But, comparing the situation of the research reactor under investigation with the situation at a small research reactor of thermal power 250 kW, the following estimate was made. The ^{14}C inventory of the graphite reflector of the dismantled German TRIGA reactor was 1.8 x 10^9 Bq with a graphite mass of 800 kg and total produced thermal energy of 91.1 MWd. Scaling the average ^{14}C concentration by the ratio of the produced thermal energy of the TRIGA reactor and of the research reactor under investigation an average ^{14}C concentration in the graphite of 12 Bq/g can be estimated. To become more conservative, a safety calculation factor of 10 is introduced to compensate for deviations of the relevant neutron flux in the graphite a conservative estimate of 120 Bq/g results. Again assuming that 1.7 g graphite is inhaled accidentally an additional effective dose of less than 1 µSv (dose coefficient for ^{14}C from Ref.

[9]) due to ^{14}C will result. In summary, the inhalation of graphite can be neglected as relevant accident consequence.

Finally, a failure in the personal protective equipment to avoid incorporation of dust during cutting of the biological shielding has been also considered relevant for further evaluation. Based on the experiences from previous decommissioning of a reactor, it can be assumed that no more than a fraction of 3% of the concrete becomes dust due to the use of the dry wire cutting system. Conservatively assuming, that 90% of the dust can be extracted by the used suction system, the fraction of concrete in air is 0.3%. Dismantling of the activated concrete is performed during 2 week, which represents 10 working days equivalent in total to 80 hours. In total a generation of contaminated dust of 0.3% of the total inventory of the concrete over 80 h can be assumed resulting in a generation rate of radioactive material of 4 x 10^2 Bq/h. Assuming, that relevant volume is 1 m^3 dust air concentration generation rate is 400 Bq/m^3/h. Assuming that the equilibrium concentration is about twice the generation factor, the air concentration is about 800 Bq/m^3/h. Thus; for one worker on duty for 8 hours a day with a fully ineffective breathing mask or respiratory system for that day the incorporated activity of air bound ^{60}Co is about 7 500 Bq. Taking into account as dose inhalation coefficient (worst case, [4]) 1.7 x 10^{-8} Sv/Bq the dose due to internal exposure is 130 µSv.

All accident scenarios discussed above will not contribute to any significant equivalent dose of the hand of a worker. The estimates for the normal scenario can be regarded to be conservative and enveloping with respect to any accident scenario, in which contact with either the reactor core vessel or with the recombiner will be during removal of the components after any drop. As these situations allow preparation of effective protection measures the doses will be much lower than those estimated.

4.3.4. Preliminary analysis for members of the public from accident conditions

There are only a few accident scenarios during the decommissioning of a small research reactor that may lead to a release of radioactive material to such an extent that exposure of the public would be radiologically relevant. A screening analysis therefore needs to define a sufficiently conservative source term for the release of the radioactive material and calculate doses on the basis of a plain and enveloping model for dispersion and exposure pathways. Suitable generic assumptions for such an approach are taken from IAEA Safety Reports Series 19 [8] and the Procedure for Calculating Doses from Accident Scenarios according to Section 49 of the German Radiation Protection Ordinance [10].

(a) Determination of the source term

A sequence of events that may lead to a substantial release of radioactive inventory into the environment is a fire in the research reactor. As a result of that part of the activity inventory will be mobilized and some parts of the building will be affected in such a way that part of the radioactive inventory being released into the room atmosphere can further migrate into the environment. It is conservatively assumed that a fire will start in the reactor hall, leading to combustion of the entire graphite and thus to the release of the total radioactive inventory of the graphite. It is further assumed that the fire will mobilize 10 % of the entire contamination on metallic surfaces, while any activation in metallic components is retained. This leads to the following activity inventory that will be affected:

— Activation in graphite: 100 % of 1 x 10^8 Bq (^{152}Eu); and

— Contamination: 10 % of 3 x 10^9 Bq (^{137}Cs) and of 5 x 10^7 Bq (α).

The heated fumes are expected to leave the building over the roof via the filters and stack which will have lost their effectiveness. The fire will further have destroyed some windows in the reactor hall increasing the air flow. The fire is assumed to burn for 2 hours (7 200 s). The release rate of activity is assumed to be constant during this time. This leads to the following source term estimated to be released into the environment:

— ^{152}Eu - 1 x 10^8 Bq/7 200 s = 1.4 x 10^4 Bq/s;

— ^{137}Cs - 3 x 10^8 Bq/7 200 s = 4.2 x 10^4 Bq/s; and

— α - 5 x 10^6 Bq/7 200 s = 6.9 x 10^2 Bq/s.

(b) Modelling of the dispersion in the research reactor and in the environment

The release of the contaminated air takes place over the roof. The heat content in the air from the fire leads to a buoyancy of the air plume, so that the effective height of release is larger than the roof top (about 25 m). The effective height is therefore assumed to be 40 m.

For the subsequent dose calculations, it is necessary to know the maximum activity concentration in the air (for doses from inhalation and from beta/gamma submersion), as well as the surface activity concentration on the ground (for doses from external gamma irradiation from the ground and secondary ingestion via various food pathways). For screening purposes, the short-term dispersion factors are taken from reference data [10] and the pathways are limited, while in a more sophisticated analysis the dispersion factors would be calculated yielding more realistic results and all pathways would be taken into account.

Short-term dispersion factors for gamma submersion are provided in tabulated form e. g. in [10], where they are listed for effective release heights of 20, 50, 100, 150 and 200 m and diffusion categories A (unstable) to F (stable). For screening calculations, the highest dispersion factor for an effective release height of 50 m and diffusion category E is used, which is χ = 0.04 s/m². This value refers to a wind speed of 1 m/s. The corresponding wind speed in a height of z = 50 m can be estimated as 1.8 m/s. The effective dispersion factor for gamma submersion therefore becomes (0.04/1.8) s/m² = 0.022 s/m².

The short-term dispersion factor relevant for calculation of doses from inhalation, gamma depletion etc can be calculated as follows:

$$\hat{\chi}_j = \frac{1}{\pi \cdot \sigma_{y,j}(x) \cdot \sigma_{z,j}(x) \cdot u} \cdot \exp\left(-\frac{H_e^2}{2 \cdot \sigma_{z,j}^2(x)}\right) \cdot \exp\left(-\frac{y^2}{2 \cdot \sigma_{y,j}^2(x)}\right) \quad (4)$$

Where

H_e : effective height of emission, 50 m (see above);

u : wind speed in the effective height of emission, 1.8 m;

x, y : coordinates of receptor point (here, y = 0); and

σ_y, σ_z : diffusion parameters, in m (see below).

The diffusion parameters can be calculated using the following approach:

$$\sigma_y = p_y \cdot d^{q_y} \text{ and } \sigma_z = p_z \cdot d^{q_z} \qquad (5)$$

Where

d	:	distance to the receptor point, in m;
p_y	:	coefficient, 0.801;
p_z	:	coefficient, 0.264;
q_y	:	exponent, 0.754; and
q_z	:	exponent, 0.774.

This results in the following values for the diffusion parameters for a distance of 500 m:

$$\sigma_y = 87 \text{ m}, \sigma_z = 32 \text{ m}$$

Using these data and the coordinates (x,y) = (500 m, 0 m) as inputs into the equation for the short-term dispersion factor above yields:

$$\chi = 1.9 \times 10^{-5} \text{ s/m}^3 \qquad (6)$$

Fallout and washout factors are relevant for providing an estimate of the activity which is depleted on the ground. The fallout factor F is the product of the short-term dispersion factor and the deposition velocity v_g (default value 1.5 x 10^{-3} m/s):

$$F = \chi \times v_g \qquad (7)$$

The washout factor is based on a precipitation intensity of 5 mm/h and is calculated according to the following equation:

$$\hat{W}_j = \frac{\Lambda}{\sqrt{2\pi} \cdot \sigma_{y,j}(x) \cdot u} \cdot \exp\left(-\frac{y^2}{2 \cdot \sigma_{y,j}^2(x)} \right) \quad (8)$$

Where the parameters have the meaning given above and Λ is the washout coefficient, in 1/s.

Using the relevant data as inputs into the equations for the fallout and washout factors yields:

$$F = 2.8 \times 10^{-8} \text{ m}^{-2} \text{ and } W = 6.4 \times 10^{-7} \text{ m}^{-2} \qquad (9)$$

(c) Modelling of inhalation

The inhalation is calculated on the basis of the breathing rate, the airborne activity concentration and the inhalation dose coefficient as follows:

$$H_{inh} = g_{inh,r} \cdot \dot{Q}_r \cdot t_{rel} \cdot \chi \cdot \dot{V} \qquad (10)$$

Where

t_{rel}	:	duration of the release;
$g_{inh,r}$	:	inhalation dose coefficient for radionuclide r;
$\dot{Q}_r$	:	release rate for radionuclide r; and
$\dot{V}$	:	breathing rate, 1.2 m³/h.

The inhalation dose coefficients for the three radionuclides [4] are presented in Table 9.

TABLE 9. INTERNAL DOSE COEFFICIENTS [Sv/Bq]

^{137}Cs	**^{241}Am**	**^{152}Eu**
3.9 x 10^{-8}	9.6 x 10^{-5}	4.2 x 10^{-8}

Using these data as inputs into the equation for the inhalation dose above yields the doses from the accident scenario as presented in Table 10.

TABLE 10. INHALATION DOSE [Sv]

^{137}Cs	**^{241}Am**	**^{152}Eu**	**Sum**
7.4 x 10^{-8}	3.0 x 10^{-6}	2.7 x 10^{-8}	3.1 x 10^{-6}

(d) Modelling of external exposure from ground deposition and from submersion

The dose from external irradiation of radionuclides depleted on the ground can be calculated by estimating the dose for an adult as the follow-up dose until the age of 70 years (i.e. to cover the lifetime between the age of 18 and 70).

$$H_T = \left[(1-\vartheta)\cdot g_{b,r}^{>17a} + \left(1-\vartheta^{52}\right)\cdot b \cdot g_{b,r}^{>17a}\right]\cdot \frac{1}{\lambda_r}\cdot (F+W)\cdot \dot{Q}_r \cdot t_{rel} \quad (11)$$

With the abbreviation $\vartheta = \exp(-\lambda_r \cdot t_1)$ (12)

Where

t_1 : duration of 1 year;

λ_r : decay constant for radionuclide r; and

$g_{b,r}^{>17a}$: dose coefficient for external irradiation from ground deposit for adults.

(> 17 years) for radionuclide r.

The dose coefficients for external irradiation from ground deposit, the half-lives and decay constants for the three radionuclides [10] are presented in Table 11.

TABLE 11. DOSE COEFFICIENTS FOR GROUND DEPOSIT, HALF-LIVES AND DECAY CONSTANTS

	^{137}Cs	**^{241}Am**	**^{152}Eu**
Dose coefficient [Sv·m²/(Bq·s)]	5.30 x 10^{-16}	1 x 10^{-18}	1 x 10^{-15}
Half-life [y]	30.17	432.2	13.6
Decay constant [1/s]	7.29 x 10^{-10}	5.09 x 10^{-11}	1.62 x 10^{-9}

Using these parameters in the equation for the estimation of doses from external irradiation from ground deposits yields the doses from the accident scenario presented in Table 12.

TABLE 12. EXTERNAL EXPOSURE FROM GROUND DEPOSIT [Sv]

^{137}Cs	^{241}Am	^{152}Eu	Sum
5.3 x 10^{-5}	2.7 x 10^{-9}	2.0 x 10^{-5}	7.4 x 10^{-5}

The dose from external irradiation from the immersion in the cloud at ground level can be calculated, using a simplified approach, as follows:

$$H_{\lambda} = g_{\gamma,r} \cdot \dot{Q}_r \cdot t_{rel} \cdot \chi_{\gamma} \qquad (13)$$

Where

$g_{\gamma,r}$: dose coefficient for external irradiation from immersion in the cloud for radionuclide r.

The dose coefficients for the three radionuclides [10] are presented in Table 13.

TABLE 13. DOSE COEFFICIENTS FOR IMMERSION

	^{137}Cs	^{241}Am	^{152}Eu
Dose coefficient [m²/(Bq·s)]	1.10 x 10^{-16}	1 x 10^{-18}	3.60 x 10^{-16}

Inserting these data into the equation for the doses from external irradiation from immersion above yields the doses from the accident scenario presented in Table 14.

TABLE 14. EXTERNAL DOSE FROM IMMERSION [Sv]

^{137}Cs	^{241}Am	^{152}Eu	Sum
7.3 x 10^{-10}	1.1 x 10^{-13}	7.9 x 10^{-10}	1.5 x 10^{-9}

(e) Calculation of the total dose

The total dose from the scenarios (see Table 15) is calculated as the sum from the above three scenarios.

TABLE 15. TOTAL DOSE FOR THE ACCIDENT SCENARIO [Sv]

^{137}Cs	^{241}Am	^{152}Eu	Sum
5.3 x 10^{-5}	3.0 x 10^{-6}	2.0 x 10^{-5}	7.7 x 10^{-5}

This result indicates that the total dose from the accident scenario will be in the range of less than 100 µSv. The main contribution is expected to be from external irradiation from the ground where the radionuclides have been deposited and where the person is assumed to stay for a very long time. The contribution from ^{137}Cs and ^{152}Eu is of the same order of magnitude, while ^{241}Am has the main contribution via the inhalation pathway with a lower absolute value.

The overall result of this evaluation is that dose limits for accidents are met by several orders of magnitude.

(f) Limitations of the preliminary analysis

This preliminary analysis relies on the following assumptions and simplifications:

— The distance of the critical group is set to be 500 m as this roughly corresponds to the position of the family homes to the east of the research reactor site. The analysis has also to be performed for shorter distances with properly adjusted conditions (no dwellings).

— The dispersion has been calculated only for diffusion category E, i.e. stable conditions, which is a generally conservative assumption but does not guarantee to estimate the maximum potential exposure.

— The analysis has been carried out only for adults and needs to be expanded for other age groups as well.

— The analysis did not take into account ingestion of food grown on the contaminated land. These pathways would have to be included in a more refined analysis. It is nevertheless justified to exclude these more complicated pathways in this first screening analysis as the ingestion pathways constitute medium and long term effects and might in principle be ruled out by appropriate administrative measures (prohibition of harvesting crops, etc.).

4.3.5. Analysis of waste management

Hazards during temporary storage and handling of radioactive waste in the reactor hall are already addressed in the earlier sections. The risks are in the same order of magnitude or smaller, since the waste is handled without elevation:

— Direct exposure is relevant during health physics measurements (e.g. dose rate at the surface and at a small distance from the waste packages), but the use of extension tools will reduce the exposure to a sufficiently low level. In addition, the dose rates are low, especially after placing the more activated components as reactor core vessel and recombiner in the waste drums and using reactor lead as internal shielding in the drums.

— Further contributions from the stored waste to the exposure during the decommissioning is not anticipated, as the interim storage cell is place behind shielding walls of sufficient size.

— The waste will be handled to another building for measurement and final packaging. Due to the proposed conditions on surface contamination and dose rates of the waste drums no significant contribution to the transport personnel is expected to occur. The main risk during handling within the reactor hall may result from drop of a drum. With respect to the potential worker’s exposure, the relevant scenarios due to spills from the core vessel or from the recombiner are very unlikely and conservatively contained in the preliminary analysis of the worker’s accident scenarios in Section 4.3.3.

In general terms the resulting risks associated with radioactive waste handling in the reactor hall are twofold:

— Conventional (industrial/chemical) risks; and

— Mixing waste streams.

The main industrial/chemical risks to be considered are:

— Falling of waste packages due to improper pile up of waste packages;

— Cutting to sharp edges from cutting metal parts;

— Accidents during transportation e.g. with forklift trucks;

— Ingestion of lead dust, from shielding materials; and

— Residual chemicals from decontamination.

The risk are planned to be minimized by using quality tools fit for the job, training of workers and good planning and communication. All waste handling can be performed with normal industrial tools. Development of specialized tools (with new risks unknown to the operators) is considered not necessary for the Research Reactor Test Case.

Mixing of waste streams could give exposure to the public (e.g. if radioactive waste without proper labeling would be mixed up with industrial non-radioactive waste and be transported to a conventional waste dump). The probability of mixing will be minimized by the consequent use of zoning in the reactor hall (segregation of several types of waste in different corners of the building) and good labeling and registration of all waste items. This will be performed immediately after taking the waste item out of the reactor. In addition, the waste will temporarily stored in dedicated waste storage cell with limited access options (e.g. by barriers to enter the cell).

Procedures and checklists will be part of the quality management system. Only a small group of qualified personnel will perform the decommissioning and waste handling according to strict procedures. Procedures, checklists and the involvement of qualified personnel will contribute to a low level of risk related management of radioactive waste.

4.3.6 Results from the hazard identification and the preliminary hazard analysis

Hazard identification and preliminary analysis of the most relevant scenarios show that only a few scenarios are relevant for the exposure or potential exposure of the workers and of the public that are summarized below:

(a) Normal scenarios for workers

— The removal of activated core vessel results in an estimated external dose of about 550 µSv for a worker. The contribution due to the removal of the recombiner needs to be calculated during the more detailed analysis.

— The estimated equivalent dose for the hand of a worker is below 10 mSv and thus does not require a more detailed analysis.

— Incorporation during normal operation can be excluded as basically no graphite dust will be produced during normal operation and personal protective equipment will be worn during the drilling in the graphite and as contaminated inner surfaces of components will not become assessable due to missing cutting (either for the recombiner nor for the core vessel). In addition, the dry cutting of the biological shielding will be implemented using personal protective equipment appropriate to reduce the dose due to intake of concrete dust to a very low value.

In summary, the external exposure during dismantling of the activated components can be considered the most important scenario.

(b) Normal scenarios for the public

— The screening shows that the doses are comparatively low with regard to the dose criteria of 300 µSv/a. Depending on the national regulations a detailed analysis may be required or the hazard screening may be regarded to be appropriate especially under the light of the graded approach. For reasons of illustration of the DeSa safety assessment methodology (see Volume I) and with respect to the public acceptability and opinion a further analysis is performed to show, that the doses will be yet lower in reality than the value calculated in the hazard screening.

— The most relevant exposure pathway is the ingestion. Thus, major effort must be laid on the modelling of the dietary habits resulting in incorporation and internal exposure – thus leaving out any ingestion pathways needs a clear justification.

— In addition to complement the data on public exposure the position of highest concentration of radioactive material at the site of the research centre needs to be analysed to take into account the maximum exposure on the site of the research centre. This modelling needs to be limited to external exposure and inhalation with air as no member of the public is consuming goods grown and produced at the site.

(c) Accident scenarios for workers

— The drop of the core vessel resulting in a spill of available inside remaining contamination may lead to an intake of radionuclides needs to be considered in more detail and the external exposure due to the spill can be neglected.

— In case of any accident the dose for the hand of a worker will be below the estimate for the normal conditions which can be regarded as the envelope scenario.

— The scenario of spill of remaining contamination from the recombiner needs no further analysis taking into account the foreseen dust mask of type P3 (reduction factor of 97 %) as a personal protective equipment. This protective measure will result in a reduced inhaled activity of about 1.1×10^3 Bq corresponding to an internal exposure of less than 0.02 mSv. External exposure due to that spill can be neglected.

— The inhalation of graphite in case of an accident needs no further analysis due to the very low dose. As during normal operation a mask will be worn a further reduction of inhaled graphite will result so that in total the dose due to an accident is very low.

— The analysis of the failure of personal protective equipment during the dry wire cutting does not require further detailed analysis as the estimate shows that the internal exposure is low and additional administrative measure will reduce the likelihood of such a long lasting failure.

Thus, with respect to a graded approach, no further in detailed analysis of the accident scenarios for workers is needed.

(d) Accident scenarios for the public

— The preliminary analysis uses a plain screening model while the detailed analysis will be based on a computer based advanced modelling. The preliminary analysis also shows that the dominate contribution is resulting from the external exposure from the ground deposit.

— The preliminary analysis does not take into account dietary habits. This might be subject of a more detailed analysis (see Section 5.2).

(e) Special aspect of waste management

No additional consequences for the workers or the public resulting from the waste management activities are envisaged.

Note, that depending on the national requirements the level of detail of the preliminary hazard analysis (as presented in the previous Sections) may be regarded to be commensurate with the hazard and hazard potential of the research reactor. In this case no further detailed analysis will be required and justified. Nevertheless, if a more detailed analysis is required, this analysis needs to be based on the scenarios summarized above. In this case a justification (e.g. by summarizing the results of the hazard identification, preliminary hazard analysis and screening according to Section 4.3.5) has to be provided. The following Section 5 provides the conduct and the results of such a more detailed analysis.

5. HAZARD ANALYSIS AND EVALUATION

In this section, a detailed hazard analysis is performed applying different methods of calculation and modelling. The level of detail is commensurate with the low hazard potential of the research reactor, as demonstrated in Section 4. The following calculations complement the preliminary hazard analysis presented in Section 4 by using more detailed assumptions and assessment tools.

5.1. ANALYSIS OF NORMAL ACTIVITIES

5.1.1 Analysis of normal scenarios for workers

The initial planning and work breakdown schedule is shown in Appendix I. It is known from the preliminary hazard analysis (see Section 4.3.1) and from the dose rate measurements (see Appendix I, III and IV) that the dose rates after reactor's shutdown are not very high. Modelling the decommissioning tasks with available modelling tools (e. g. MicroShield [11] or VISIPLAN [12]) is sufficient to estimate the dose rate during individual decommissioning tasks and for optimization of the planned work. VISIPLAN is used for the following analysis.

The first step is to build the general model of the reactor using the known geometry data, material data and activity data (either from sampling/analysis or by estimation based on operating history or dose rate measurements). Figure 17 shows the schematic representation of the reactor.

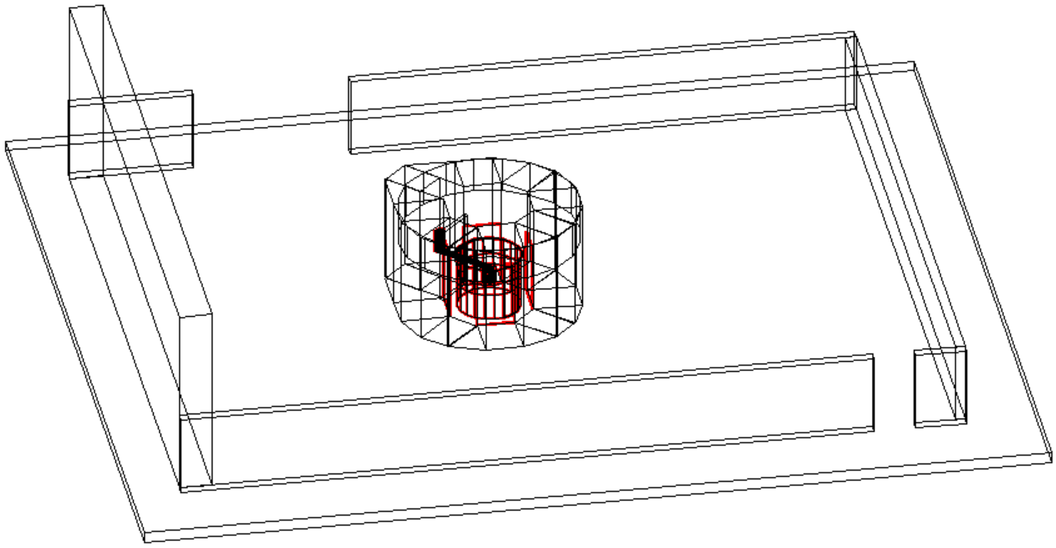


FIG. 17. The model before start of decommissioning.

All known γ-sources taken from Table 1 and from Appendix III and IV are modelled. With respect to the low inventory and the estimated doses (see Section 4);

— Simplified activity distributions are used to reduce the assessment effort; and

— Related doses are used to gain a more detailed characterization of the activity distribution in components and systems. For example, a line source is used for the pipe connecting the reactor tank and the recombiner, contamination of the reactor tank is evenly distributed as a thin film on the inside of the tank and activation of the steel plates at the inner side of the shield is assumed to be homogeneous.

A dose-rate distribution is calculated to check the model and determine the areas with highest dose rates (see Figure 18). As a result the reflector tank activation gives the main contribution to the dose rate in the reactor hall.

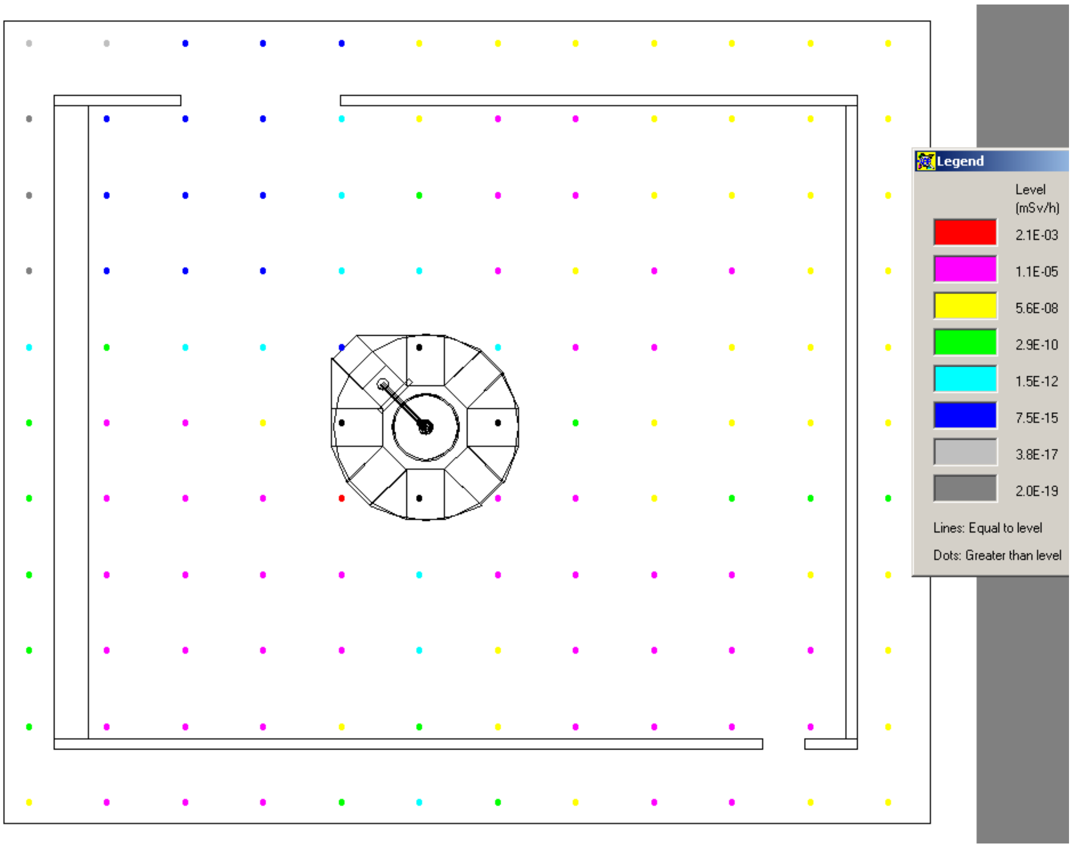


FIG. 18. Initial dose rate estimation before start decommissioning.

As a next step of the modelling, all relevant steps of the work breakdown schedule are modelled in a sequence. Steps are relevant when people have to work close to the sources or have to work at increased dose rates environment for a longer period. It is assumed that normal precautions are taken, e.g. workers will not go into increased dose rate areas any longer than necessary, for the task to be carried out. Optimization is performed for example by calculating dose uptake with additional lead shielding (it takes more time very close to the source to install the shielding, but dose rate will be lower once installed) and compare it to dose uptake when working without additional shielding.

The most significant tasks in terms of exposure to workers are the removal of the recombiner and the removal of the reflector and the reactor vessel:

— The dose rate near the recombiner is calculated to be up to 1.5 mSv/h. The initial estimated individual dose for this task is 0.5 mSv. By using a 2 cm lead slab to partly cover the recombiner, the dose slightly decreases. By using remote tools (e.g. extension shafts on spanners) the individual dose for this task can be reduced to 0.2 mSv.

— Once the recombiner is removed, the calculated dose rate decreases to several tens of µSv near the reflector tank. Once the reflector tank lid is removed and the upper layers of graphite are removed the dose rate near the surface of the reactor tank at the working place is calculated to approximately be 2 mSv/h. Since the removal of the graphite and reactor tank take several hours, it is necessary to use remote control tools for the tasks. As long as the workers are on top of the shielding, dose rates are 100 times lower than in the vicinity of the reactor tank. By using remote controlled tools and stay as far away from the tank as practical the individual dose will be from 0.2 to 0.4 mSv.

The above calculations are based on the assumption that the same worker will do all the decommissioning tasks in high dose rate areas. If this work is shared by several workers, the doses to individual workers will be accordingly lower.

In total, the maximum dose of a worker involved in all working steps will be less than 0.6 mSv. This result is in good agreement with the estimate performed during the preliminary analysis (see Section 4) and demonstrates the high importance of appropriate working tools.

As a result of the hazard identification and preliminary hazard analysis the internal exposure due to incorporation of radionuclides can be excluded due to the protective measures in place.

5.1.2. Analysis of normal scenarios for members of the public

Analysis of normal scenarios for members of the public was carried out following all the assumptions and input parameters described in 4.3.2. These assumptions were related to the source term, geometrical conditions, meteorological conditions, location of the nearest houses, except the height of the exhaust stack of the research reactor which initial value of 15 m (used in the preliminary analysis) was agreed to be changed to 25 m for this detailed analysis.

The main input parameters are presented in Table 16 and Table 17.

TABLE 16 SOURCE TERM USED IN THE NORMAL SCENARIO FOR THE PUBLIC

Source Term [Bq/y]	
^{137}Cs	3×10^7
^{60}Co	2×10^6
^{154}Eu	1×10^6
^{90}Sr	1×10^5

TABLE 17 GEOMETRICAL AND METROLOGICAL DATA USED IN THE NORMAL SCENARIO FOR THE PUBLIC

Geometry and Meteorology	
Source height	25 m
weather condition	Stable
Wind speed	2 m/s
Wind direction	West
Dry deposition rate	1 000 m/d

This analysis consists of the following two main steps:

(a) Simulation of the airborne dispersion from the exhaust stack of the research reactor to the environment to obtain values of the ground surface concentrations in the air and the ground deposition concentrations for the radionuclides of interest (^{137}Cs, ^{90}Sr, ^{60}Co and ^{154}Eu); and

(b) Calculation of the exposure for members of the public from inhalation and direct exposure from the ground deposits, as well as from the secondary ingestion via radioecological pathways.

The first step of the analysis was performed based on the straight line Gaussian model for the routine discharge [13, 14, 15 and 16] incorporated in the Fortran 77 computer code developed in Vinča Institute of Nuclear Sciences (Serbia). This code is routinely used for the calculation of the atmospheric dispersion of the radionuclides from the nuclear facilities in Vinča Institute for the needs of the safety assessments and environmental impact assessments (see also Appendix VI.1).

The second step of the analysis was performed by using RESRAD code, version 6.3, developed in Argonne National Laboratory [17] using results of the first step as input data (see also Appendix VI.2).

The results of the analysis are explained in detail in Appendix VI.3. The total annual dose due to external exposure or due to incorporation of radionuclides discharged during one year is less than 8 µSv/y for the first year of discharge. This dose is expected to decrease exponentially within the following year to a total dose of less than 1 µSv/y after 40 years of discharge. An additional contribution of less than 0.1 µSv/y in the first year due to inhalation of radionuclides, which are still in the air above the contaminated ground surface, has also to be taken into account. This contribution will decrease accordingly within the following years.

The results summarized are obtained for the area of maximum radionuclide concentration, which is about 800 m east from the emitting stack of the research reactor and thus is relevant for the public living in the near distance. Thus, the total dose is far below the dose limit of 1 mSv/y for the public.

For those employees working at the research centre but not involved in the decommissioning of the research reactor, are also to be regarded as public. Different to the population living close to the site for this group of public it is not required to assume any dietary behaviour with respect to contaminated food. Taking into account the results of the calculations for the site the main contribution to the exposure of this critical group will result from the external exposure and from the intake from radionuclides still in air. Taking into account the fact that this group is present 2 000 h/y on site only, instead of 8 760 h/y for the public, the resulting annual dose is far below 300 µSv/a.

5.2. ANALYSIS OF ACCIDENT SCENARIOS

5.2.1. Analysis of accident scenarios for workers

The preliminary analysis of accident scenarios for workers in Section 4.3.2 shows that:

— The internal exposure in case of inhalation of radioactive material (consisting of ^{137}Cs and ^{90}Sr) due to dropping contaminated liquids from the recombiner when working below the recombiner will result in about 1.5 mSv;

— Due to the low dose rates and risk, the scenario of inhalation of graphite during the removal of graphite can be ignored; and

— Due to the alpha contamination inside the core vessel, the preliminary analysis shows that a drop of the vessel with resulting spill and aerosol is the scenario with the highest internal exposure (about 23 mSv).

On the basis of the results and as a consequence and measure of precaution, the workers need to wear personal protective equipment and especially dust mask during the removal of the core vessel, if they are entering the reactor cave. As far as possible, the handling of the core vessel will need to be performed with remote tools to minimize the direct vicinity of workers during transfer of the core vessel from the reactor to the waste drum. In addition, the waste drum will be placed at the top of the reactor to minimize the transfer path of the core vessel at the crane hook. Finally, some technical precautions are followed: exhaustion, air monitor. Furthermore, there are adequate instructions to explain to the workers how to react if the core vessel is dropping.

5.2.2 Analysis of accident scenarios for members of the public

The full analysis of doses to members of the public from accident conditions is carried out according to Ref. [10]. The accident scenario is chosen identical to the situation depicted in Section 4.3.4. It covers a fire that will start in the reactor hall, leading to combustion of the entire graphite and thus to the release of the total radioactive inventory of the graphite. It is further assumed that the fire will mobilize 10 % of the entire contamination on metallic surfaces, while any activation in metallic components is retained.

The calculation procedure includes the following pathways:

— External gamma irradiation from radionuclides deposited on the ground;

— External gamma irradiation from submersion in the radionuclide cloud;

— External beta irradiation from submersion in the radionuclide cloud;

— Inhalation of radionuclides; and

— Ingestion of radionuclides via various radioecological pathways.

The details of the model, including the default parameters and formulae, are described in Ref. [10]. The pathways listed above comprise all those described in Section 4.3.4 for the screening calculations as well as ingestion pathways.

The calculations are carried out for the radionuclides listed in Table 18 - ^{152}Eu, ^{137}Cs and ^{214}Am, using the source term that corresponds to the assumptions in Section 4.3.4. ^{14}C (in the form CO_2) has been included to account for any releases from the graphite during the accident.

TABLE 18.RADIONUCLIDES AND SOURCE TERMS USED FOR THE DETAILED ANALYSIS OF DOSES FROM ACCIDENT CONDITIONS

Radionuclide	**Released Amount [Bq]**
^{152}Eu	1.008×10^8
^{137}Cs	3.024×10^8
^{241}Am	4.968×10^6
^{14}C (as CO_2)	1.000×10^{11}

The calculation procedure laid down in Ref. [10] requires some cases to be distinguished. First of all, there is the way to model the release into the atmosphere. The influence of the building needs to be taken into account if the effective height of release, H_e, is smaller than the sum of the height of the building, H_g, and the smaller value of either the height or the width of the building, I_g, measured in wind direction. i.e.:

$$\text{if} \quad H_e < (H_g + I_g) \qquad (14)$$

The value of H_e is estimated to be on the order of 50 m, resulting from the height of the point of release (roof of the building, stack at about 25 m) and the thermal buoyancy caused by the heat of the fire, estimated to 0.3 MW, leading to a thermal lift of 25 m. As the value of H_e of 50 m and the two dimensions of the building are both on the order of 25 m, giving also 50 m, the influence of the building does not need to be taken into account.

The meteorological situation during the accident is chosen conservatively, using a wind speed of 1 m/s and precipitation of 5 mm/h rain for the diffusion categories C, D and E. The calculation is carried out for all six diffusion categories and for all 12 sectors. However, only the sector opposite to the wind direction is of importance. The results are calculated for distances from 40 m to 2 500 m. As the distance from the reactor to the private houses is about 500 m, the detailed results are evaluated here for this distance only (while the complete analysis of course provides the complete dose data for all distances). The ingestion pathway is calculated under the assumption that the ingestion of any locally grown crops will be terminated within one day and that it is not be restarted until the following year. This is assumed for an area with a radius of 2 000 m around the reactor site.

The results obtained by this procedure are summarized in Figure 19. This figure shows the resulting doses for all six age groups as a function of distance. It can be observed that the results for all age groups have only minor differences. The diffusion category D is identified as the one yielding the highest doses.

Depending on the radiological properties of the radionuclides, the pathways with the highest contribution to the total dose are external gamma irradiation from ground deposits, inhalation and ingestion, as can be seen from Table 18. The percentage of these contributions varies slightly with the age group. This is also to be seen from the data in Table 19 where the results for infants and for adults have been compared. The highest doses from external irradiation from ground deposits and for the ingestion pathways is found in a distance of 40 m which is of course meaningless for this analysis as this spot is on the premises of the research centre where nobody would stay for a prolonged time. The highest doses for inhalation as well as gamma and beta submersion are calculated for a distance of 340 m which roughly corresponds to the distance of the dwellings, making these results conservative.

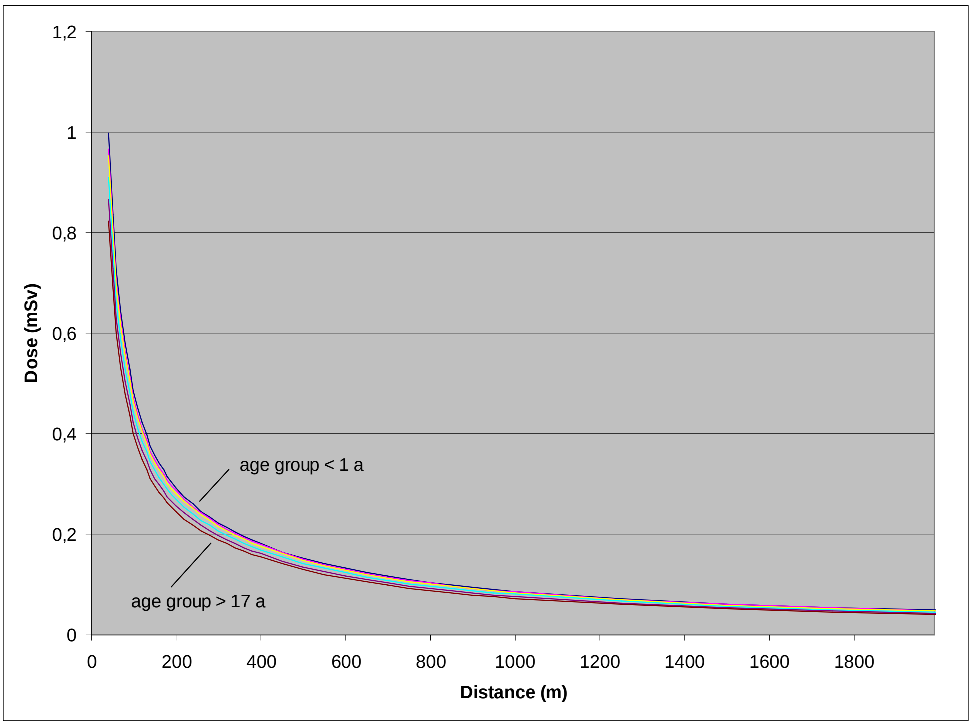


FIG. 19. Dose distribution as a function of distance from the reactor in accident conditions (all age groups).

TABLE 19. DOSE CONTRIBUTIONS FROM VARIOUS PATHWAYS FOR TWO AGE GROUPS, AT 500 M DISTANCE FROM THE REACTOR

Radionuclide	**γ, Ground [mSv]**	**γ Submersion [mSv]**	**β Submersion [mSv]**	**Inhalation [mSv]**	**Ingestion [mSv]**	**Sum [mSv]**
			Age group 0 – 1 y			
^{152}Eu	3.03 x 10^{-2}	7.42 x 10^{-8}	1.63 x 10^{-10}	1.66 x 10^{-5}	1.58 x 10^{-4}	3.05 x 10^{-2}
^{137}Cs	8.45 x 10^{-2}	1.13 x 10^{-7}	8.61 x 10^{-10}	4.98 x 10^{-5}	2.95 x 10^{-2}	0.114
^{241}Am	1.07 x 10^{-4}	1.42 x 10^{-10}	0	1.34 x 10^{-3}	1.33 x 10^{-3}	2.78 x 10^{-3}
^{14}C	0	0	6.75 x 10^{-9}	2.85 x 10^{-6}	4.91 x 10^{-3}	4.91 x 10^{-3}
Sum	**0.115**	**1.87** x 10^{-7}	**7.78** x 10^{-9}	**1.41** x 10^{-3}	**3.59** x 10^{-2}	**0.152**
			Age group > 17 y			
^{152}Eu	2.46 x 10^{-2}	5.18 x 10^{-8}	1.63 x 10^{-10}	4.72 x 10^{-5}	5.36 x 10^{-5}	2.47 x 10^{-2}
^{137}Cs	6.50 x 10^{-2}	6.28 x 10^{-8}	8.61 x 10^{-10}	1.32 x 10^{-4}	3.04 x 10^{-2}	9.55 x 10^{-2}
^{241}Am	7.60 x 10^{-5}	7.88 x 10^{-11}	0	5.32 x 10^{-3}	2.40 x 10^{-4}	5.64 x 10^{-3}
^{14}C	0	0	6.75 x 10^{-9}	6.92 x 10^{-6}	3.73 x 10^{-3}	3.74 x 10^{-3}
Sum	**8.96** x 10^{-2}	**1.15** x 10^{-7}	**7.78** x 10^{-9}	**5.51** x 10^{-3}	**3.44** x 10^{-2}	**0.13**

These results can be interpreted as follows:

— The doses from all radiological pathways at the nearest point where people would reside for a longer time (family homes, 500 m from reactor) are given in Table 19. These doses are lower than 0.2 mSv for all age groups.

— This shows that the consequences of an enveloping accident scenario are negligible and fall below any dose limit used for these types of analyses.

— There is reasonable agreement between the results of this refined analysis and the results of the screening analysis performed in Section 4.3.4, indicating that the first approach gave a good indication that the doses from accident scenarios would be marginal.

— The calculation model contains considerable conservatism concerning the dietary habits and food production of the people who might be affected by the releases from this accident scenario. Any doses incurred on members of the general public in real situations would be much lower than the dose values calculated here.

5.3 MODELLING AND CALCULATION OF CONSEQUENCES

The modeling and calculation of the consequences of normal operation and accident scenario is carried out either by straight forward calculations or by use of proven software tools and models (see Sections 5.1 and 5.2 for details). The selected scenarios are conservative.

Due to the low radioactive inventory of the research reactor and the related low hazards no risk criteria need to be considered. Accordingly no probabilistic models are used but only deterministic ones in the Research Reactor Test Case.

6. ENGINEERING ANALYSIS

In Section 3.2 all safety related structures, systems and components (SSCs) are summarized, which will be needed to ensure safety during the conduct of the decommissioning. They comprise of existing, modified and new systems and components. The impact of the SSCs on the safety is explained in Sections 4 and 5. The SSCs need to be available as soon as the individual decommissioning activity, for which they deliver a safety function, will be executed. If they are not available, the decommissioning activities will be stopped immediately or not launched.

The systems and components will be designed according to relevant national standards and with a view to fit for purpose. With respect to the low radioactive inventory and to the potential consequences and also depending on the national system not the full set of standards may need to be applied which are relevant for operation of the research reactor. As far as conventional safety is concerned relevant standards will be fulfilled.

Depending on the national requirements, the SSCs might be categorized according to a safety categorization system. Therefore different categories of SSCs might be subject to more detailed regulatory control and reporting.

In addition, all systems and equipment not classified as safety related structures, systems and components (e.g. the crane of the reactor hall, measuring systems for clearance measurements) are considered to be compliant with national conventional standards and requirements to ensure proper operation and to reduce any risk to workers and the public from malfunction.

7. EVALUATION OF RESULTS AND IDENTIFICATION OF CONTROLS

7.1. COMPARISON OF THE ANALYSIS RESULTS WITH CRITERIA

According to Section 2.5 and Section 2.6 the safety assessment needs to demonstrate that:

— The dose criteria for the public for normal operation (0.3 mSv/a) and for accident situations (1 mSv);

— The dose constraints and dose criteria for workers for normal operation (20 mSv/a) and for accident situation (50 mSv); and

— The dose limit for the hand of a worker is 500 mSv/y in normal conditions, are fulfilled, and that:

— The ALARA principle is taken into account for the workers; and

— Finally the clearance levels for release for unrestricted use of the reactor hall are met.

Based on the preliminary analysis (Section 4) and the detail analysis in Section 5 all dose criteria or dose constraints are fulfilled and the implementation of the ALARA principle is demonstrated:

(a) For the public

— *Normal scenarios*

The maximum dose for a member of the public of the critical group is determined to below 0.1 mSv/a for the first year, conservatively assuming a discharge of 1 % of the radioactive inventory within one year. For the following years the doses resulting from the discharge will decrease.

— *Accident scenarios*

Independent from the age of the member of the public, the dose due to an accidental release of 10 % of radioactive inventory results does not exceed 0.2 mSv.

(b) For workers

— *Normal scenarios*

Relying on appropriate working tools and personal protective equipment the maximum external exposure will be limited to 0.6 mSv. This number is conservative as it assumes that one worker is involved in all dose relevant activities. Thus, by optimizing the human resource planning this dose can be reduced further on. In addition, the dose of the hand of that worker will be less than 10 mSv based on a conservative approach and assuming that one worker is involved in all dose relevant activities. This leads to the conclusion that, the low potential doses and the low dose rates allow to use (as proposed in Section 3) less advanced decommissioning tools, e.g. for remote handling, resulting in doses to be regarded to be as low as reasonable achievable taking into account the conservative, but still low dose of a worker.

— *Accident scenarios*

Based on the use of personal protective equipment, it has been estimated that no accident scenario will result in an additional exposure of more than 0.02 mSv. The preliminary analysis shows that the use especially of dust mask of very high reduction factor will have significant influence of the inhalation of spilled radioactive material from the core vessel (worst case scenario). If the mask fails the resulting additional dose will be about 0.7 mSv. If the dust masks fail for one full day during dismantling of the biological shielding the resulting internal exposure will be about 0.13 mSv. Thus, in total no scenario will result in an excess of the relevant dose criterion of 50 mSv. Finally, it has to be mentioned, that these accident dose calculations do not include calculations for decontamination or clean-up measures which might become necessary as these are regarded to be subject to planned – if needed.

The procedure for clearance of the reactor hall as explained in Section 3.6.2 will ensure, that the clearance levels will be met. Depending on the national regulatory system the release for unrestricted use of the reactor hall might require some additional agreement by the Regulatory Body before the controlled area can be terminated.

7.2. TYPES AND TREATMENT OF UNCERTAINTIES

Within the safety assessment performed and documented in this safety assessment report no significant uncertainties needed to be managed. Detailed information on the hydrology, on the geology and especially on the dietary behaviour of the public are not needed with respect to the low radioactive inventory of the research reactor and the application of the graded approach.

As contamination of the graphite stringers, the reflector tank or of the steel plates of the inner side of the biological shielding cannot be excluded, related measuring procedures and personal protection measures are foreseen to avoid incorporation of radioactive material.

The radioactive inventory is known at a sufficient level of accuracy except the information on the ^{14}C concentration in the graphite to be dismantled. This missing information can be substituted by taking into account experiences from the dismantling of another small research reactor; the results show, that there is no significant impact on the dose of a worker and that thus the missing information is of less relevance for the safety of the worker (see Section 4.3.3).

The radioactive inventory is based on measurements performed in the context of a radiological characterization. The instruments and systems used were compliant with national requirements and fit for purpose. As far as dose rates were measured online the lower detection limit was between 0.06 µSv/h and 0.08 µSv/h corresponding to the background level. Different measurement campaigns and systems and components the statistical uncertainty of the measurements was between 1.5 % and 12 %. More detailed information can be obtained from Appendix III and Appendix IV.

7.3. SAFETY MEASURES

Safety is ensured during the whole stage of the decommissioning project. As explained in Section 7.1 all relevant dose criteria and dose constraints can be met. This is a result of the low radioactive inventory and the related hazards and of the safety controls in place. Only a few engineered systems are needed to ensure safety, procedural controls are straight forward according to the low radioactive inventory. The relevant safety controls are summarized below:

(a) Engineered systems:

— *Air monitoring systems*

Air radiation monitoring systems will need to operate during the decommissioning and dismantling activities to monitor i) the potential release of radioactive material into the environment and ii) the potential release of radioactive material into the reactor hall under normal and accident conditions. The systems are part of the preventive measures to avoid incorporation of radionuclides. Procedures for maintenance and daily functional testing are subject to the management system in place at the research reactor.

— *Fire detection system of the reactor hall*

The fire detection system will need to operate during the whole period of decommissioning of the reactor hall. It will be inspected and maintained according to the related programmes proven to be appropriate during operation of the research reactor. Inspection and maintenance programmes are compliant with the relevant national standards and requirements.

— *Covering of the biological shield and local ventilation during dismantling*

During dismantling of the biological shield grinding or chipping and dry wire cutting will be used. To avoid distribution of the potentially contaminated concrete dust in the reactor hall, the biological shield will need to be enclosed by a tent with local ventilation system (including a filter system and a monitoring system). The tent will be operated at lower pressure with respect to the pressure in the reactor hall to enable an adjusted airflow from the reactor hall to the tent. Procedures for maintenance and daily functional testing of the ventilation system will be subject to the management system in place.

— *General ventilation in the reactor hall*

The reactor hall is subject to a general ventilation to exchange part of the air with fresh air. The related system will be adapted according to the progress of the decommissioning activities.

(c) Procedural controls:

— *Preparation of work activities*

Procedures will need to be in place during decommissioning to ensure, that before start of each decommissioning activity preparations are performed to ensure that (i) all required information on the work step is available; (ii) all related hazards are briefly re-assessed; (iii) all protective measures are explained in detail; (iv) tools are tested and that especially systems influencing safety (e. g. the crane for the lifting of the recombiner, the suction of the graphite stringers and the lifting of the core vessel) are tested according to the test procedures before every use, and (v) all personal protective equipment is available.

Although no thermal cutting and dismantling techniques will be used, due regard will need to be given to any risk of ignition of combustible material to avoid fire. Mobile fire protection equipment will need to be close to the work place if a risk of fire cannot be ignored.

— *Control of work activities*

Start of any decommissioning activity requires the agreement by the project management and for those activities with potential exposure (especially dismantling of the recombiner, the graphite stringers and the core vessel) the agreement by the radiation protection officer. The project management and radiation protection officer will need to ensure that all protective measures are in place and that all personal protective equipment is used by the workers. This will need to involve continuous on-site inspections during the execution of the work activities.

— *Task specific controls*

During the dismantling of the recombiner, connection pipe, graphite stringers and the reactor core vessel, as far as possible, remote tools need to be used to reduce the external exposure. In addition, as a measure of precaution the workers will wear dust/breathing masks for the unlikely case of spills of radioactive material or generation of graphite dust due the drilling or due to a drop of stringers.

The grinding or chipping of activated parts of the biological shielding and the following dry wire cutting of the biological shielding will be carried out under a tent with separate ventilation system. Work will be allowed only, if the ventilation system is in normal operation. In addition, the workers will need to wear dust masks for conventional and radiological protection reasons.

— *Ensure reliable protection of the workers by use of personal protective equipment*

The personal protective equipment is essential to avoid incorporation of radionuclide. Therefore detailed procedural controls will need to be in place to ensure, that the wearer are familiar with the equipment and are able to test the required functionality (in case of devices as electronic dose meter) or to check, that the equipment is best fitting (e. g. in case of the dust masks).

It should be noted, that in addition to these procedural controls which will be established to avoid or at least to mitigate radiological consequences, further procedural controls and measure will need to be in place to ensure conventional safety of the worker during conduct of any decommissioning activity according to the national requirements.

8. GRADED APPROACH

The graded approach has been applied at several steps within the DeSa safety assessment methodology during the conduct of the safety assessment for the research reactor.

Due to the low radioactive inventory and the related low exposure of workers and also low potential exposure of the public, the level of detail of the description of site specific parameters and of the reactor, the level of detail of and complexity of models for the hazard analysis and hazard analysis and evaluation has been adapted:

- General assumptions on the climate conditions at the research centre were made and earthquake has not been taken into account. Air plane crash is not considered within the safety assessment as this was not subject to the safety assessment related to the operation of the research reactor nor is it regarded to be justified for the decommissioning safety assessment of this research reactor.
- Missing data on the radioactive inventory were substituted by conservative estimates from similar decommissioning projects.
- The hazard identification used a generic checklist, as proposed in Volume I of this report, as well as the "What-If-Technique". The HAZOP approach was regarded not to be justified with respect to the radioactive inventory, the complexity of the research reactor, both resulting in minor consequences for the workers and the public, and of the decommissioning activities and the effort for conduct of a HAZOP.
- For illustration reasons and due to the fact that some national regulations may require so, a detailed hazard analysis and evaluation on base of the results of the previous hazard analysis is performed. Fit for purpose, but still easy to use generic models are used for calculation of the consequences for workers and the public. Hereby deterministic and conservative scenarios are used.

In addition, no risk calculations were performed as the hazards and the related consequences are low so that no risk approach and the related effort could be justified. Accordingly, no related risk criteria are applied but compliance with dose criteria is regarded to be appropriate.

9. CONFIDENCE BUILDING IN THE SAFETY ASSESSMENT

In the following Section the measures to build confidence in the reliability of the performed safety assessment are presented. In a real safety assessment documentation all measure, including the relevant procedures of the safety management system, will be explained which are applied during the development of the safety assessment and which will be applied during the conduct of the decommissioning activities which were subject to the safety assessment. In the case of this safety assessment, no fictional safety management system is described, but the measures to ensure correctness and high quality in applying the DeSa safety assessment methodology by the Research Reactor Test Case Working Group are explained.

9.1 QUALITY MANAGEMENT SYSTEM

9.1.1. Quality management with regard to the development of the safety assessment

The safety assessment was performed by a relatively small group of international experts during the DeSa project. For the purpose of the Research Reactor Test Case it was assumed that a management system was in place at the reactor site. That is why the management system was nod discussed in this report but some features of the management system can be recognized though.

A common element of a management system is traceability of data. For calculations, an additional requirement is that the result can be reproduced when using the same dataset. A common method to reduce errors or mistakes is review of the work by an independent expert.

The starting point for the assessment was the description of a real decommissioning project, but some data were adapted to the needs of this test case. The calculation of consequences (see Sections 4 and 5) was performed with validated software, which is fit for this type of calculations.

9.1.2. Quality management with regard to the decommissioning of the research reactor

For the purpose of this test case it is assumed that a fit for purpose quality management or safety management system is in place and described in related supplementary documents that give confidence that all proposed safety controls and additional measures will be available and taken into account during work conduct.

9.2. INDEPENDENT REVIEW AND APPROVAL PROCESS

The system includes measures to build confidence in the safety assessment and its results and therefore includes appropriate review processes (by the safety assessment team and by independent reviews – but not by the Regulatory Body). Reviews of the Research Reactor Test Case report were performed according to clear procedures and recognizing clear responsibilities.

For this test case no dedicated system for an independent review was developed and applied. Instead following measures were performed:

— The starting points of scenarios, calculations and results were checked and reviewed by members of the working group itself in several iterative processes, mainly in the central working group meeting once a year.

— In addition, reviews were performed by participants of the DeSa project belonging to the Regulatory Review Working Group and to the Graded Approach Working Group. As far as these reviews result in recommendations on consistency and clearness of the report they were taken into account in this report. The results of both review processes are summarized in Section 10 “Summary and Lessons Learned”.

In summary, the various reviews performed during the development of the safety assessment for the decommissioning of the research reactor provided significant recommendations. These were taken into account in the current safety assessment report, i.e., in the main report. In a real decommissioning project this safety assessment documentation would be submitted to the Regulatory Body for consideration and approval in the context of the authorization process on the proposed decommissioning.

10. SUMMARY AND LESSONS LEARNED

Part B of Annex I documents the safety assessment for decommissioning of a relatively small research reactor. The test case was based on a real research reactor of thermal power of 2 kW.

This assessment was performed to illustrate the application of a new safety assessment methodology proposed by the DeSa project (main report). The methodology comprises of a general steps that need to be followed and of a concept on grading the efforts during the development and review of safety assessment.

With respect to the DeSa safety assessment methodology and the graded approach, the research reactor of the test case represents a nuclear facility which required a low level of detail concerning the safety assessment due to the low radioactive inventory (less than 5×10^9 Bq) and the research reactor's low complexity. Thus, this test case allowed a maximum grading.

The conduct of the safety assessment showed, that the safety assessment methodology is applicable to such facilities. It also demonstrated that graded approach can be applied and that requirements for the safety assessment can be complied without compromising the quality and safety. Grading could be applied to the site and facility description and to the selection of tools and models used to identify, analyse and evaluate hazards. Only available tools were applied and generic models and related computer codes were used.

Depending on the national regulations and requirements of the Regulatory Body further grading than that presented might be possible, e. g by omitting the more detailed hazard analysis and evaluation (Section 5) as part of the DeSa safety assessment methodology. In some cases the results of the preliminary hazard assessment and screening might be regarded to be of sufficient quality and significance for this nuclear facility with its low radioactive inventory and less complexity. The hazard analysis and evaluation (Section 5) is mainly provided for illustration on grading during this safety assessment step and thus corresponds to those national regulations which require more detailed analysis.

The application of the safety assessment methodology and the recommendations on the graded approach in the Research Reactor Test Case were subject to a review of the Graded Approach Working Group (Annex II) and the Regulatory Review Working Group methodology (Annex III). The results of the Graded Approach Working Group confirmed the application of the graded approach and included recommendations related to the need for additional information. The Regulatory Review Working Group provided recommendations to improve the quality of this report and of the safety assessment itself. Emphasis was also laid on the elimination of missing information due to the character of the test case.

APPENDIX I: DETAILED SCHEDULE OF THE DECOMMISSIONING ACTIVITIES

Figure I.1. provides details on the schedule of the planned decommissioning activities at the research reactor.

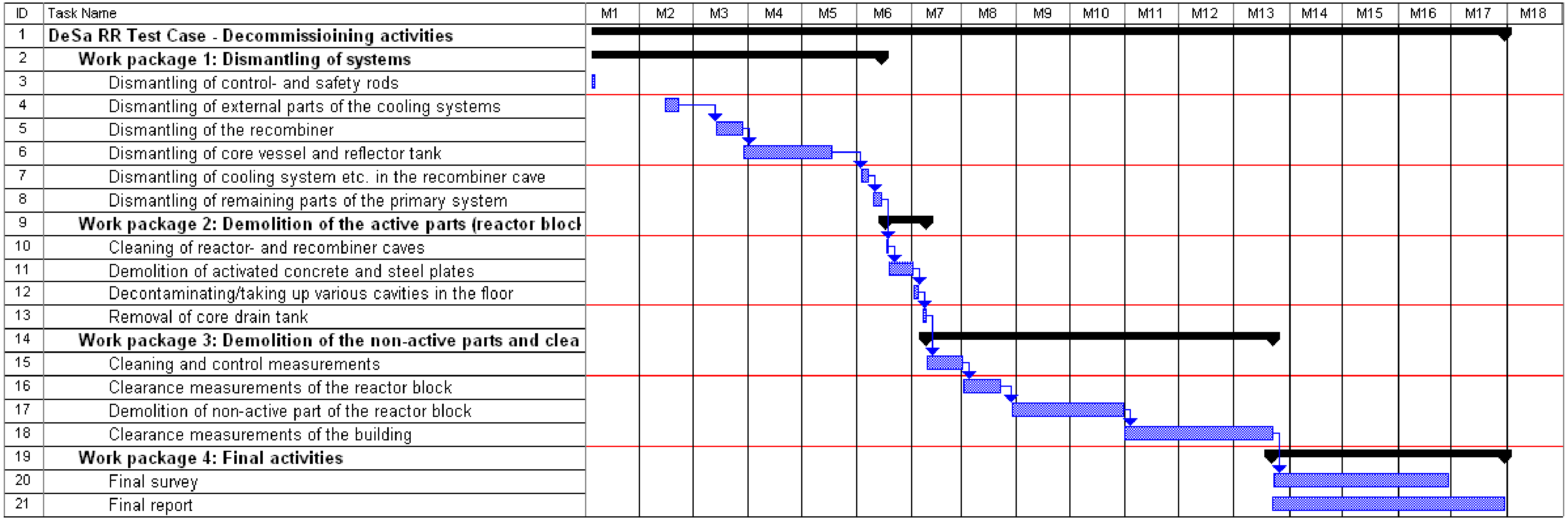


FIG. I.1. Decommissioning schedule.

Further information on the detailed work planning, including information on the expected duration of individual tasks can be obtained from an EXCEL document (Appendix VII).

APPENDIX II: DESCRIPTION OF THE RESEARCH REACTOR

Note: Depending on the national regulatory system a safety assessment report can be supported by a set of additional documents, providing inter alia information on the research reactor to be decommissioned. For the Research Reactor Test Case these documents do not exist. Therefore in the following Sections an example of such information on the research reactor to be decommissioned is provided.

The research reactor consists of a ball-shaped stainless steel vessel (the core vessel) with a diameter of 32 cm (see Figures 21 and 22 and also Figure 15). Around the core vessel is a graphite reflector in a cylindrical steel tank, called the reflector tank, with a diameter of 1.5 m and a height of 1.3 m. On its sides, the reactor is shielded by a l.2 m thick heavy concrete wall, while on top the shield consists of 85 cm thick concrete blocks.

The spherical core vessel is made of stainless steel with a diameter of 31.8 cm and a wall thickness of 1.78 mm. It is provided with a 5 m long helical cooling coil (outer tube diameter 9.5 mm and wall thickness 0.89 mm) through which the cooling water from the reactor cooling system could be circulated (see Figures 23 and 28 and also Figure 10). The core vessel is connected to a stainless steel tube (diameter 6.0 cm) running up to the gas recombiner, which is situated outside the reflector tank, but is inside the concrete shield in the recombiner vault (see Figures 24 and 28 and also Figure 11). Here the hydrogen and oxygen, produced by hydrolysis of water during reactor operation, is recombined. Recombination is effected by means of a platinum catalyst heated to 70-100°C. Together the core vessel, the recombiner and the connecting pipe form a closed system kept at a negative pressure.

The core vessel is placed in the middle of a cylindrical graphite reflector (see Figures 23 and 25 and also Figure 15). The diameter of the reflector is 152 cm and the height 135 cm. The reflector consists of graphite stringers with a cross section of 10.2×10.2 cm. The density of the graphite is 1.6 to 1.7 g/cm^3 and the total graphite mass is about 4 Mg. The reflector is contained in a steel tank with a wall thickness of 7 mm.

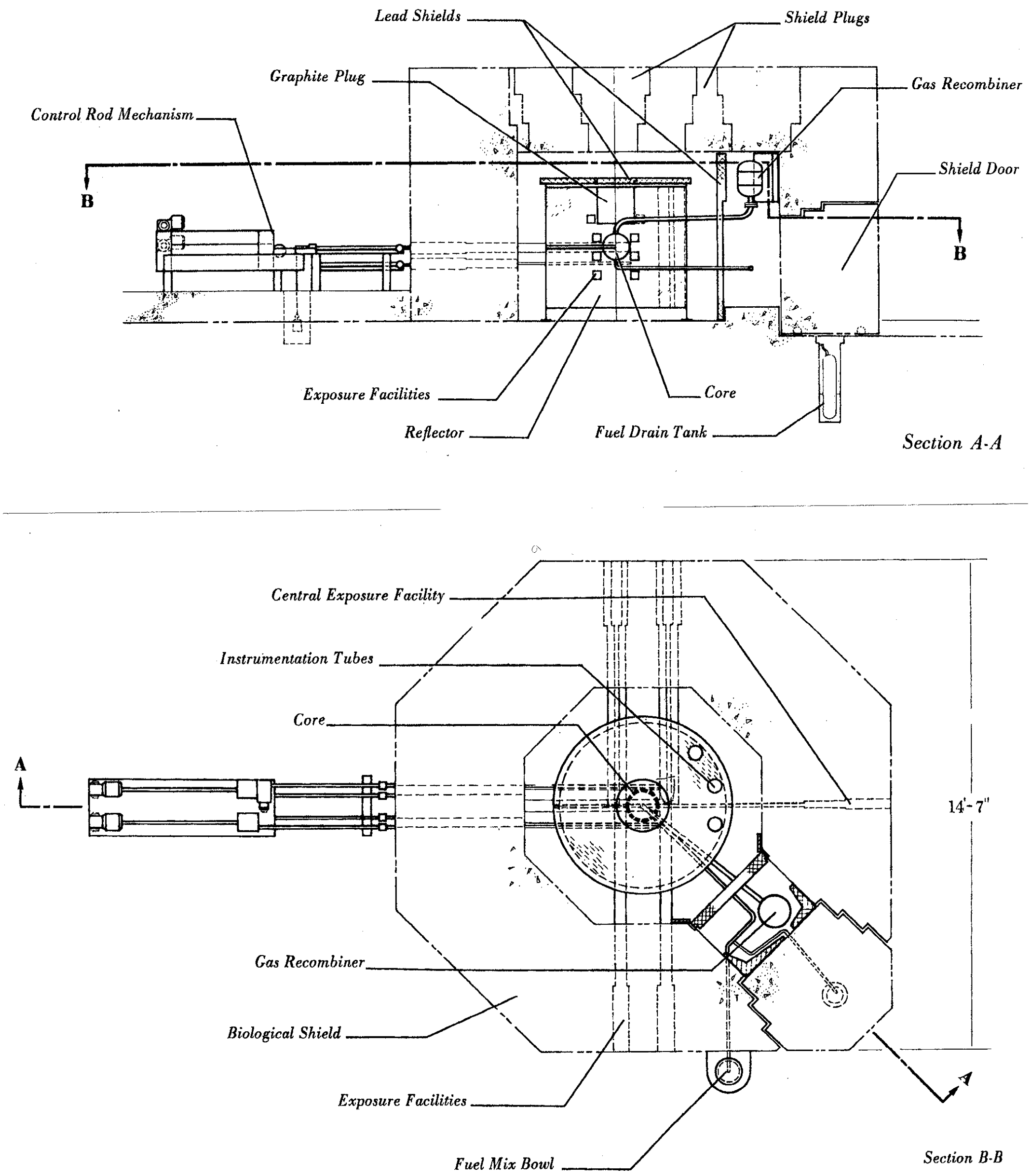


FIG. 21. Vertical and horizontal cross-sections of the research reactor.

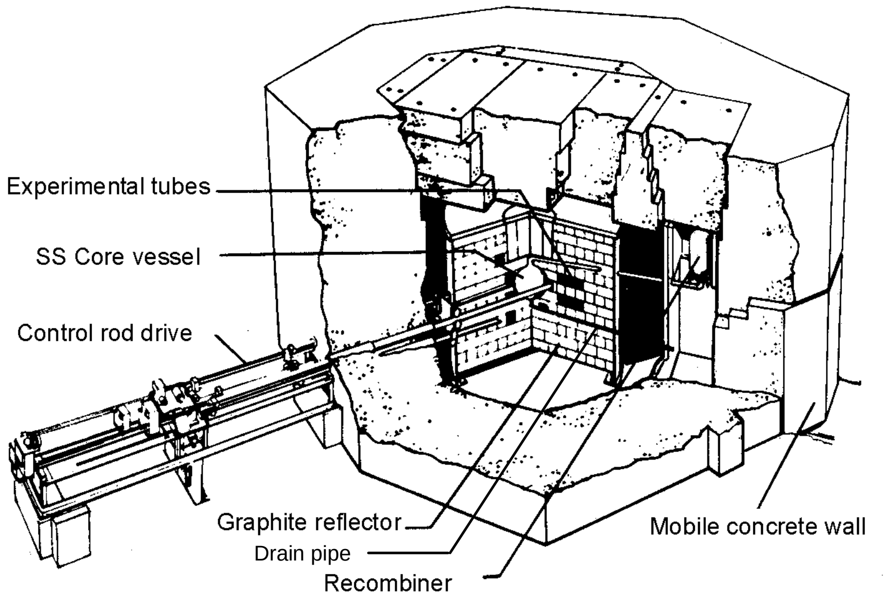
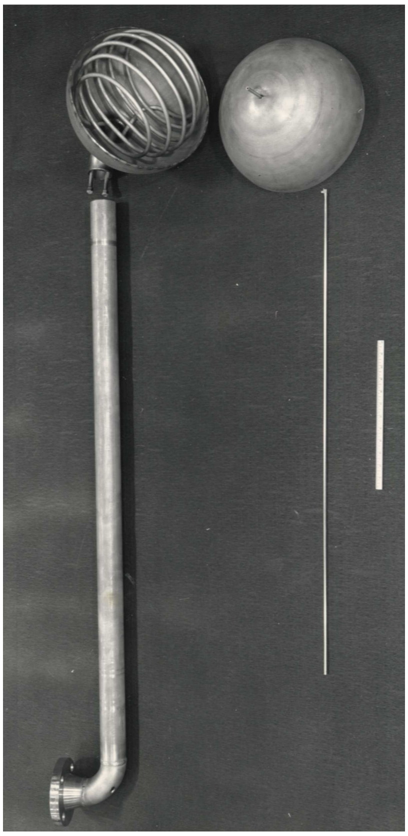


FIG. 22. Sketch of the structure of the reactor.



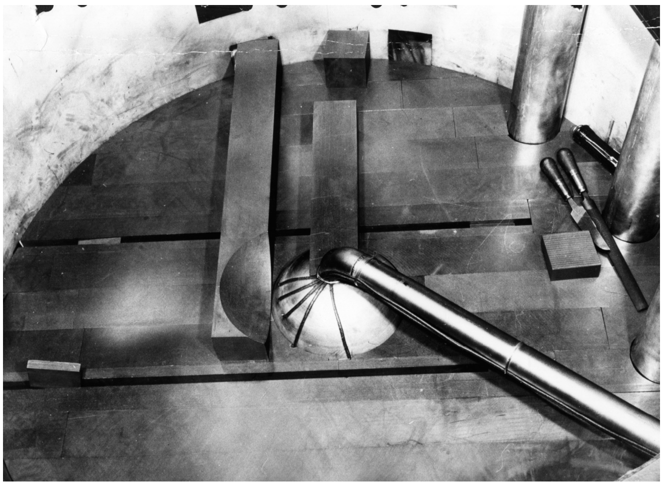


FIG. 23. Details on the core vessel (left: core vessel internals; right: during installation of the reactor).

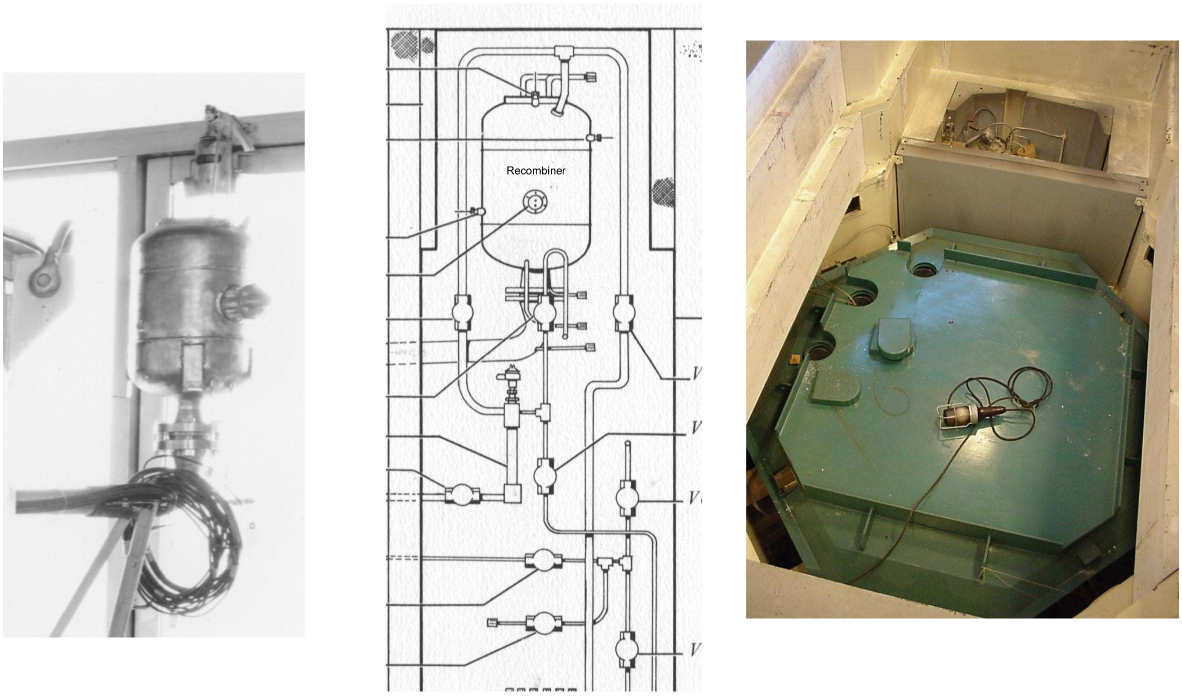


FIG. 24. Details on the recombiner (left: before assembling; middle: pipe plan; right: view on reflector tank (with steal lid) and recombiner vault (at upper part of the picture)).

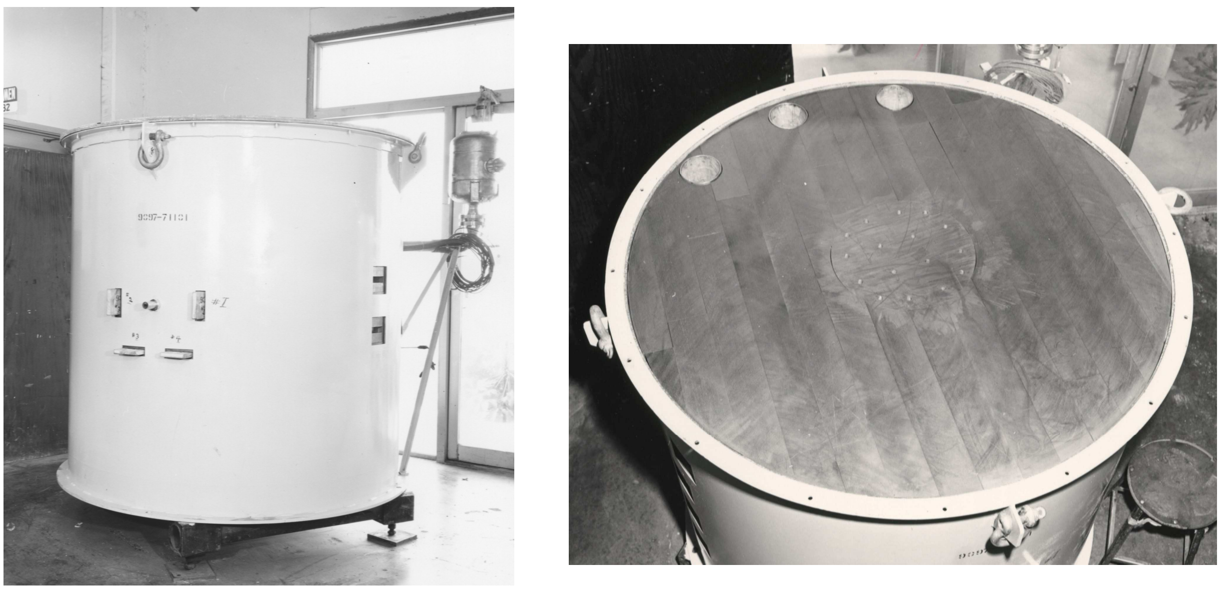


FIG. 25. Reflector tank (left: before assembling; right: with graphite stringers before assembling (note: for assembling the graphite stringers were removed again)).

The reactor is surrounded by an octagonal shield of magnetite concrete (heavy concrete) with a density of 3.6 g/cm^3. The horizontal thickness of the shield is 1 to 1.2 m, while the thickness above the reactor is about 85 cm. The shielding above the reactor consists of concrete blocks, which can be removed without major problems. Initially the graphite reflector was covered by a circular lead lid. This lid was removed in connection with the construction of the EXPO experiment as described in Section 3.4. Instead, a steel lid was placed on top of the reflector tank substituting the lead lid (see Figure 24).

The reactor is controlled by four control rods, two regulating rods and two safety rods (see Figures 12, 13 and 14). All rods are of the same design. They are moved horizontally in the reflector tank just outside the core vessel. The absorber part consists of flattened stainless steel tubes containing boron carbide. The absorber part is 125 cm long, 10 cm wide and 1.3 cm thick. Each rod governs approximately l.5 % reactivity. The control rod drive mechanisms are situated in the control rod house at the eastern end of the reactor. From here the control rods are inserted into the graphite reflector close to the core.

The reactor is provided with a number of beam tubes as shown in Figure 26. The beam tube numbers used in this report are given in Figure 26. Four of the tubes, S1, S10, V1 and N1, are not beam tubes. S1, V1 and N1 tubes enter the reactor cavity just above the top of the reflector tank, and S10 tube stops at the surface of the reflector tank. The purpose of these tubes is not known.

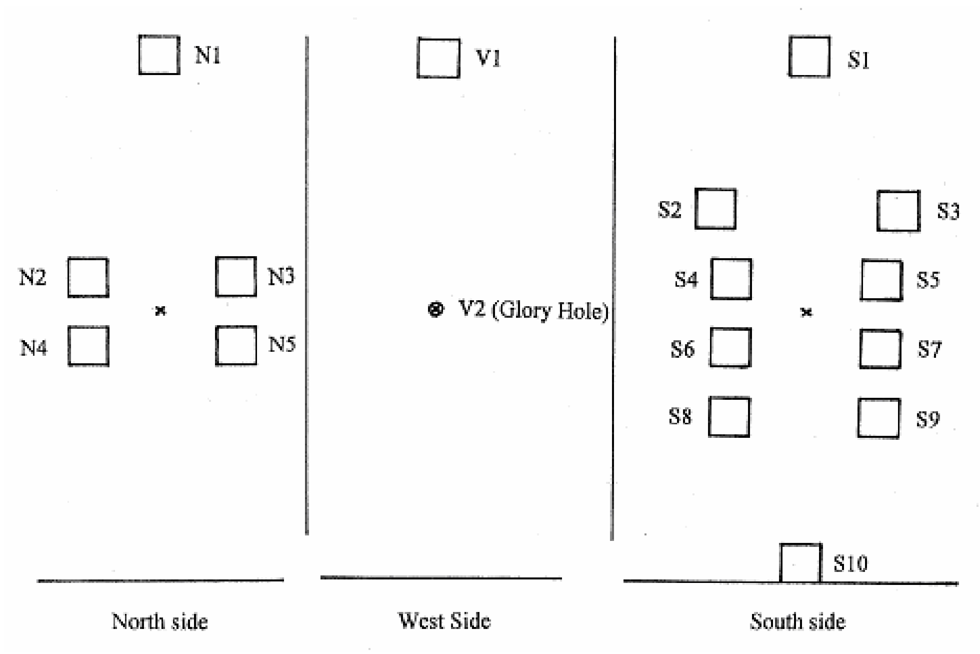


FIG. 26. The research reactor beam tubes on the north, west and south side. The core centre is indicated by a cross.

One of the beam tubes, the “Glory Hole” or V2, is placed on the western side of the reactor. It passes all the way through the reactor tank, which is provided with a stainless through-tube. This tube has an outer diameter of 3.33 cm and a wall thickness of 1.65 mm. An aluminium tube with an outer diameter of 2.85 cm and a wall thickness of 1.47 mm is placed inside the steel tube. A pipe with a 2.54 cm diameter goes horizontally through the centre of the core vessel. At a thermal power of 2 kW the maximum thermal flux in the pipe is approximately $6 \times l0^{10}$ n/(cm^2 s.).

The beam plug of the “Glory Hole” is a 116 cm long steel tube with a step reduction of its diameter from 6.0 cm at the outer part (length 62 cm) to 2.2 cm at the inner part (length 54 cm). The steel tube is filled with concrete.

The other beam tubes, which are placed at the northern and southern sides of the reactor, were initially provided with two beam plugs, an inner and an outer, through the concrete shield. The outer has a cross section of 15 × 15 cm and a length of 55 cm. The dimensions of the inner are 10 × 10 × 60 cm. They consist of steel boxes (wall thickness about 3.5 mm) filled with concrete. Inside the concrete shield the beam tubes continue into the graphite reflector. Each tube is in line with a graphite stringer, which can be pulled out through the beam tube. This gives access to areas near the core with high neutron flux.

When experiments were inserted into the reactor the original beam plugs were usually replaced by new ones. Often the corresponding graphite stringers were also removed. For this reason there are surplus beam plugs and graphite stringers in the reactor hall. The beam tubes S2, S3, S5 and N4 have all been provided with new beam plugs.

The neutron flux at the centre of the core, i.e. in the “Glory Hole”, was about:

— 1×10^8 n/cm^2/s at 5 W;

— 1×10^{10} n/cm^2/s at 500 W; and

— 4×10^{10} n/cm^2/s at 2 kW.

The flux distribution through the “Glory Hole”, the measured data and the calculated data are presented in Figure 27. Curve “I” was measured with indium foils with and without cadmium cover, and the measurements with cadmium cover was subtracted from the measurements without to get the thermal flux. Curve II was measured by activation of a copper wire placed in the “Glory Hole”.

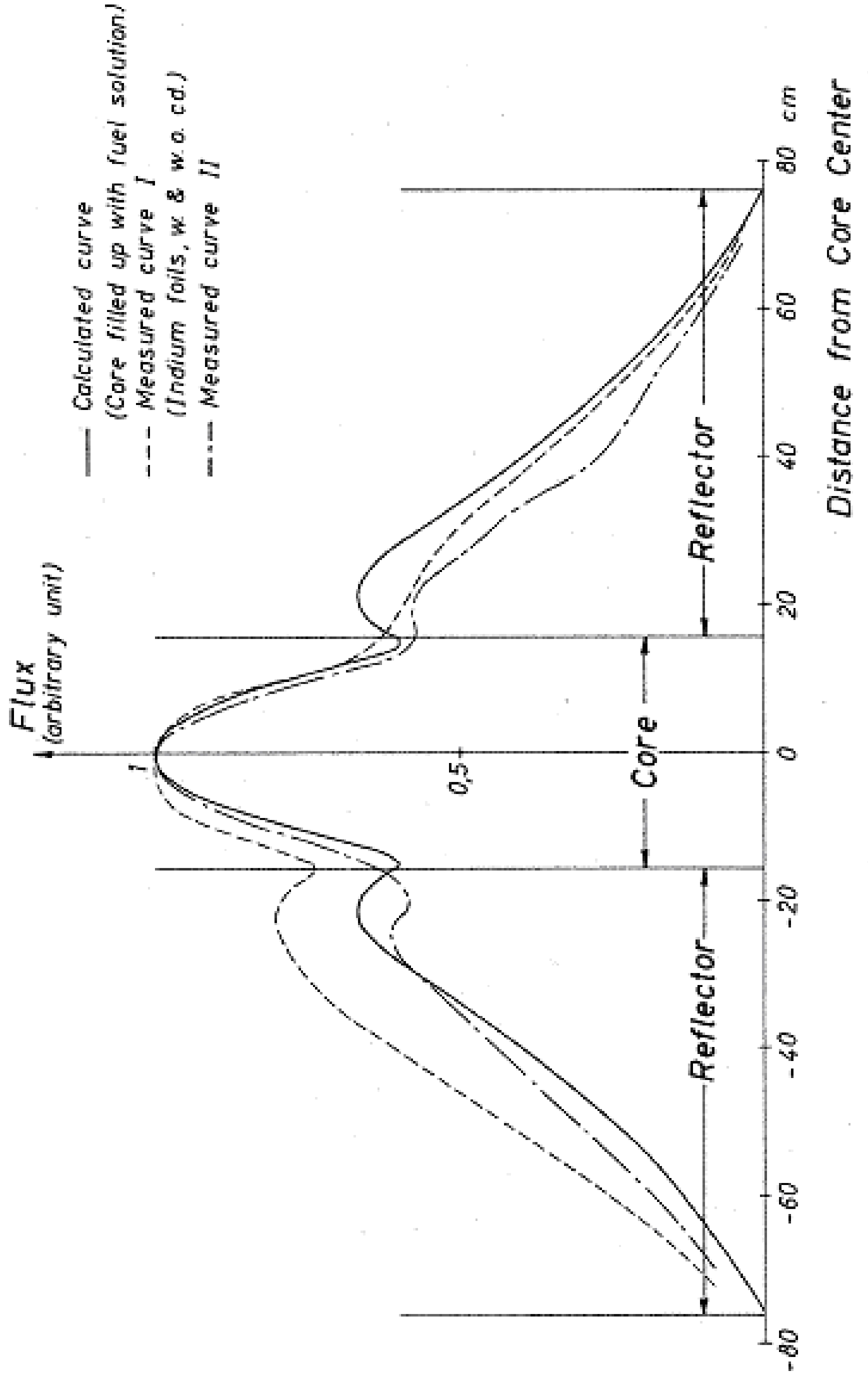


FIG. 27. The thermal flux distribution through the research reactor.

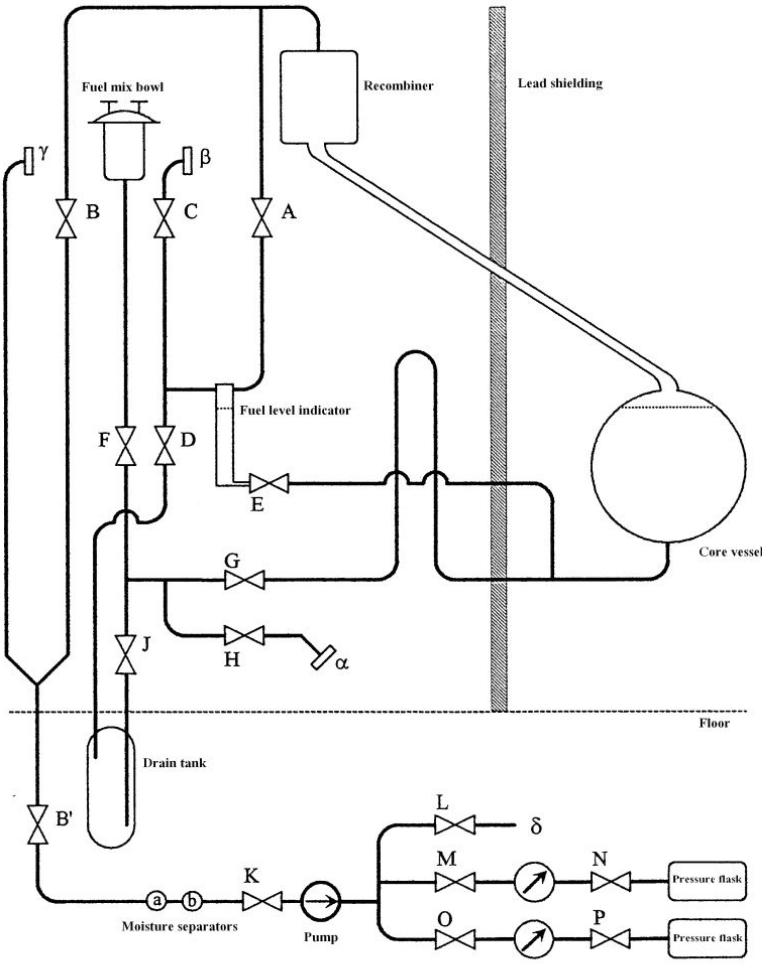


FIG. 28. Block diagram of the primary core system.

Figure 28 shows the components and piping connections in the primary system. The Fuel Mix Bowl was used for entering fuel solution into the reactor. Apart from the initial filling of the reactor this was only relevant once a year when samples of the fuel solution, which had been taken the previous year, were returned to the system. Once the fuel had been entered into the Fuel Mix Bowl it was transferred to the core vessel by means of helium pressure after opening of valves F and G. The samples of the core solution that were taken once a year were taken at the pipe stub marked α, applying helium pressure to the core vessel via pipe stub γ with valve B open. The Drain Tank was intended for use in case a leak occurred in the primary system; it could accommodate all the fuel solution from the core vessel in a criticality-safe geometry (in addition, it was wrapped with a sheath of cadmium). No records exist that suggest that the drain tank had ever been in use. The moisture separators, the pump and the pressure flasks, shown at the bottom of the Figure 28, comprise the "Fission gas station" where fission gases collected in the core vessel and recombiner were transferred once a year in order to reduce the pressure in the system. The transfer took place via valves B and B'; the latter valve had been installed after it was suspected that valve B was not completely tight.

In 1959, the reactor was equipped with two independent cooling systems, cooling the core and the recombiner, respectively. Each of the systems consists of a primary system and a secondary system. The primary systems contain demineralized water and the secondary systems are connected to the domestic water system. A thermal sensor in the core cooling system governs the water flow in the secondary system by means of a valve, thereby ensuring that the temperature remains at the desired value of between 20 ^oC and 40 ^oC.

The recombiner cooling system removes the heat generated in the recombiner during recombination. The water flow in the secondary system is controlled manually.

The essential reactor instruments are located in the control room. The most important instruments are:

— The four independent neutron flux channels including a period meter; and

— The instruments for recording the temperature of the core vessel and the catalyst in the recombiner, as well as the pressure in the core vessel/recombiner.

Furthermore, the radiation level in the ventilation pipe from the reactor block and in the reactor hall, as well as the temperature of the cooling circuits are known.

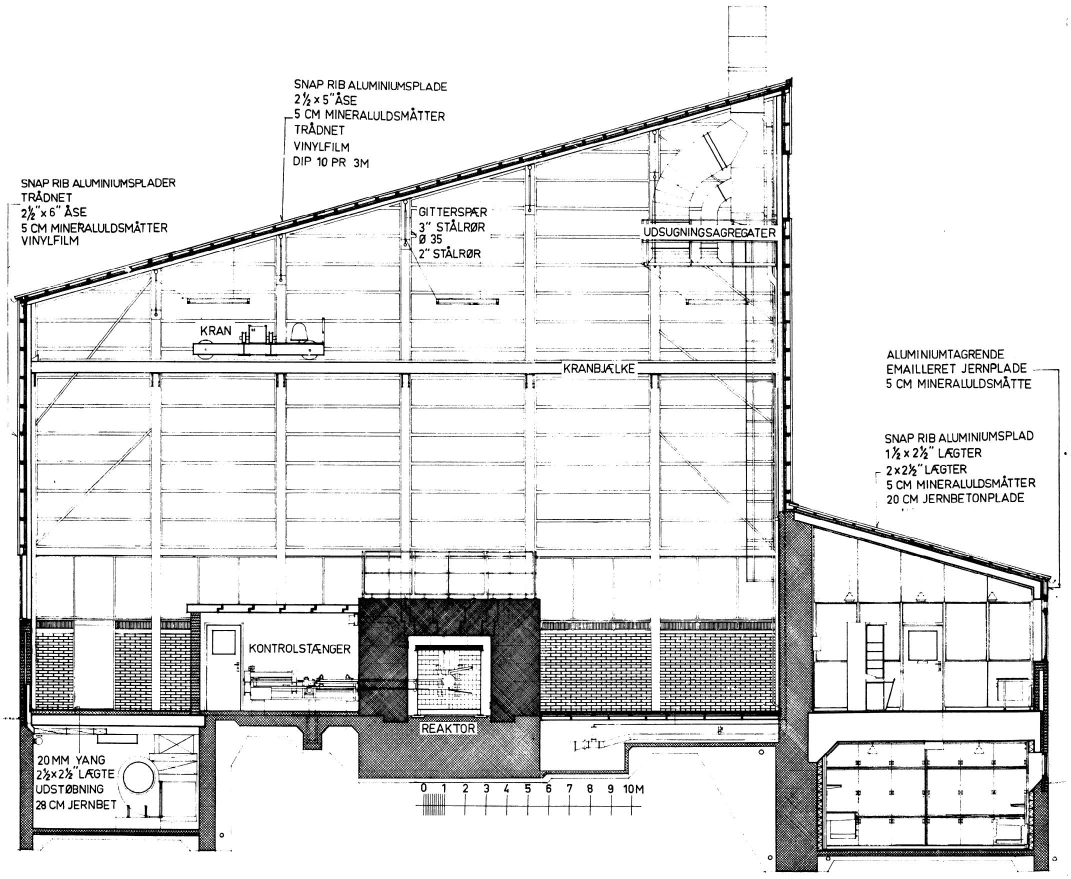


FIG. 29. Vertical section of the reactor building seen from the North.

The reactor building (see Figures 29, 30 and 31) consists of a reactor hall, a control room with an office and an entrance section, a counter room in the basement under the control room and an aggregate room for the air-conditioning system under the eastern end of the reactor hall.

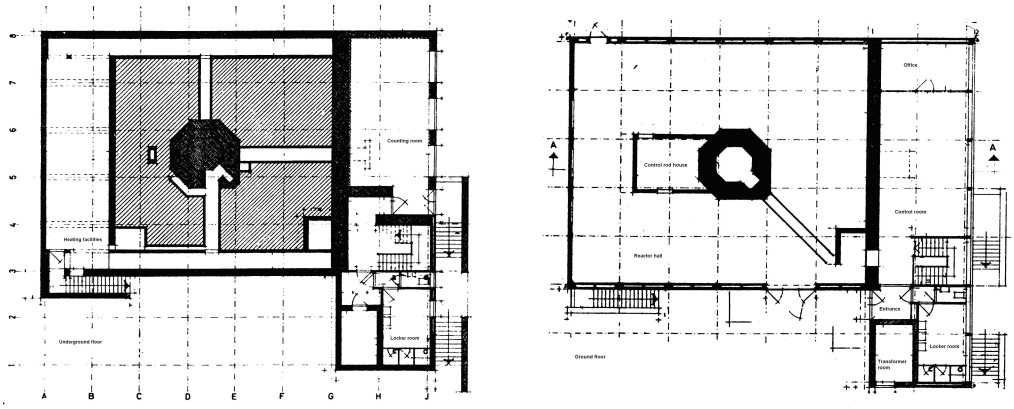


FIG. 30. Horizontal section of the underground floor and ground floor of the building.

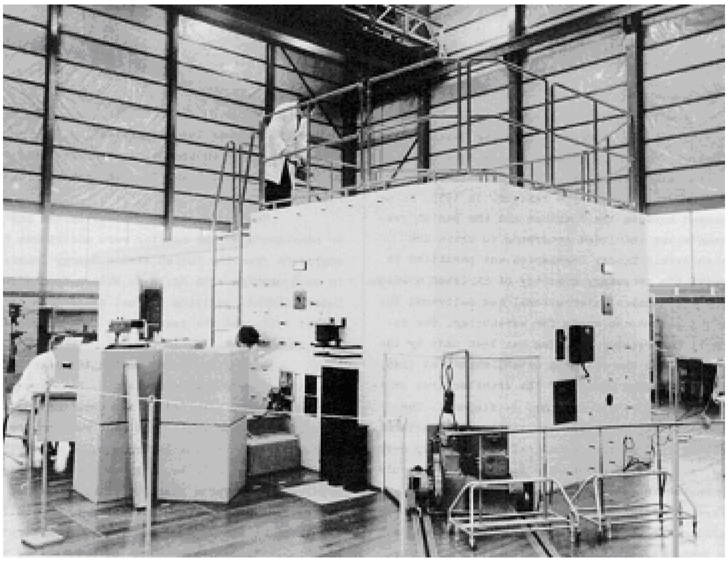


FIG. 31. Research reactor.

The air-conditioning system blows warm air through the floor ducts along the facades and from here through ducts in the hollow parapets to injection grates underneath the windows (see Figure 32). Under normal conditions, the ventilation capacity was 9 000 m^3/h, of which 6 000 m^3/h was recirculated. This means that fresh air intake corresponded to one exchange of air per hour. However, the ventilation was in general not used when the reactor was operating at low power levels, which was the case during the last 20 years of its lifetime. The 150 m^3/h ventilator in the channel below the floor, also known as the "Argon suction pump", was used to extract argon produced in the air around the reactor to the exhaust stack. The radiation from the air passing through the ventilator is measured constantly. A carbon filter has been installed in front of the ventilator. The carbon filter has been routinely exchanged biannually.

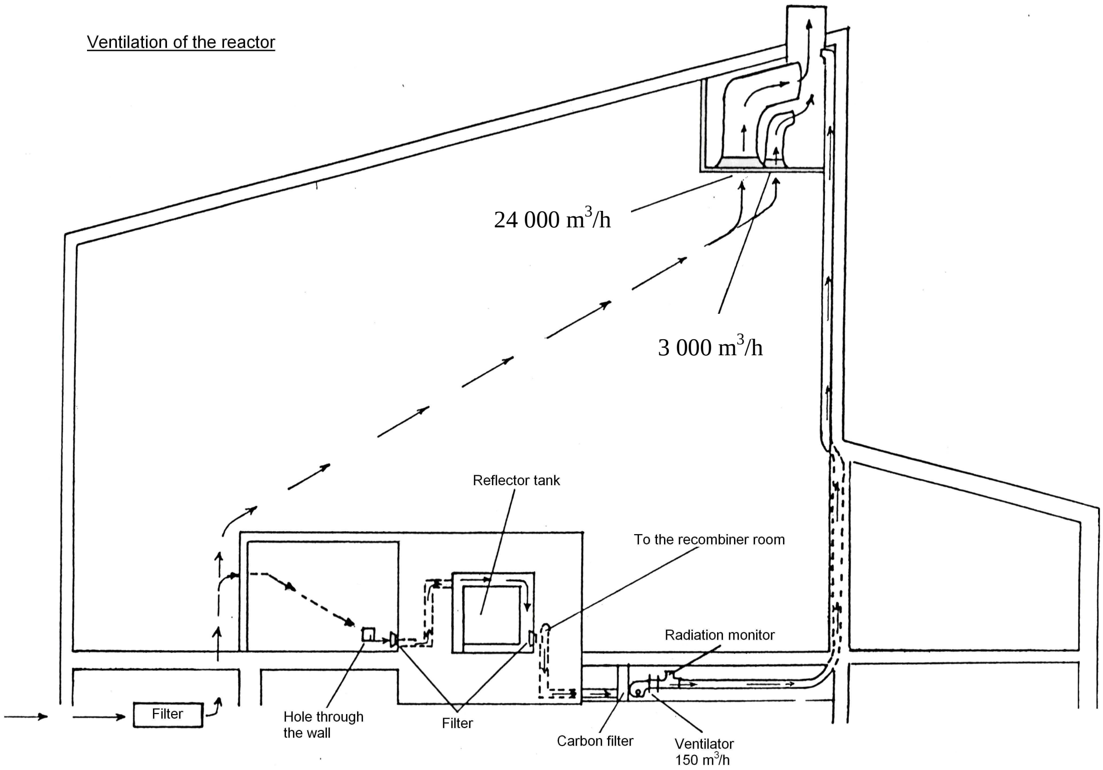


FIG. 33. Block diagram of the ventilation system.

Note, that further details on the research reactor can be obtained from Section 3.5, which contains further details to explain the decommissioning activities.

APPENDIX III: CHARACTERIZATION OF THE RADIOLOGICAL CONDITIONS

General remarks on the determination of the radiological conditions at the research reactor by performing an appropriate characterization in preparation of the decommissioning:

— When the characterization started, the core solution had already been transferred to another reactor at the site, and the core system had been flushed with water (see also Section 3.4.3). Thus, only a small, but unknown amount of fission products remained in the reactor. The results of the characterization are summarized below.

— All measurement systems used during radiological characterization are proven to be adequate, e.g. the detection limits are appropriate, the measurements are reliable in low and higher dose rates and the systems are calibrated and comply with relevant technical and quality standards.

— Based on experiences from similar decommissioning projects a dedicated β-analysis was performed for a few samples from the biological shield by a certified laboratory. All radionuclides, usually to be expected to occur in activated magnetite concrete (e.g. Fe-55, Ni-63), were considered but were found to be of no safety relevance (either with regard to exposure or to clearance due to missing or very low concentration).

III.1. MEASUREMENTS ON THE RECOMBINER

The first measurement aimed at determination of the radiation from the recombiner. It was performed by use of an Eberline RO20 monitor. The concrete block above the recombiner was removed and the γ-dose rate was measured in three distances (one above it and two below it). Next the γ-spectrum was measured with the Canberra spectrometer hanging above the recombiner in the hook of the crane. The results of the dose rate measurements are presented in Table 20.

TABLE 20. MEASUREMENT OF THE γ-DOSE RATE FROM THE RECOMBINER

Distance from Top of the Recombiner	**Dose Rate**	**Distance from Bottom of the Recombiner**	**Dose Rate**
[cm]	[μSv/h]	[cm]	[μSv/h]
50	320	-	-
100	150	100	90
150	70	50	220

The measurement with Canberra spectrometer yielded the following activity:

$$8.5 \times 10^{8}\ \mathrm{Bq}\ (^{137}\mathrm{Cs}).$$

The determination of the activity assumes that the source is a point source. Since this is not the case (the recombiner is about 50 cm high and its diameter is about 25 cm) the actual activity is expected to be larger.

III.2. COARSE OF MEASUREMENTS

To get an idea of the activities the workers need to handle, a number of quick, but less advanced measurements were performed. These include measurements of the dose rate at the surface of all easily removable beam plugs and graphite stringers in the reactor and in the reactor hall. The dose rate at the surface of the individual components was measured by use of a rate meter (Automess Szintomat 6134A) at both ends and at the middle of the component.

The first measurements were performed on beam plugs and graphite stringers in the reactor. The components were removed, measured, marked with the codes used in this report (usually by use of “Risø tape”) and returned to the reactor. The results of these measurements are given in Table 21. The statement ”< 0.1 μSv/h” used in that table means that the radiation is at the background level which in the control room is 0.06 to 0.08 μSv/h.

TABLE 21.”COARSE” MEASUREMENTS ON BEAM PLUGS AND GRAPHITE STRINGERS IN THE REACTOR

	Outer End	**Inner End**		**Outer**	**Middle**	**Inner**		**Outer**	**Middle**	**Inner**
[μSv/h]										
S1Y	< 0.1	< 0.1	S1I	< 0.1	0.1	1.1		-	-	-
S2	< 0.1	< 0.1		0.15	0.2			4	50	14
S3Y	< 0.1	< 0.1	S3I	< 0.1	0.1	2.5	Graphite	n.m	n.m	n.m
S4Y	< 0.1	< 0.1	S4I	< 0.1	< 0.1	0.9	Graphite	n.m	n.m	n.m
S5Y	< 0.1	< 0.1		< 0.1	0.15	6	-	-	-	-
			S6I	< 0.1	< 0.1	0.2	Graphite	n.m	n.m	n.m
			S7I	< 0.1	< 0.1	0.4	Graphite	n.m	n.m	n.m
			S8I	< 0.1	< 0.1	1.25	Graphite	0.3	4	7
			S9I	< 0.1	< 0.1	1		0.3	3	2
S10	< 0.1	< 0.1								
V1 Y	< 0.1	< 0.1	V1I	< 0.1	< 0.1	1		-	-	-
V2	< 0.1	< 0.1		< 0.1	< 0.1	0.1				
N1 Y	< 0.1	< 0.1	N1I	< 0.1	< 0.1	0.4		-	-	-
N2 Y	< 0.1	< 0.1	N2I	< 0.1	< 0.1	1.25	Graphite	0.7	4	2
N3 Y	< 0.1	< 0.1	N3I	< 0.1	< 0.1	1	Graphite	0.3	5	3
			N4I	< 0.1	< 0.1	3.5	Graphite	1.25	5	5
N5 Y	< 0.1	< 0.1	N5I	< 0.1	< 0.1	0.7	Graphite	0.45	6	4

TABLE 22. "COARSE" MEASUREMENTS ON BEAM PLUGS AND GRAPHITE STRINGERS STORED IN THE REACTOR HALL

	Outer Component	**Middle End**	**Inner End**
	[μSv/h]		
Inner beam plug A	< 0.1	< 0.1	< 0.1
Inner beam plug B	< 0.1	< 0.1	0.25
Inner beam plug C	< 0.1	< 0.1	< 0.1
Inner beam plug D	< 0.1	< 0.1	0.2-0.25
Graphite stringer G1	0.2	0.9	0.6-1
Graphite stringer G2	0.2	1.5	2
Graphite stringer G3	< 0.1	< 0.1	< 0.1
Graphite stringer G4	< 0.1	< 0.1	< 0.1
Graphite stringer G5	< 0.1	< 0.1	0.15

The code used in Tables 21 and 22 refer to the outer shielding plug as "Y", for the inner it as "I" and for the graphite stringer corresponding to the beam hole as "Graphite". The abbreviation "n.m." means "not measured".

Then a number of components stored in the reactor hall were measured. The results of these measurements are given in Table 22

It is seen from Tables 21 and 22 that the radiation levels are quite modest and that they are not expected to result in handling problems. Two of the beam plugs listed in Table 22 (A and C) are not active, so they were placed between the two cupboards at the south wall of the reactor hall. The two active beam plugs (B and D) were together with all the graphite stringers (G1, G2, G3, G4 and G5) placed in the locked concrete cell in the reactor hall.

III.3. MEASUREMENTS ON BEAM PLUGS AND GRAPHITE STRINGERS

Based on the coarse measurements a number of components were selected for more detailed examination, primarily those with the highest dose rates. The examination consisted of a γ-spectrum measurement from which the activities of the components could be determined by use of a computer programme. The distance between the γ-detector and the most active part of the component varied from 0.1 to 1 m, depending on the size of the component and its activity. The measuring time varied from 400 to 800 s. In the determination of the activities the self-shielding in the components was not taken into account, nor was the component rotated during the measurements.

The examination included an analysis of the γ-dose rate distribution along the component. This was carried out by use of a rate meter. In the cases where the γ-spectrum measurement indicated the presence of more than one radionuclide a number of measurements with the Canberra-spectrometer was also made along the component.

III.3.1. Outer Beam Plugs

As seen from Table 21 the measurements of the outer beam plugs all resulted in radiation levels corresponding to the background level. Thus no additional measurements were made on these components.

III.3.2. Inner Beam Plugs

The dose rate distribution along five of the inner beam plugs was determined. As seen from Figure III.1 the dose rates of all the plugs reached the background level 15-20 cm from the inner, most active end. It is also seen that the radiation level is higher for the inner plugs of N4 and S3 (N4I and S3I) than for the other plugs. The reason is undoubtedly that these two plugs were used in connection with an experiment (the pile oscillator in N4 and its ion chamber in S3). Both plugs were provided with a through hole and have presumably been made for the purpose. The other plugs from beam tube S8, N3 and N5 (S8I, N3I and N5I) are all the original plugs. Measurements were also made on the plug of V2, but since they all were at background level they have not been included in Figure 33.

The activity of the inner shielding plugs presented in Figure 33 and of the plug of the "Glory Hole" (V2) was also measured. The results are given in Table 23.

TABLE 23. γ-ACTIVITY OF SIX SHIELDING PLUGS

Plug	**Radionuclide**	**Activity [Bq]**
N3I	^{60}Co	2 870
N4I	^{60}Co	15 000
N5I	^{60}Co	925
S3I	^{60}Co	7 840
S8I	^{60}Co	3 610
V2	^{60}Co	0.1

The statistical uncertainty of the activity values of Table 23 is about 5%. From the comparison between the results of Table 23 and Figure 33 it is seen that the curves of S3I and N4I are the upper curves of Figure III.1 and their activities are also the highest. However, while the activity of N4I is almost twice as large as that of S3I, the curve of S3I is slightly above that of N4I. The activity of S8I is somewhat higher than that of N3I and the same is true for corresponding curves. N5I has the lowest activity and the bottom curve.

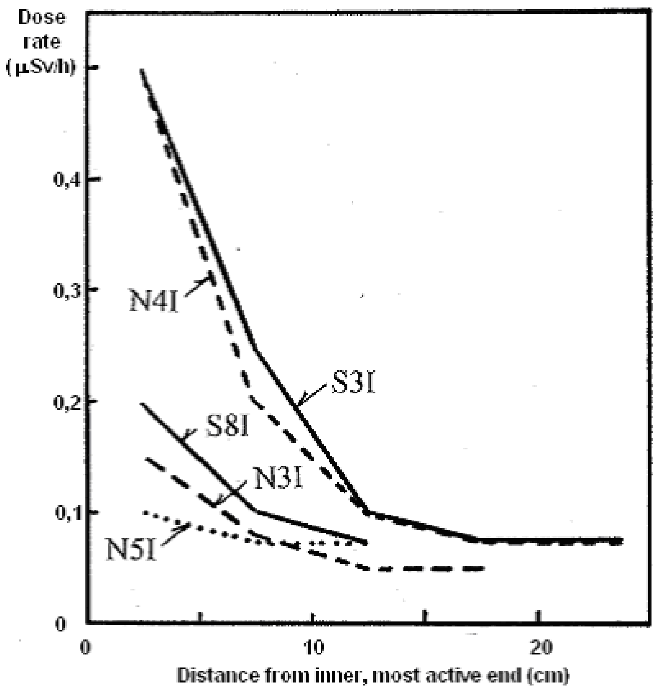


FIG. 33. γ-dose rate along the active end of five inner plugs.

III.3.3. Graphite stringers

Figure 34 gives the measured dose rate distributions of graphite stringers. Of these G2 and G5 were stored in the reactor hall. These stringers have undoubtedly only been in the reactor during a limited period of time. Therefore their activity is also limited. The graphite stringers from beam tube S8, N3 and N5 are the original stringers, but it needs to be mentioned that while the length of S8G and N5G is about 85 cm, the length of N3G is slightly less than 60 cm. Therefore, when comparing the curves, the N3G-curve needs to be parallel displaced about 25 cm to the right. Hereby reasonable agreement between the curves is achieved. The dose rates at the inner end of the stringers vary somewhat. S8G is above than N3G and N5G. This is presumably due the effect of the control rods. LGS is the stringer removed from the graphite plug at the top of the reflector. It is only about 40 cm long, but its distribution is in reasonable agreement with the other, original stringers. N4G is the stringer from the pile oscillator beam tube, and this is presumable the reason for its deviating distribution.

The activity of the five graphite stringers of Figure 34, which were in use for a longer time, was measured and the results are presented in Table 24. The uncertainty of the activities given in Table 21 due to counting statistics is about 2%. A comparison between the results of Table 21 and of Figure 34 shows that G2 give the lowest values in both Table 20 and Figure 20 (no γ-spectrum measurement was made on G5 and LGS). Next follows N3G. N4G, N5G and S8G have according to Table 21 about the same activity, while inner, most active end of S8G has a dose rate significantly higher than the other two according to Figure 34. Therefore it was to be expected that S8G had the highest activity.

TABLE 24. γ-ACTIVITY OF FIVE GRAPHITE STRINGERS

Stringer	Radionuclide	Activity [kBq]
G2	^{152}Eu	119
N3G	^{152}Eu	153
N4G	^{152}Eu	409
N5G	^{152}Eu	429
S8G	^{152}Eu	422

III.3.4. Control rods

Figure 35 gives the dose rate distribution along the two upper control rods, Ø2 (regulating rod) and Ø3 (safety rod). The inner, most active ends of the two rods, has as expected rather high dose rates (up to 100 µSv/h), but the dose rate drops rapidly with the distance from the inner end. The regulating rod Ø2 has higher dose rates than the safety rod Ø3. The reason is undoubtedly that the regulating rod is close to the core all the time during operation, while the safety rod is situated further out. Note the vertical drop of the Ø3-curve around 80 cm. The reason for the drop is that it was necessary at this point to increase the distance between the detector and the safety rod at the collimator from 6 to 10 cm.

The activity of the two control rods (see Figure 35) was measured and the result is presented in Table 25. The statistical uncertainty of the activities is about 1.5 %.

TABLE 25. γ-ACTIVITY OF TWO CONTROL RODS

Rod	Radionuclide	Activity [MBq]
Regulating rod Ø2	^{60}Co	15.5
Safety rod Ø3	^{60}Co	5.1

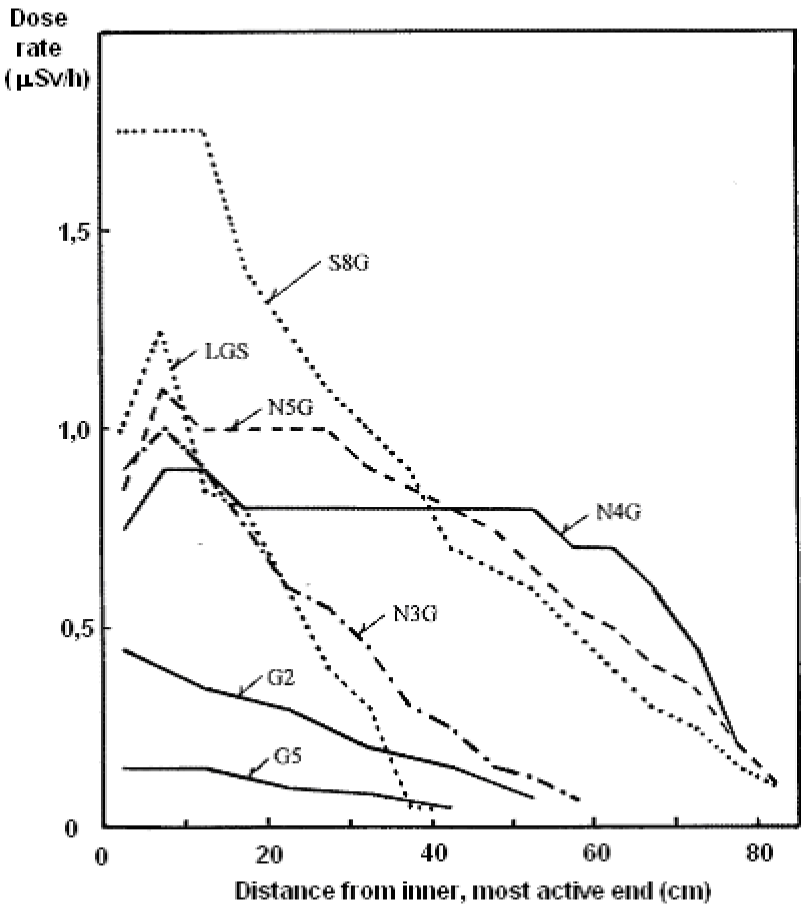


FIG. 34. γ-dose rate along seven graphite stringers.

Initially it was the intention to measure the activity of and the dose rate distribution along both regulating rods. However, it was not possible to remove regulating rod Ø5 without cutting up the control rod drive mechanism. Instead it was possible to compare rate meter measurements with the rate meter in contact with the inner, most active ends of all three rods. Here 75 mSv/h was measured for the regulating rod Ø2, 80 mSv/h for regulating rod Ø5 and 40 mSv/h for safety rod Ø3. Thus there is amble reason to expect that the activity and the dose rate distribution of the regulating rod Ø5are practically identical to that of the regulating rod Ø2.

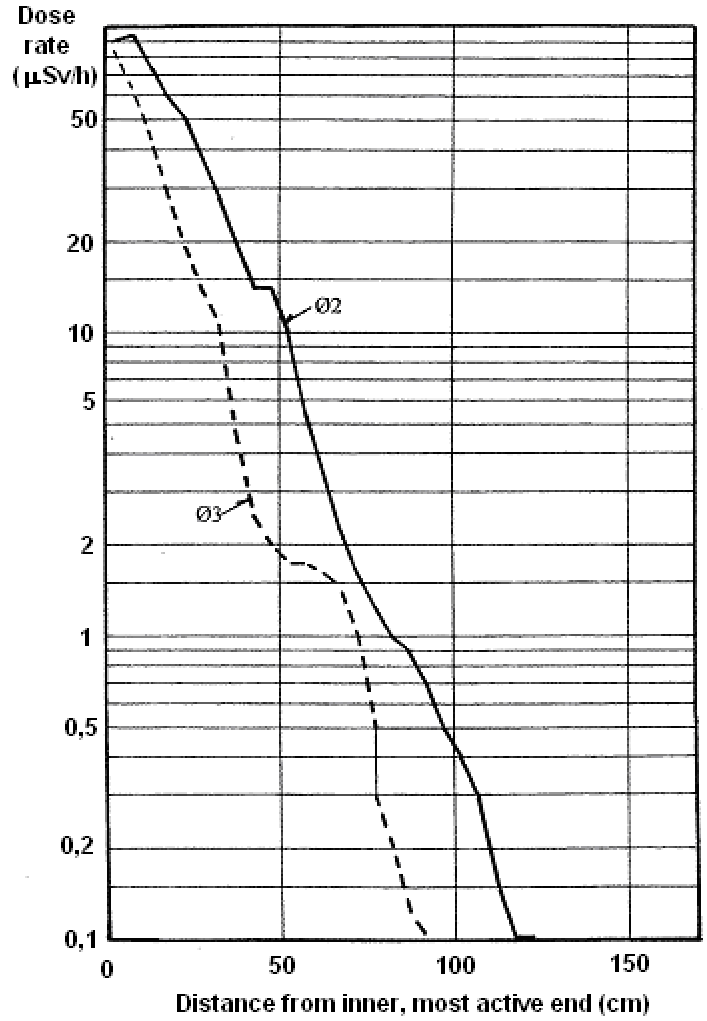


FIG. 35. γ-dose rate along a regulating rod (Ø2) and a safety rod (Ø3).

III.3.5. Experimental plugs

Finally measurements were made on four experimental plugs that had been manufactured to permit experiments, i.e. the plugs of S2, S3, S5 and N4. S2 contains a long plug, which extends from the outer surface of the reactor into the graphite reflector. The plug consists of (starting from the outer reactor surface) 10 cm lead, an air space, a paraffin layer, an air space, a paraffin layer and at the end in the graphite reflector of a long aluminium box. A tube with two electric wires passes all the way from the box to the reactor surface. The use of the plug S2 is unknown.

S3G is the “graphite stringer” of the plug S3. This beam tube contains the ion chamber of the pile oscillator, and the “stringer” consists of a short graphite stringer at the end of which is placed an aluminium box with the ion chamber. An electric cable runs from this box through holes in the graphite stringer and the beam plugs all the way out to the outer surface of the reactor. Plug S5 contains a hollow steel box with a length that is roughly equal to the thickness of the concrete shield. The use of the plug S5 is unknown. N4S is a steel rod, which was part of the pile oscillator.

The results of the measurements of the dose rate distribution along the four components are shown in Fig. 36.

It is seen that plug S2 gives dose rates up to 50 µSv/h while the maximum dose rate of S3G is 10 µSv/h and that of N4S 6 µSv/h. The measurements started at the inner, most active end and were stopped when the dose rate reached less than 0.10 µSv/h. The tube of plug S5 gave only modest dose rates.

The activity of the four components of Figure 36 was measured and the results are given in Table 26. While the components discussed so far have only contained one radionuclide, S2 and S3G contained two, ^{60}Co and ^{152}Eu.

TABLE 26. γ-ACTIVITIES OF EXPERIMENTAL PLUGS

Plug	**Radionuclide 1**	**Activity 1 [kBq]**	**Radionuclide 2**	**Activity 2 [kBq]**
S2	^{60}Co	2 400	^{152}Eu	167
S5	^{60}Co	43.4		-
S3G	^{60}Co	450	^{152}Eu	76.3
N4	^{60}Co	651		

The statistical uncertainty of the ^{60}Co activities (see Table 26) is about 2%, while for ^{152}Eu it is 4 to 12 %. The plug S2 also contained ^{65}Zn and ^{108m}Ag, but the counting time was too short to permit a quantitative determination of their activity.

A comparison between Table 26 demonstrates that plug S2 has the highest activity and dose rates. Next follows S3G and N4S, which have about the same activity, but somewhat different dose distribution curves and finally S5, which has the lowest activity and the lowest dose rate curve. As a result a good agreement is observed between the table and the figure.

As mentioned above measurements of the γ-spectrum and hence the activity were performed along some of the reactor components, e.g. when the activity measurements had shown that more than one radionuclide was present.

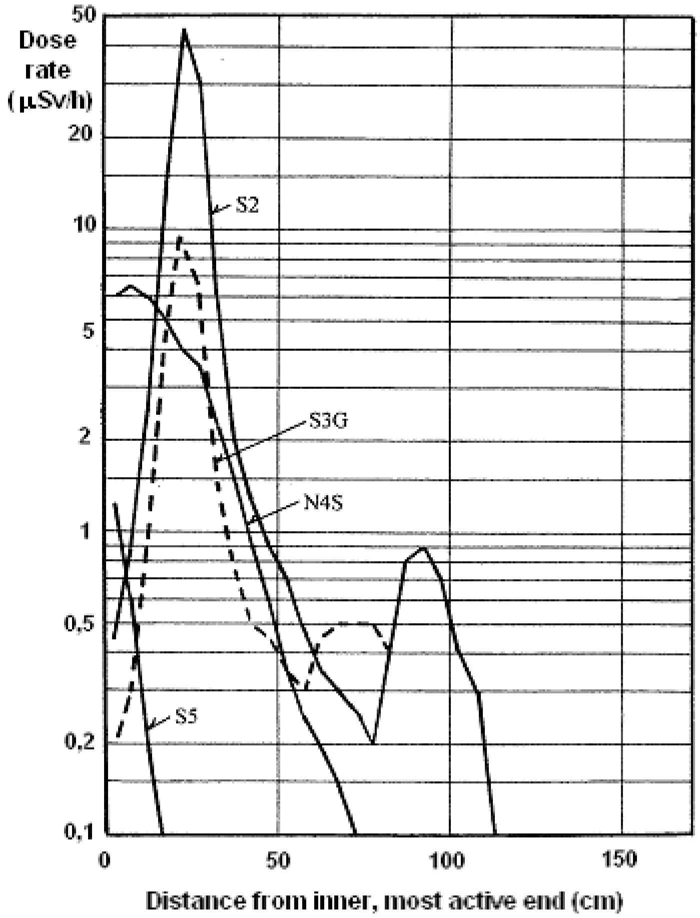


FIG 36. γ-dose rate along experimental plugs and the pile oscillator rod.

Figure 37 shows curves of the measured ^{60}Co and ^{152}Eu activities for S2 and S3G. The dose rate curves for the two components are also given. It can be seen that for both components there is reasonable agreement between dose rate and the corresponding ^{60}Co curves, even if there is no constant proportionality factor. The ^{152}Eu activity curve deviates from the other curves. It needs to be mentioned that ^{152}Eu is only present in parts of the two components.

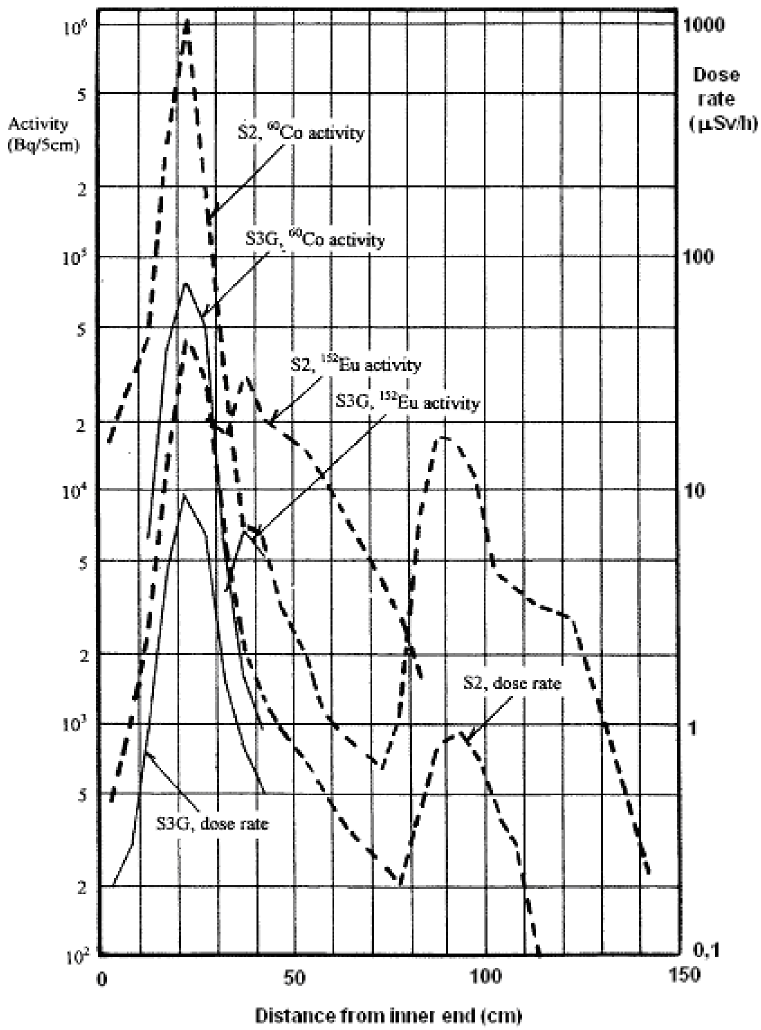


FIG 37. Activity and γ-dose rate distributions along S2 and S3G. ^{60}Co as well as ^{152}Eu activities are shown.[2]

III.3.6. Cored boreholes through the concrete shield

During the characterization three cored holes were initially drilled through the concrete shield of the research reactor:

— A vertical hole drilled down through the central block of the shield at the reactor top (drilling T);

— A horizontal hole through the western wall of the shield, parallel to and close to the “Glory Hole” (drilling V); and

— A horizontal hole through the southern wall at the same level as the centre of the core (drilling S).

[2] The unit "Bq/5cm" was used because the activity content was derived from measurements of 5 cm long segments of drill-cores. The diameter of the cores was also 5 cm, giving a volume of the segment of 98 cm^3. Thus 1 Bq/5 cm ~ 1 Bq/98 cm^3 ~ 0.01 Bq/cm^3.

The length of the cores of drilling T was 87 cm; of drilling V - 105 cm and of drilling S - 115 cm. During the drillings the cores were cut in pieces of varying lengths, and these core pieces were numbered with A for the innermost piece, B for the next, etc. After termination of the drilling, the pieces were placed on angle plates of aluminium in the correct order. For drilling T the pieces were called AT, BT, CT, DT and ET and they were all marked with their individual name by use of a marker pen. The same procedure was used for drilling V (AV, BV,. ....and HV) and for drilling S (AS, BS,.....and FS).

The activity of all pieces was then measured. The results showed that only AT (6 cm long) had a measurable activity of

$$5.29 \times 10^3 \text{ Bq } (^{60}\text{Co}).$$

AT had mass of 363 g and hence the specific activity was

$$14.6 \text{ Bq/g } (^{60}\text{Co}).$$

The activity measurements were performed with the Canberra spectrometer. The measuring time was in all cases 3 600 s, the measuring distance 1 m, and the source strength was calculated as if it was a point source.

The fact that activity could only be measured in the AT piece, but not in any of the others, in particular in BT, AV and AS was a surprise, but the reason probably is that the neutron flux in the concrete shield has been low and that the activation in horizontal direction has been reduced by the steel plate on the inside of part of the shield

To investigate the importance of the steel plate at drilling V and S, an extra drilling was later carried out through the south-western wall in direction of the reactor core centre (drilling SV). Here there is no steel plate. The length of drilling SV was 118 cm. The core pieces were named SVA, SVB, SVC, SVD, SVE, where SVA is the inner piece. The activity of SVA (32 cm long) was found to be

$$\text{about } 400 \text{ Bq } (^{60}\text{Co}),$$

i.e. significantly lower than AT, but in contrast to AV (11 cm) and AS (17 cm) with measurable activity. It needs to be mentioned that the measuring time for SVA was 72 000 s. The length of the pieces is of minor importance since the activity is situated at the inner end. The higher activity of SVA as compared to AV and AS may presumably be explained by the neutron absorption of the steel plate and the longer measuring time. However, the significant difference between the activity between AT and SVA was unexpected. It may be explained by a larger neutron leakage in vertical as compared to horizontal direction, since the height of the reflector is smaller than its diameter. Or it may be explained by a different composition of the horizontal and the top shield. On the other hand, the central plug in the top shield was not irradiated during the EXPO experiment (see Section 3.4.1).

III.4. OPENING OF THE TOP SHIELD OF THE REACTOR

The shielding plugs at the top of the reactor were removed. The dose rate at the bottom of the central shielding plug was found to be 8 μSv/h. This relatively high, measured dose rate at the bottom of the central shielding plug cannot be compared directly to the lower, measured dose rates at the inner end of the inner shielding plugs in the horizontal beam holes. The measured volume is significantly higher for the top plug and the neutron absorption in the wall of the reflector tank and the steel plate on the inner side of the concrete shield reduces the activation of the horizontal concrete shield.

The reactor top could then be accessed, and measurements of the dose rate at various places in the open reactor were carried out (see Table 27).

TABLE 27 RESULTS OF MEASUREMENTS AT THE REACTOR PARTS

Part of the reactor	**Dose rates [µSv/h]**
At the centre of the steel lid above the reflector tank	50
At the ion chambers at the rim of the steel lid:	20
1 m above the centre of the steel lid	0.10
1 m above the top of the reactor	3.5

The steel lid at the top of the reactor tank was lifted and placed at the floor of the reactor hall. A material sample of steel cuttings was drilled out of the centre of the steel lid (see Section III.5 and Table 29) and the radiation levels measured at the top and at bottom side of the steel lid are presented in Table 28.

TABLE 28. RESULTS OF MEASUREMENTS OF THE REACTOR LID

Bottom side	
Centre of lid	29 µSv/h
Lid rim	**15 µSv/h**
Top side	
Centre of lid	28 µSv/h
Lid rim	15 µSv/h

While the steel lid was removed the dose rate at the graphite surface of the reflector tank was measured - 25 µSv/h. One of the central, vertical graphite stringers was removed, measured and returned after a material sample of its lower, most active end had been taken (see graphite stringer LGS in Figure 34 and Table 29). While the vertical stringer was removed the dose rate at the top of the open hole in the reflector was measured to be 25 µSv/h. This increase in the dose rate is due to the fact that the rate meter was pointed at the more activated graphite, but probably also that it received a contribution to the dose rate from the reactor tank. After these measurements the reactor top was closed again.

III.5. COLLECTION OF MATERIAL SAMPLES

During the characterization a number of material samples were collected in addition to the borehole cores. The activity of these samples were determined by using a γ-spectrometer.

Samples were taken of the inner, most active ends of the graphite stringers in beam tube N3, N5 and S8, of one of the vertical stringers in the graphite plug at the top of the reflector (LGS) and of graphite stringer G2.

In addition, the inner and most active 10 cm of the pile oscillator rod (POS) was cut off as a material sample. As mentioned above steel cuttings were drilled out of the centre of the lid above the reflector tank (RTL). A disk was drilled out of the two steel plates situated on the inner side of the concrete shield on its western and southern part (SPV and SPS) and a disk was drilled out of the reflector tank on its southern side. The inner, most active 10 cm of the shielding plug of V2 or the “Glory Hole” was also cut off. Finally the inner 5 cm of the inner shielding plugs of beam tube S3 and S8 (S3I and S8I) were cut off as material samples.

After the cutting the activity of all material samples was determined by use of the γ-spectrometer. The result of these measurements is given in Table 29.

TABLE 29. γ-ACTIVITY OF MATERIAL SAMPLES FROM THE REACTOR

Materials	**Isotope**	**Activity [Bq]**	**Mass [g]**	**Specific Activity [Bq/g]**
Graphite sample, N3G	^{152}Eu	210	3.2	65
Graphite sample, N5G	^{152}EU	201	4.5	45
Graphite sample, S8G	^{152}Eu	730	6.3	115
Graphite sample, LGS	^{152}Eu	400	4.2	95
Graphite sample, G2	^{152}Eu	178	6.1	25
Steel, pile oscillator, POS	^{60}Co	185 000	87.1	2 120
Steel cuttings, reflector tank lid, RTL	^{60}Co	630	3.2	200
Steel plate, western shield wall, SPV	^{60}Co	3540	150.2	25
Steel plate, southern shield wall, SPS	^{60}Co	5 900	144	40
Steel, reflector tank, RTS	^{60}Co	18 600	70.1	265
Steel and concrete, V2 or Glory Hole *)	^{60}Co	170	150	1
Beam plug, S3I **)	^{60}Co	8 180	825	10
Beam plug, S3I **	^{152}Eu	520	1 000	0.5
Beam plug, S8I **)	^{60}Co	2320	825	3
Beam plug, S8I **)	^{152}Eu	280	1 000	0.3

*) The ^{60}Co-activity is assumed to originate from the steel only, not from the concrete. Thus “Mass” is the mass of steel.

**) The ^{60}Co-activity is assumed to originate from the steel, the ^{152}Eu-activity from the concrete.

It is seen that the activities are modest and they are not expected to give rise to significant consequences during the decommissioning.

It has to be noted, that in addition a β-analysis was performed for a few samples from the biological shield by a certified laboratory. Relevant radionuclides to be expected in the activated magnetite concrete were considered not of safety relevance.

APPENDIX IV: MEASUREMENTS FOR DETERMINATION OF THE RADIATION LEVEL FROM THE CORE VESSEL

Prior to removal of the active components from the reactor, measurements were carried out to determine the radiation levels from these components. The results formed part of the basis for the choice of dismantling methods. The core vessel is expected to be the component causing the highest radiation levels during the decommissioning of the reactor.

Through the centre of the core vessel an irradiation tube is running, in which items to be irradiated could be placed by means of perspex rods with small cavities for each 2 cm. This irradiation tube was used to determine the radiation level across the core vessel and 8 cm beyond on each side by placing TL dosimeters in the cavities of the perspex rod. Three measurements were carried out in order to determine the influence of the fuel solution and the Ra-Be source. The first measurement was made with the fuel solution still in the core vessel and the Ra-Be source in place. The second measurement was made after the fuel solution had been drained, but with the Ra-Be source still in place. Finally, the third measurement was made after the removal of the Ra-Be source, resulting in the dose rates that will be relevant during dismantling operations.

IV.1. FIRST MEASUREMENT (FUEL SOLUTION IN THE VESSEL AND THE RA-BE SOURCE IN PLACE)

Sixteen dosimeters were placed across the core vessel, placed in a standard irradiation rod. They remained inserted for one hour. The measured doses and dose rates are presented in Table 30, and the distribution of the dose rates across the vessel can be seen from Figure 38.

IV.2. SECOND MEASUREMENT (NO FUEL SOLUTION IN THE VESSEL, THE RA-BE SOURCE IN PLACE)

Fifteen dosimeters were placed across the core vessel, placed in a standard irradiation rod. They remained inserted for 23.5 h. The measured doses and dose rates are shown in Table 30 and the distribution of the dose rates across the vessel presented on Figure 38. The fact that the doses around the Ra-Be source are higher in this measurement than in the first one, probably can be attributed to a larger uncertainty in the first measurement due to the shorter irradiation time.

IV.3. THIRD MEASUREMENT (NO FUEL SOLUTION IN THE VESSEL, THE RA-BE SOURCE REMOVED)

Nineteen dosimeters were placed across the core vessel, placed in a standard irradiation rod. They remained inserted for 4.1 h. The measured doses and dose rates are shown in Table 30 and the distribution of the dose rates across the illustrated on Figure 38. The dose rates in this measurement reflect the radiation level from the core vessel itself.

TABLE 30. DOSE AND DOSE RATE DISTRIBUTION ACROSS THE CORE VESSEL FOR THE THREE SERIES OF MEASUREMENTS[3]

Distance from the Facade of the Concrete Shield *The distance from the centre of the core vessel to the facade is 222 cm*	First Measurement *Dosimeters inserted for 60 min.*		Second Measurement *Dosimeters inserted for 1 418 min.*		Third Measurement *Dosimeters inserted for 246 min.*	
[cm]	**Dose [mSv]**	**Dose rate [mSv/h]**	**Dose [mSv]**	**Dose rate [mSv/h]**	**Dose [mSv]**	**Dose rate [mSv/h]**
198	35	35	58	2.4		
200						
202	51	51	130	5.5	11.1	2.7
204					16.6	4.1
206	96	96	256	10.8	32.2	7.9
208					38.0	9.3
210	130	130	345	14.6		
212					36.3	8.9
214	138	138	422	17.8	41.2	10.0
216	143	143	450	19.0	39.9	9.7
218						
220	182	182	580	24.6	55.0	13.4
222	196/195	196/195	581	24.6	46.9	11.4
224	174	174	644	27.3	47.6	11.6
226						
228	147	147	873	36.9	47.9	11.7
230	193	193	1061	44.9	40.3	9.8
232					34.7	8.5
234	204	204	1727	73.1	31.9	7.8
236					39.0	9.5
238	198	198	3211	135.9	31.5	7.7
240					17.8	4.3
242	207	207	5963	252.3		
244					7.3	1.8
246	176	176	4445	188.1	6.9	1.7

[3] Note that the dosimeters have been irradiated for different periods of time in the three series. The uncertainty for each measurement is 10-15 %.

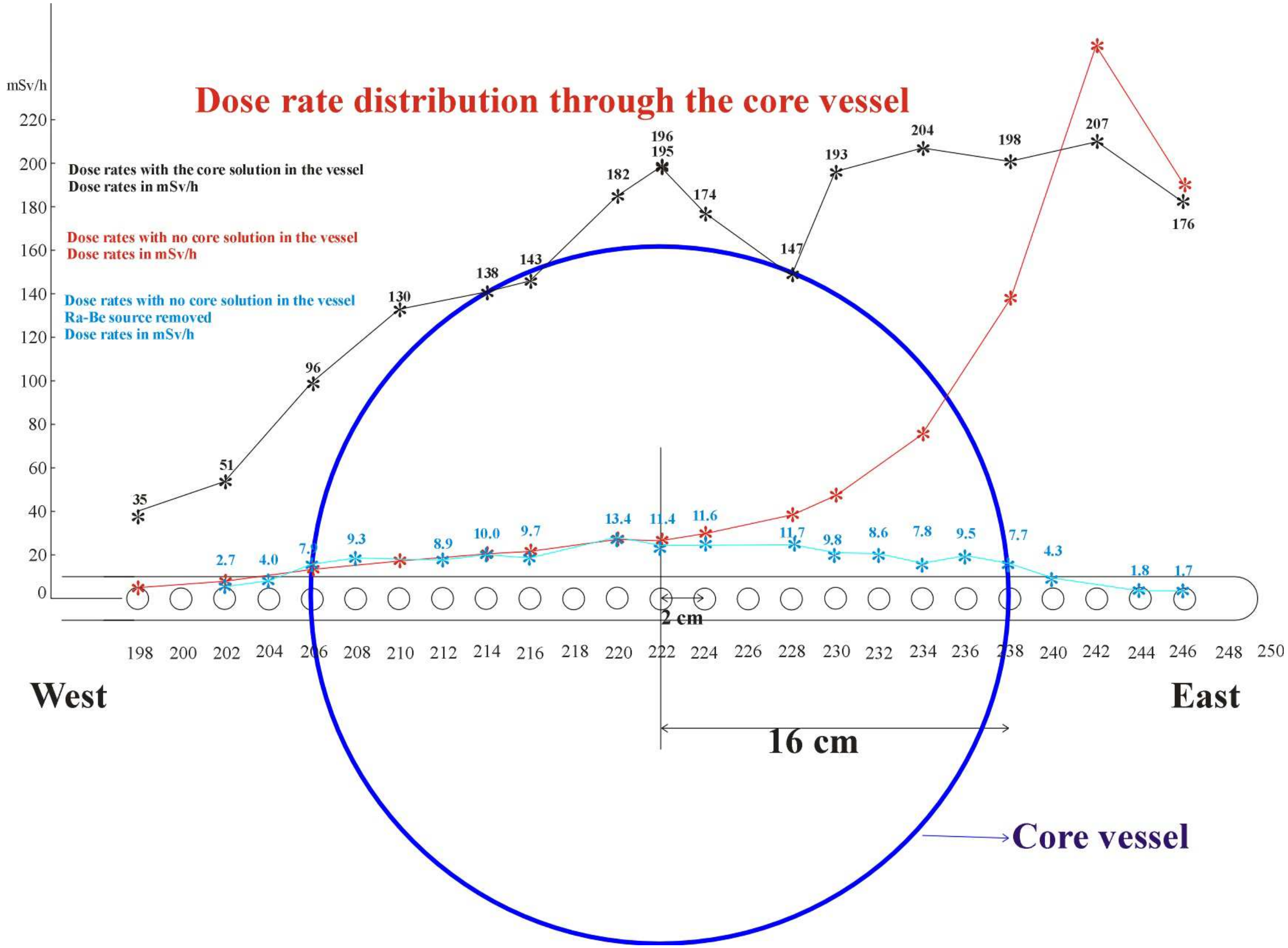


FIG. 38. Dose rate distribution. The blue circle illustrates the core vessel. The Ra-Be source was placed east of the core vessel.

APPENDIX V: DETAILS ON THE TRAPPING OF THE FUEL SOLUTION

The reactor fuel was tapped off according to detailed operational procedures of the facility. Prior to this a safety assessment was performed to ensure that the radiological safety of the staff involved in the tapping. The fuel solution was collected in four criticality safe stainless steel bottles (length - 120 cm and internal diameter - 7.3 cm). The steel bottles were placed in lead flasks providing shielding (thickness of shielding - 11.7 cm and weight of one flask – 1 300 kg). The lead flasks were supplied with leak detectors for the unlikely case that a steel bottle would leak. Prior to the actual tapping a dummy test was carried out with distilled water in order to test the procedure and train the operators. Figure 39 illustrates some of the activities involved in the tapping.

FIG. 39. Tapping of the fuel solution in preparation of the decommissioning project.

After tapping the lead flasks were transferred to a separate, safeguarded storage facility that is located on the research centre site.

APPENDIX VI: DETAILS ON THE DETERMINATION OF THE CONSEQUENCES OF NORMAL OPERATION FOR THE PUBLIC

V.1 DETAILS ON THE MODELS USED

V.1.1 Dispersion modeling

For the modelling of atmospheric dispersion Gaussian straight line plume model modified with downward sloping was used. In this model air pollution concentration at grid point (x, y, z) is calculated as:

$$C(x,y,z) = \frac{Q}{2\pi\sigma_y\sigma_z u}\exp\left(-\frac{1}{2}\frac{y^2}{\sigma_y^2}\right) \left\{\exp\left[-\frac{1}{2}\frac{(z-H)^2}{\sigma_z^2}\right] + \exp\left[-\frac{1}{2}\frac{(z+H)^2}{\sigma_z^2}\right]\right\} \quad (15)$$

Where:

$C(x,y,z)$	is the air pollution concentration at grid point (x, y, z);
Q	is the source strength;
H	is the effective height of source emission (in this test case effective stack height = physical stack height = 25 m);
σ_y, σ_z	are diffusion coefficients in y and z directions; and
u	is the average wind speed.

To account for gravitational settling of heavy gases and aerosols fixed height of emission, H is modified with the term:

$$\left(H - \frac{v_s x}{u}\right) \quad (16)$$

where v_s is thermal velocity and x is the downwind distance. It is assumed that thermal velocity v_s is equal to deposition velocity $v_d = 1000 m/day$.

VI.1.2 Dose calculations RESRAD code

RESRAD is a computer code developed at the Argonne National Laboratory for the U.S [17]. Department of Energy to calculate site-specific RESidual RADioactive material guidelines as well as radiation dose and excess lifetime cancer risk to a chronically exposed on-site resident.

A soil guideline is defined as the radionuclide concentration in soil that is acceptable if the site is to be used without radiological restrictions. Soil is defined as unconsolidated earth material, including rubble and debris that might be present.

The following two principles were applied in the Research Reactor Test Case:

(a) The annual radiation dose received by a member of the critical population group from the residual radioactive material - predicted by a realistic but reasonably conservative analysis and calculated as committed effective dose equivalent - must not exceed 1 mSv/y; and

(b) Doses must be kept as low as reasonably achievable, a concept commonly known as ALARA.

Nine environmental pathways are considered for the planned research reactor decommissioning: direct exposure, inhalation of particulates and radon, and ingestion of plant foods, meat, milk, aquatic foods, water, and soil (see Figure 40).

The code is user friendly, incorporating internal interactive help files and information on input and output data. The main menu and its submenus allow the user to easily change titles, select pathways, access and modify input data, run the programme, change screen colors and view text or graphic output.

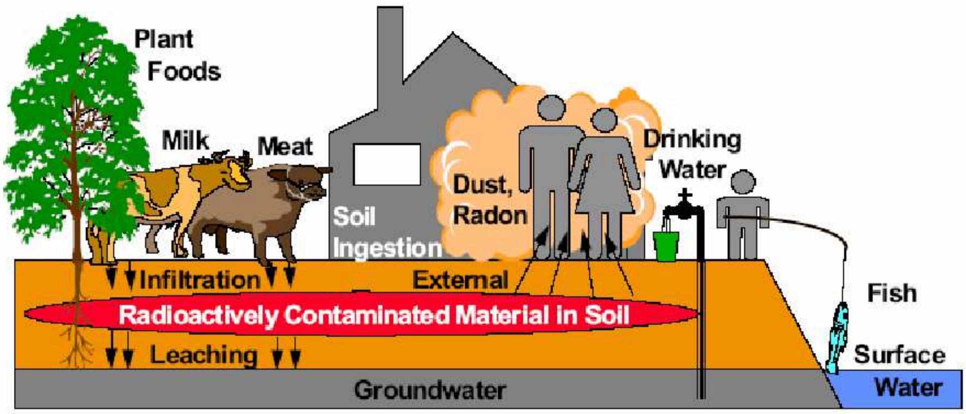


FIG 40. Environmental pathways considered in the RESRAD code (Illustration taken from RESRAD manual [17]).

Default values are provided for most of the parameters used by the code. Different exposure scenarios can be specified by adding or suppressing pathways and by modifying usage and occupancy parameters.

Four output reports are generated following each run, providing a listing of all input parameters, the maximum dose and the minimum soil guidelines. For each user-specified time, the reports list doses by radionuclide and pathway, soil guidelines, radionuclide intakes, health risks, and media concentration.

The user can plot soil guidelines, doses, or media concentrations as a function of time (see following figures). In preparing the plots, the user may specify individual radionuclides or their sum, and contributions from individual pathways or all pathways summed.

A number of popular printers, plotters, and graphic file formats are supported by RESRAD. A sensitivity analysis feature (see figure above right) allows the user to investigate the effect of input parameter variability on the calculated output.

RESRAD code uses a pathway analysis method in which the relation between radionuclide concentrations in soil and the dose to a member of a critical population group is expressed as a pathway sum, which is the sum of products of "pathway factors". Pathway factors correspond to pathway segments connecting compartments in the environment between which radionuclides can be transported or radiation emitted. Radiation doses, health risks, soil guidelines and media concentrations are calculated over user-specified time intervals. The source is adjusted over time to account for radioactive decay and ingrowth, leaching, erosion, and mixing. RESRAD code uses a one-dimensional groundwater model that accounts for differential transport of parent and daughter radionuclides with different distribution coefficients.

VI.2 ANALYSIS DETAILS

VI.2.1 Simulation of the atmospheric dispersion for determining ground surface concentrations in air and ground depositions

Initially it was planned to simulate atmospheric dispersion of four main radionuclides from the research reactor's inventory separately, each one for three stability categories (A, D, F). This approach was chosen although it was expected that maximal concentrations of the radionuclides in the zone of interest (nearest houses 500 m east of the point of discharge) will appear for neutral stability category. This assumption was proved by the initial results obtained for ^{137}Cs. That was the reason to reduce the amount of analyses and for other radionuclides only simulation for neutral stability category was performed as the most critical case.

In Table 31 results for ground surface concentrations in air and for the ground dry deposition for four main radionuclides of interests are shown. Due to some changes in the input data (increased height of the research reactor exhaust stack height, 15 m used in the preliminary analysis, now 25 m) contaminated area is increased and the point of maximal concentrations in air and the ground depositions is moved along the wind direction (now 800 m east from the point of discharge). As the radioactive inventory to be discharged is fixed while the area affected is greater, maximal concentrations are slightly lower than in those resulting from the preliminary analysis as described in Section 4.2.3.

Further analysis focused on the point 800 m east from the point of discharge considering that this point belongs to an area populated with members of the public which use local drinking water, plants and locally produced food. The model applied for the dose calculation used radionuclide concentrations obtained for the point of maximal deposition. This approach is conservative and demonstration of compliance with the dose limits for such a model allowed reaching conclusion that there is compliance for all other points (including the nearest houses 500 m east from the research reactor exhaust stack).

TABLE 31. SIMULATION RESULTS OF THE ATMOSPHERIC DISPERSION FOR STABLE WEATHER CONDITIONS

Radionuclide	Ground Surface Concentration in Air [Bq/m^3]		Dry Deposition [Bq/m^2]	
Distance from the stack	500 m	800 m	500 m	800 m
^{137}Cs	4.23 x 10^{-4}	8.69 x 10^{-4}	154.2	317.1
^{90}Sr	1.33 x 10^{-6}	2.74 x 10^{-6}	0.49	1.00
^{60}Co	2.67 x 10^{-5}	5.48 x 10^{-5}	9.74	20.04
^{154}Eu	1.33 x 10^{-5}	2.74 x 10^{-5}	4.87	10.02

Distribution of the ground surface concentrations in air for the four main radionuclides of interest is presented on Figure 41 to Figure 45. An area of 5 km x 5 km is shown where the point of discharge is located at the point (0, 0), while the nearest houses (500 m east of the research reactor) are at the point (0, 0.5). The concentrations at this point are marked on the figures, as well as the maximal concentrations observed at the point (0, 0.8) 800 m east from the exhaust stack.

Figure 46 to Figure 50 show the ground dry deposition distributions over the affected area.

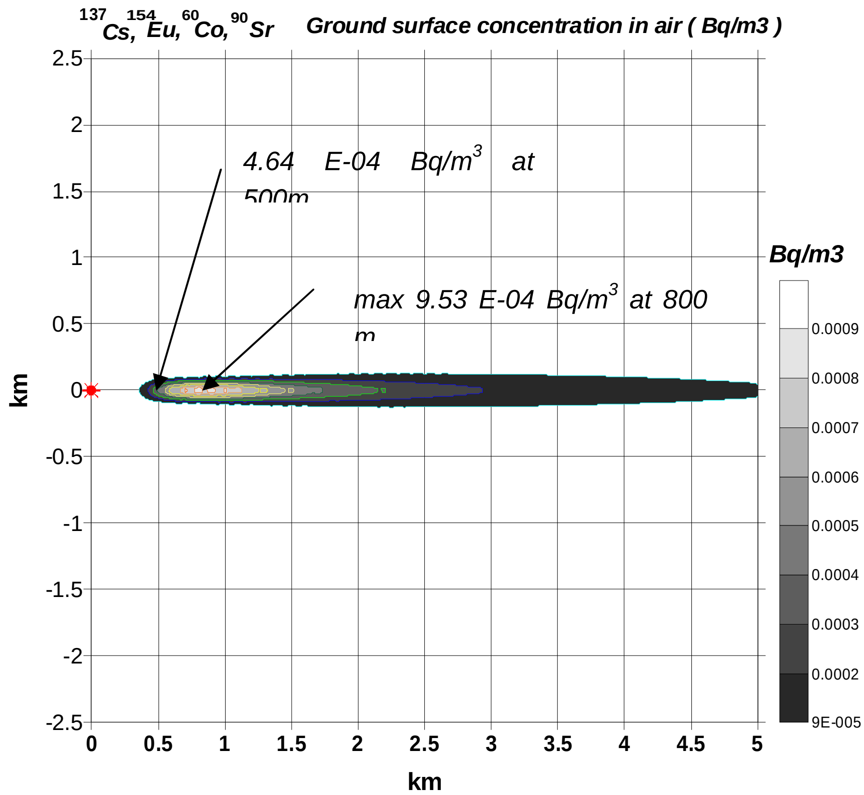


FIG. 41. Summed ground surface concentration in air (all radionuclides).

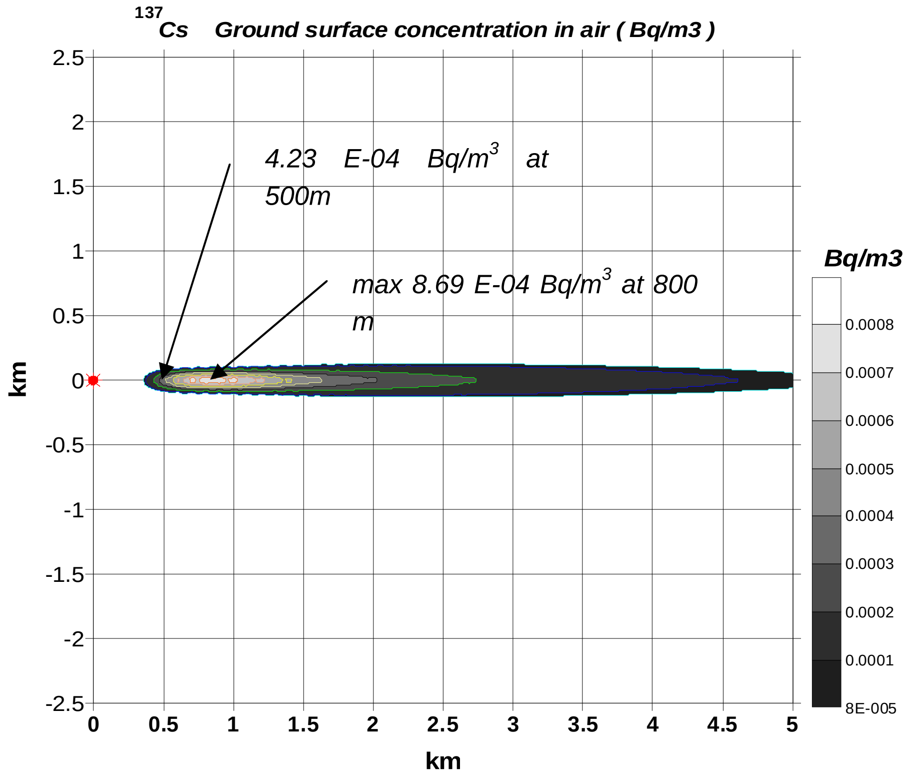


FIG. 42. *Ground surface concentration in air (^{137}Cs).*

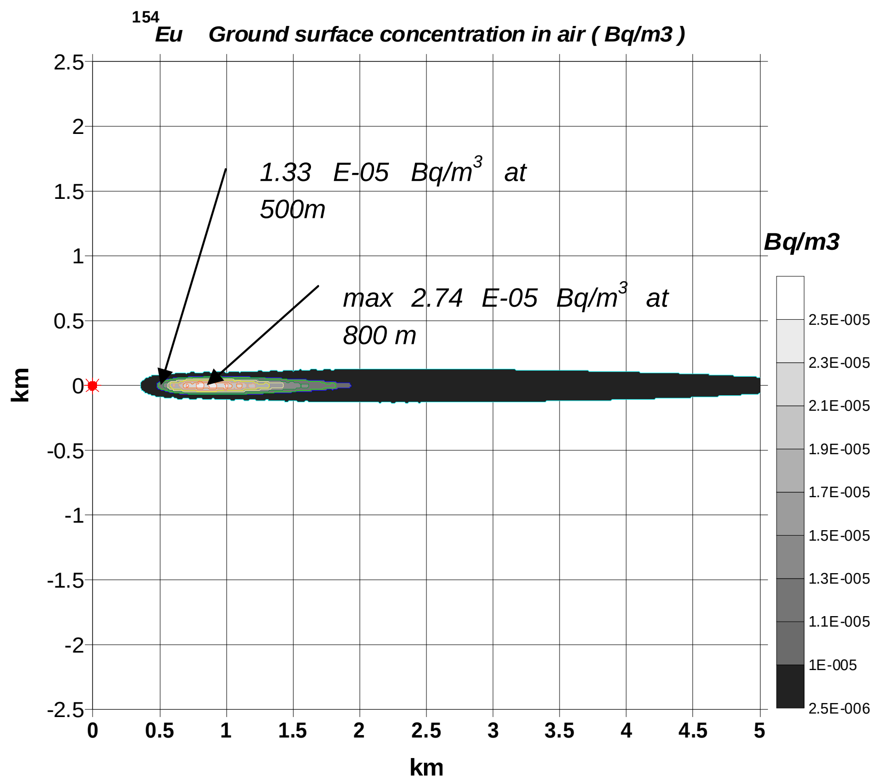


FIG. 43. *Ground surface concentration in air (^{154}Eu).*

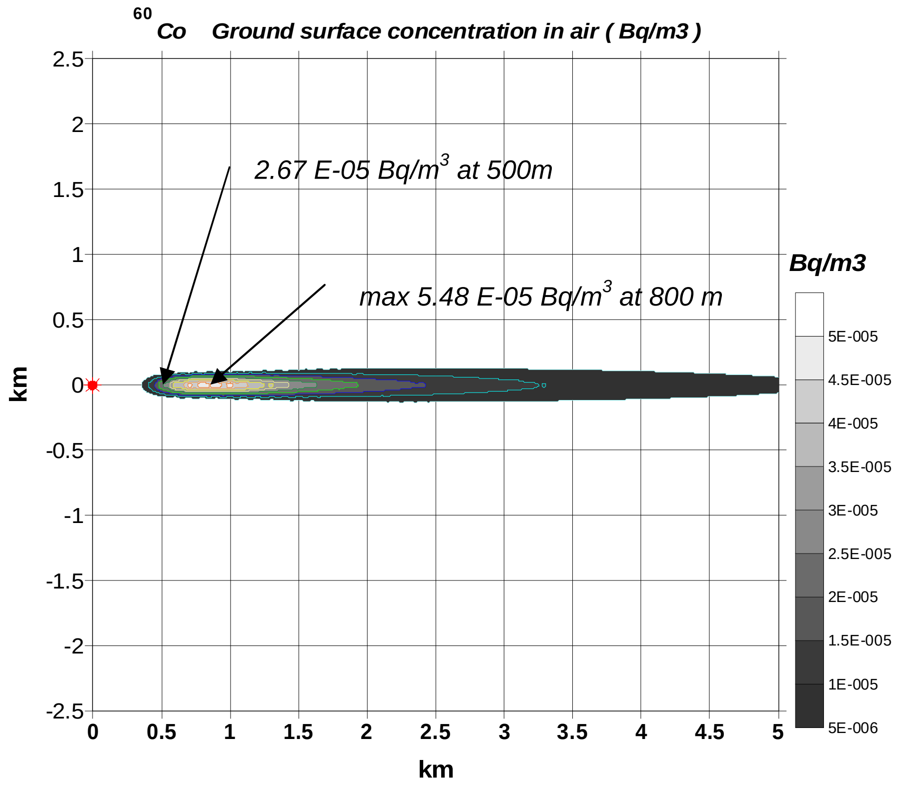


FIG. 44. Ground surface concentration in air (^{60}Co).

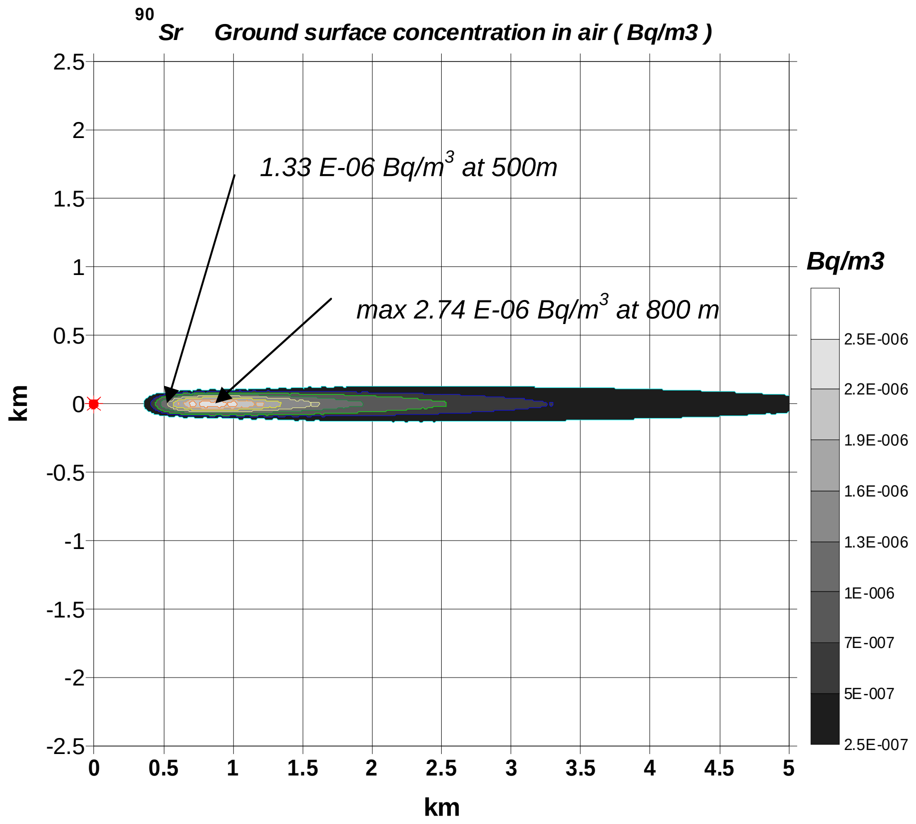


FIG. 45. Ground surface concentration in air (^{90}Sr).

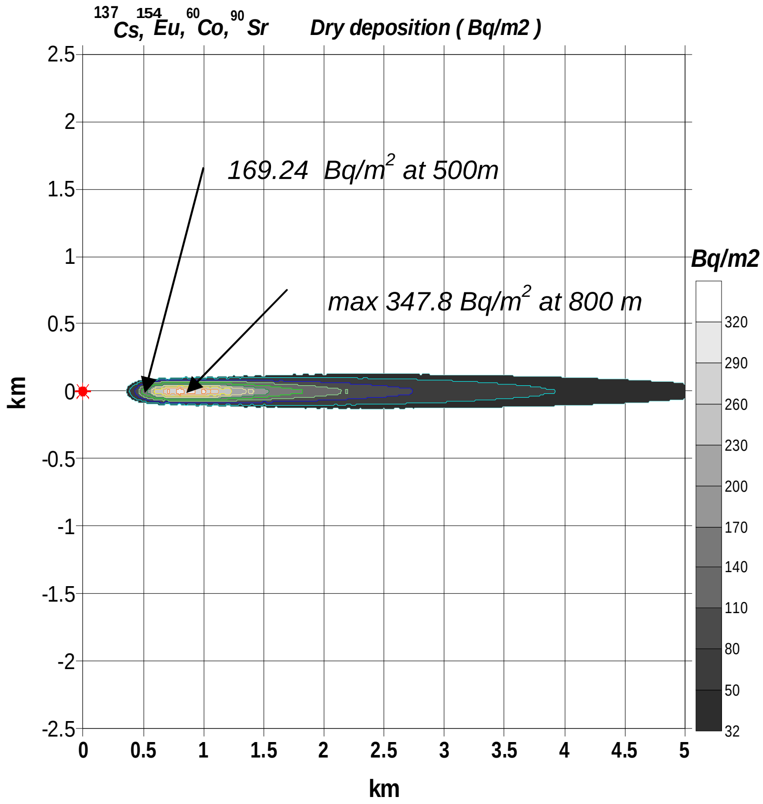


FIG. 46. *Summed ground dry deposition (all radionuclides).*

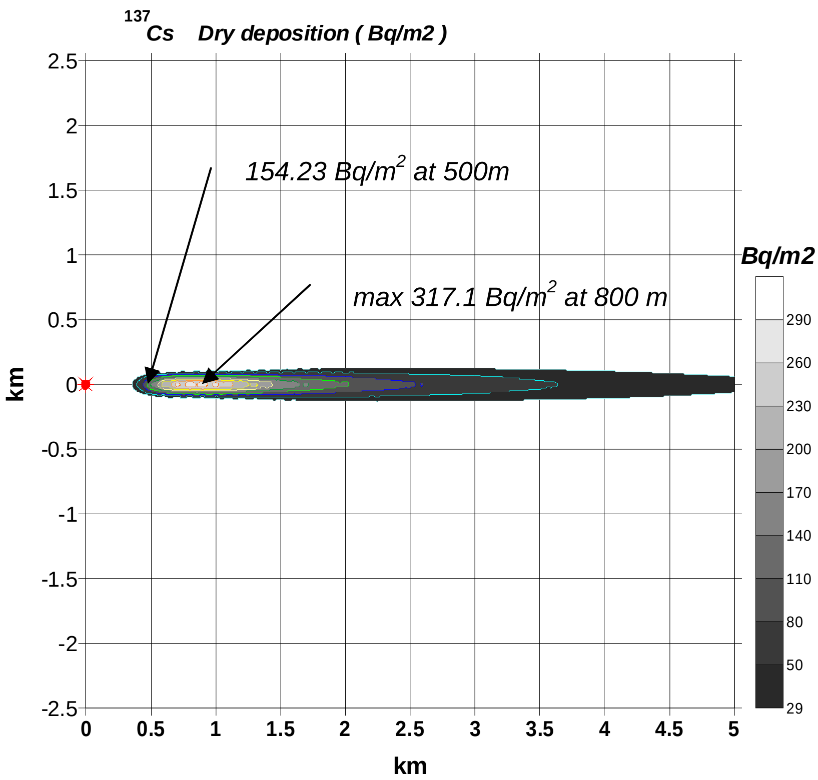


FIG. 47. Ground dry deposition (^{137}Cs).

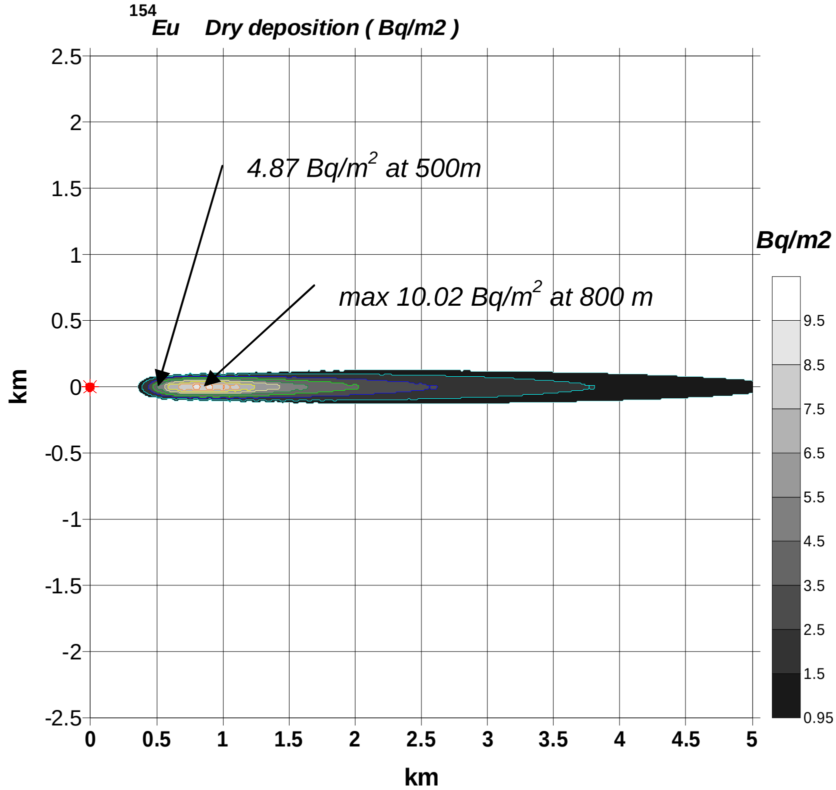


FIG. 48. Ground dry deposition (^{154}Eu).

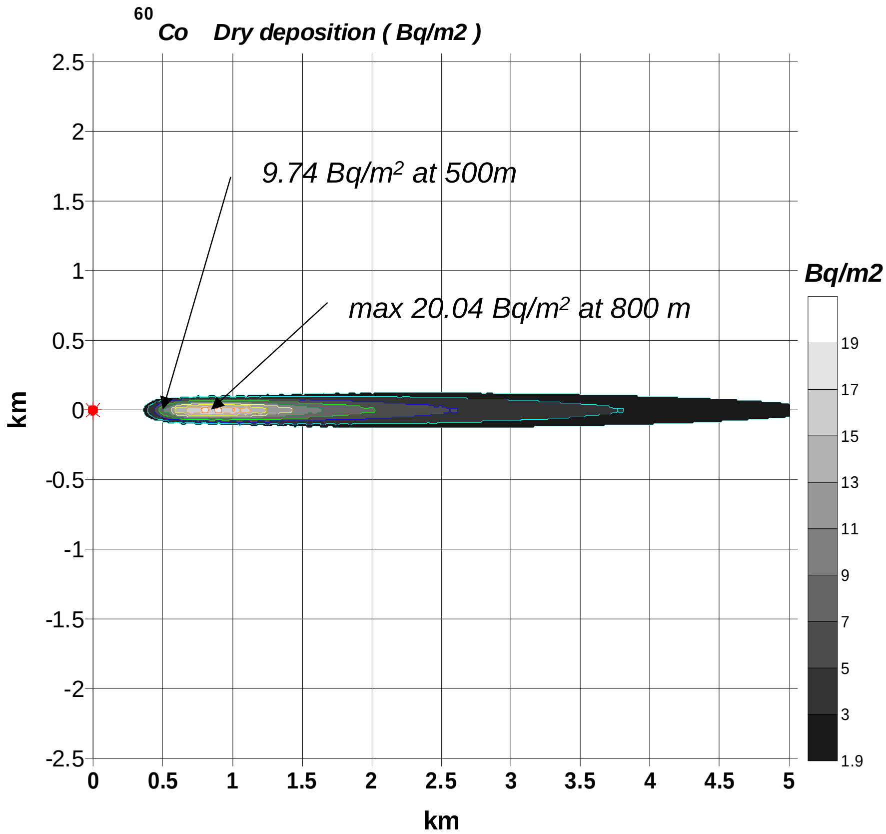


FIG. 49. Ground dry deposition (^{60}Co).

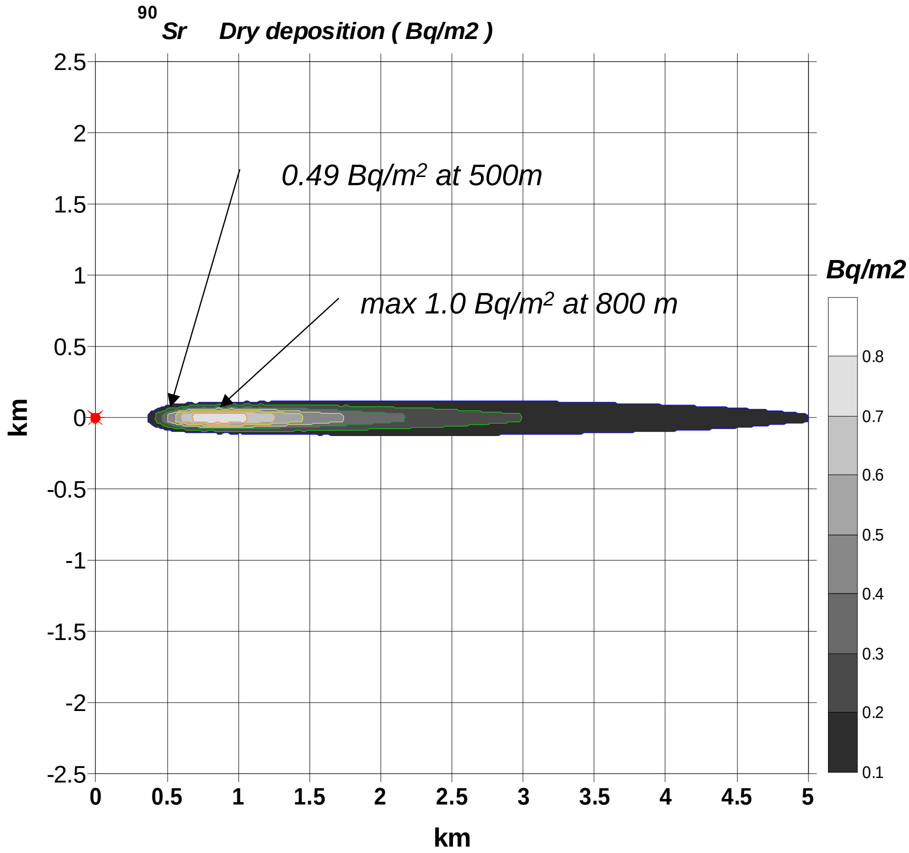


FIG. 50. Ground dry deposition (^{90}Sr).

VI.2.2 Calculation of exposures using RESRAD code

The potential exposure of members of the public was calculated using the RESRAD code and the following parameters and assumptions:

(a) Ground deposition distributions (see Figure 46 to Figure 50);

(b) Values presented in Table 32 (calculated in the step No. 1);

(c) Total radioactive material discharged during one year of normal operation (source term determination in 4.3.2);

(d) Typical values for the soil density (1.5 g/cm^3); and

(e) Area of 50 000 m^2 with uniformly contaminated surface layer of 4 cm containing mass specific soil activities (see Table 33).

Such an assumption (see (e) above) is conservative as the proposed contaminated zone for consideration contains deposited activity twice greater than the total radioactive material discharged from the research reactor.

TABLE 32 MASS SPECIFIC ACTIVITIES IN THE SURFACE LAYER OF THE CONTAMINATED AREA

Radionuclide	Mass specific activity [Bq/g]
^{137}Cs	2.114×10^{-2}
^{90}Sr	6.667×10^{-5}
^{60}Co	1.336×10^{-3}
^{154}Eu	6.680×10^{-4}

The following pathways of exposure were considered in the calculation of the doses: external gamma, inhalation (w/o radon), plant ingestion, meat ingestion, milk ingestion, aquatic foods, drinking water and soil ingestion.

As no specific data were fixed as input parameters, default RESRAD code values were used in the calculations for all the pathways considered (hydrology data, consumption of milk, vegetables, fruits, meat, fish, drinking water intake, ingestion of soil, etc.). Also, default library with the dose conversion factors based on FGR 11, food transfer factors and slope factors was used.

Annual doses for members of the public were calculated for 10 time points within the 40 years period. Figure 51 shows a RESRAD generated plot of the total doses during the years at the point of maximal ground deposition of radionuclides, while summed dose from all radionuclides and an exponential fit are shown in Figure 52. Maximal annual dose for the first year after the deposition of the activity does not exceed 7 µSv/y. Based on an exponential fit of the obtained curve and its integration over 40 years period, total dose of (105 ± 5) µSv was calculated. It is evident from Figure 53 that among the different pathways maximal contribution to the total dose to a member of the public is coming from the external exposure. This is common characteristic for all the gamma emitting radionuclides (^{137}Cs, ^{60}Co and ^{154}Eu) while for beta emitting ^{90}Sr plant, meat and milk ingestions are dominant pathways (see Figure 54).

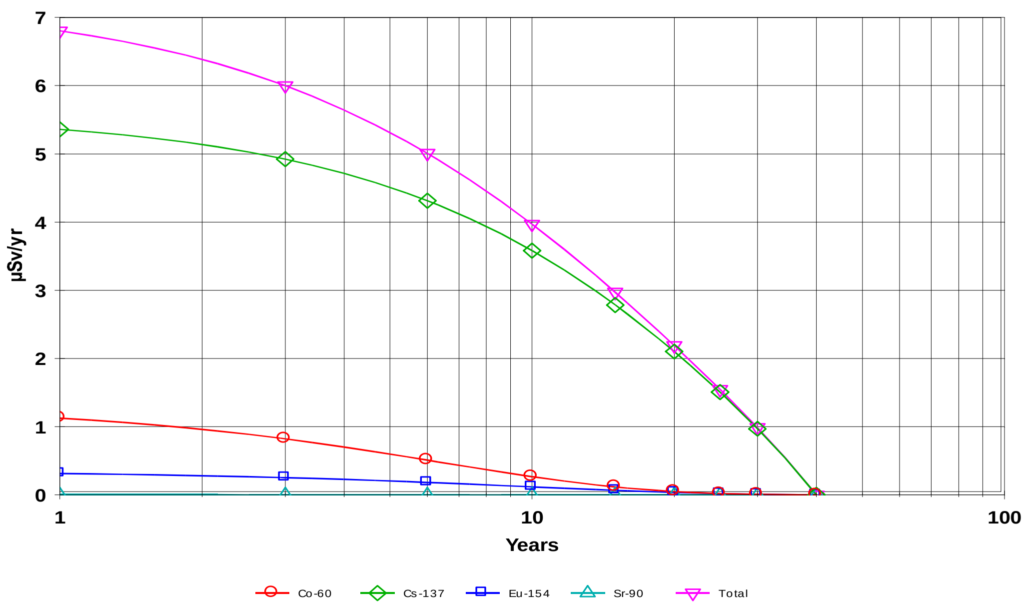


FIG. 51. RESRAD results: Summed and radionuclide specific annual doses for the public, all pathways summed.

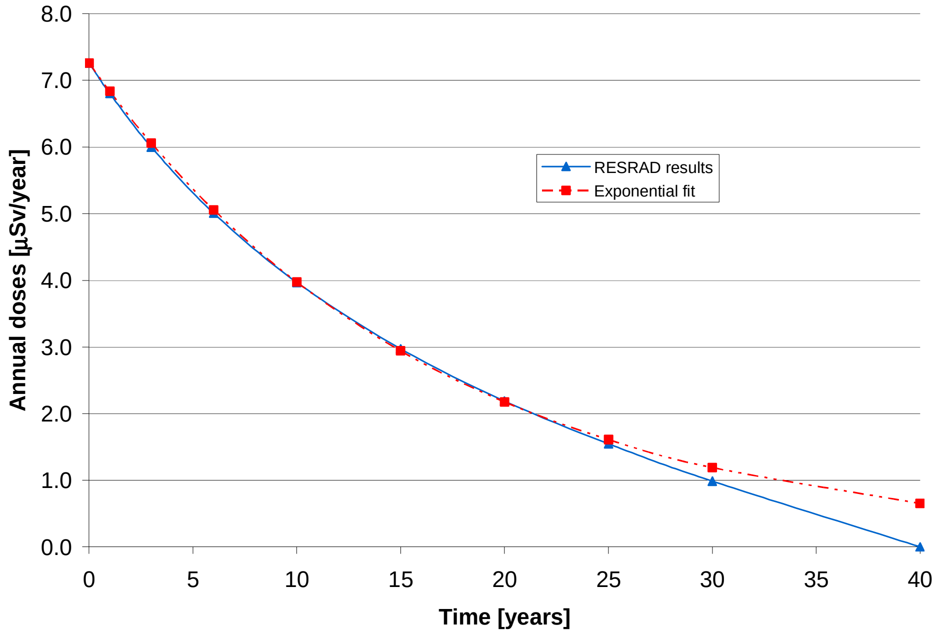


FIG. 52. Annual doses for public within 40 years period after the discharge of radionuclides into the environment at the point of maximal ground dry deposition (800 m from the source).

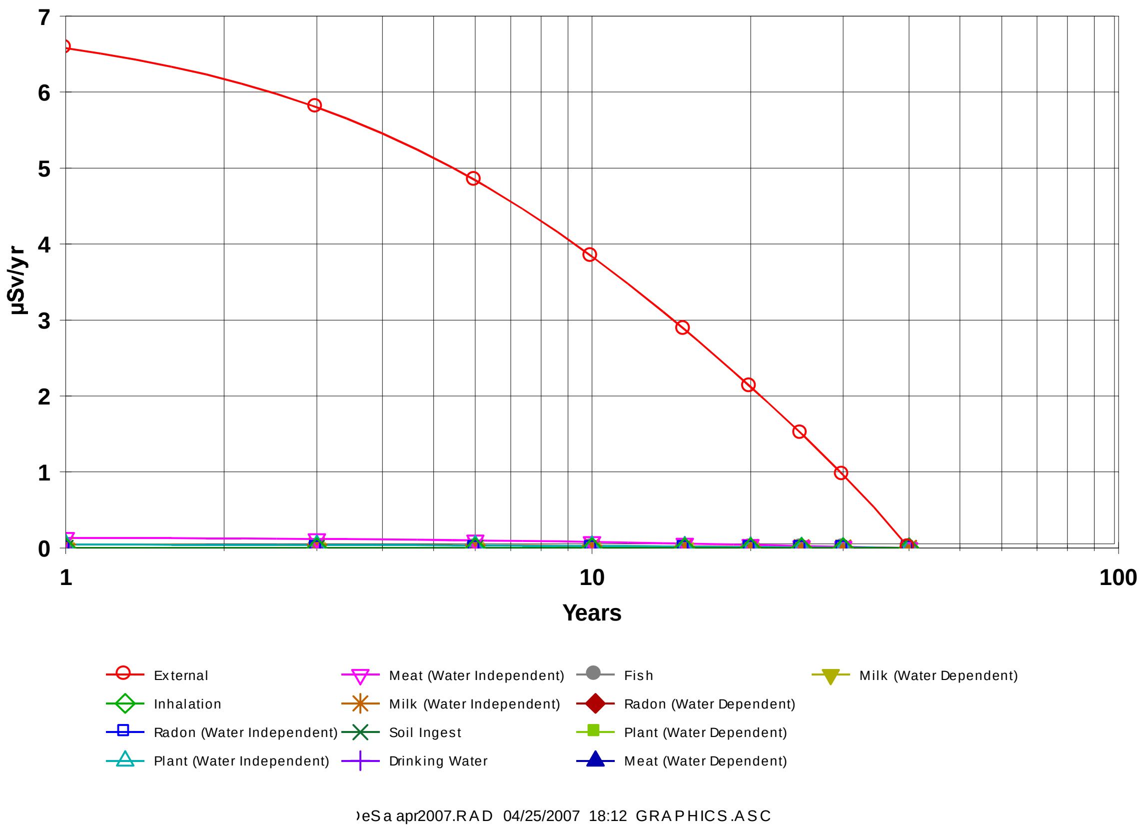


FIG.53. RESRAD results: contribution of the pathways for all radionuclides.

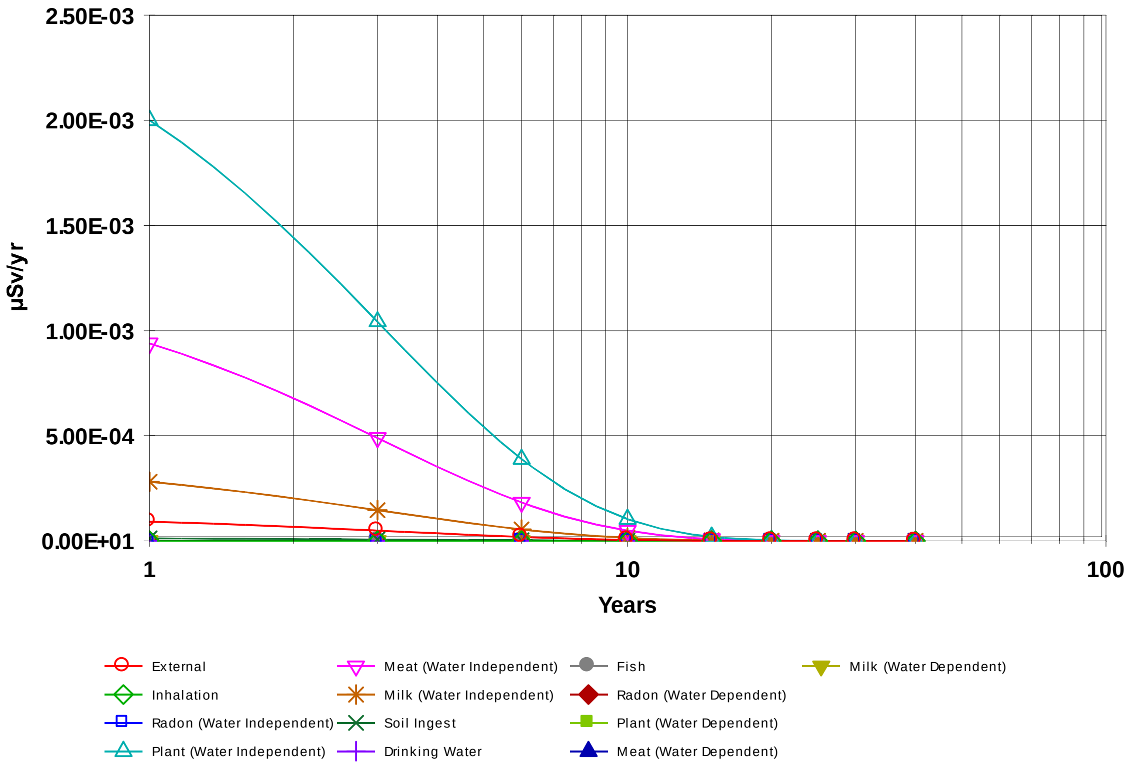


FIG.54. RESRAD results: contribution of the pathways for Sr-90.

VI.2.3 Contribution of the radionuclide ground surface concentration in air

Constant ground surface concentrations in air (saturation) for all considered radionuclides are reached soon after the beginning of the gaseous discharge. This is due to:

— A continuous discharge of radionuclides in the environment during one year;

— The stable weather conditions with constant wind direction and speed (assumptions used in the model); and

— Balance between the emission from the stack and the ground deposition.

This source will be in place until the end of the period of discharge when will disappear very fast by the ground deposition. The presence of radionuclides in the air will dominantly contribute to the doses for the members of the public via inhalation pathway.

An adult member of the public is analysed because for this age group product of breathing rate and the dose coefficient is almost equal to the maximal one (age group 12-17 years, see Table 33) while it represents maximal number of residents.

TABLE 33. DOSE COEFFICIENTS AND THE BREATHING RATES APPLIED FOR DIFFERENT AGE GROUPS

ICRP Age Group	**Dose Coefficients* [Sv/Bq]**	**Default Breathing Rate [m^3 d^{-1}] as from [18]**	**Product DC*DBR [Sv m^3 Bq^{-1} d^{-1}]**
3 months (0-1 y)	3.60×10^{-8}	2.86	1030×10^{-7}
1 year (1-2 y)	2.90×10^{-8}	5.16	1496×10^{-7}
5 year (2-7 y)	1.80×10^{-8}	8.72	1570×10^{-7}
10 year (7-12 y)	1.30×10^{-8}	15.3	1989×10^{-7}
15 year (12-17 y)	1.10×10^{-8}	20.1	2211×10^{-7}
Adult (> 17 y)	9.70×10^{-9}	22.2	2153×10^{-7}

*) Reference [4]

The doses from one year inhalation were calculated (see Table 34) by using:

- The ground surface concentrations in air (see Table 34) for the point (0, 0.8);
- The breathing rate for an adult of 22.2 m^3/day (Table 33);
- The coefficients for committed effective dose per unit intake via inhalation for members of the public for moderate absorption; and
- Age group over 17 years.

TABLE 34. DOSES FROM INHALATION CAUSED BY THE PRESENCE OF RADIONUCLIDES IN AIR

Isotope	**Ground Surface Concentration in Air [Bq/m^3]**	**Dose Coefficients* [Sv/Bq]**	**Dose [Sv]**
Cs-137	8690×10^{-4}	9700×10^{-9}	6835×10^{-8}
Eu-154	2740×10^{-5}	5300×10^{-8}	1178×10^{-8}
Co-60	5480×10^{-5}	1000×10^{-8}	4443×10^{-9}
Sr-90	2740×10^{-6}	3600×10^{-8}	7998×10^{-10}
Total			$\mathbf{8537 \times 10^{-8}}$

*) Reference [4]

From the results obtained it is evident that the contribution of the radionuclides present in the air (still not deposited) to the total dose can be neglected as the main pathway (inhalation) for that source will result in 0.1 mSv of total dose for one year exposure period.

APPENDIX VII: EXPECTED DURATION OF INDIVIDUAL TASK

See file: Annex I Part B Appendix VII.pdf

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