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Decommissioning of the nuclear facilities at Risø National Laboratory

Descriptions and cost assessment

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Abstract The report is the result of a project initiated by Risø National Laboratory in June 2000 on request from the Minister of Research and Information Technology. It describes the nuclear facilities at Risø National Laboratory to be decommissioned and gives an assessment of the work to be done and the costs incurred. Three decommissioning scenarios were considered with decay times of 10, 25 and 40 years for the DR 3 reactor. The assessments conclude, however, that there will not be much to gain by allowing for the longer decay periods; some operations still will need to be performed remotely. Furthermore, the report describes some of the legal and licensing framework for the decommissioning and gives an assessment of the amounts of radioactive waste to be transferred to a Danish repository.

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Contents

1 Introduction

The study reported here was initiated by Risø National Laboratory in June 2000 on request of the Minister of Research and Information Technology. The purpose of the study was to produce a survey of the technical and economical aspects of decommissioning the nuclear facilities at Risø National Laboratory. The survey should comprise the entire process from termination of operation to the establishment of a "green field", and give an estimate of the manpower and economical resources necessary as well as an estimate of the amounts of radioactive waste that must be disposed of. The design and costs of the establishment of a repository for radioactive waste, however, was not part of the project.

The study comprises:
- Description of the facilities in question
- Detailed description of the decommissioning work to be done at each facility
- Detailed assessment of the necessary manpower
- Assessment of the need for special equipment for various operations
- Assessment of the radiation doses to be expected
- Assessment of other safety related circumstances
- Assessment of the amounts of radioactive waste produced and its treatment
- Assessment of non-nuclear environmental aspects of the decommissioning
- Assessment of licensing aspects
- Assessment of the costs of the single activities
- Assessment of the total costs of the decommissioning of all nuclear facilities at Risø for each of the scenarios considered
- Assessment of the possibilities for reuse of buildings and non-radioactive equipment.

During the project it has become evident that for many of the decommissioning tasks the extent of the work and the costs can only be assessed with considerable uncertainty at this stage. The present report, therefore, should be taken only as the first step in a process where the estimates are improved continually as experience grows.

When the project started it was anticipated that the closure of reactor DR 3 and thereby the onset of the decommissioning work was some years into the future. However, in September 2000 Risø's management decided that the reactor should not be restarted after a period of repair and inspection. This decision meant that the planning for decommissioning all nuclear facilities at Risø had to start right away. Therefore, the question of which organisational structure should be handling the decommissioning had to be solved at that time. Originally it had been a task for the project to advise on that question. But most of the other work in the project continued as planned.

The project was carried out by a small project group of four persons with contributions from a number of other Risø staff who have written separate chapters. Furthermore, consultant assistance has been obtained from the Danish consulting company DEMEX with respect to the assessment of the expenses for demolition work. In addition the experience and advice of colleagues in other countries has been obtained through visits to similar laboratories in Germany, Switzerland and the UK.

The project work was followed by a steering committee with participation of the Risø management and relevant experts from Risø. Members of the steering committee were:
Jørgen Kjems, Managing Director, Risø National Laboratory
Lisbeth Grønberg, LL.M., Company Secretary
Knud Brodersen, MSc, Head of the Waste Treatment Plant
Mogens Bagger Hansen, MSc, Head of the Department of Nuclear Facilities
Per Hedemann Jensen, BSc, Head of the Section of Applied Health Physics
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Decommissioning of Risø's nuclear facilities. Descriptions and cost assessment.

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A panel of international experts has reviewed a draft of the present report. Valuable comments were given by these reviewers in their report and at a seminar held at Risø National Laboratory on 14 February 2001. In the present, final, version of the report the concrete suggestions for changes, given by the reviewers, have been adopted. Some of the comments from the reviewers, furthermore, are valuable input for the further planning of the decommissioning work and the refinement of the cost assessments. A summary of the review seminar is attached in Appendix 6 together with reports from the reviewers.

2 General aspects of decommissioning

Decommissioning can be defined as "All the administrative and technical operations allowing to withdraw a facility from the list of licensed facilities"[1]. The administrative operations concern particularly the elaboration of decommissioning plans and obtaining authorisations and free release certificates for the facilities and the site; the technical operations include among others the decontamination, the dismantling and the waste management. The decommissioning doesn't necessarily aim at destroying the buildings, but to release them from all obligations and controls corresponding to the class they belong to.

In the present report about Risø National Laboratory's nuclear installations it will be assumed that most of the buildings will actually be torn down and the area brought to a "green field". But some buildings will be cleaned and used for other purposes afterwards.

2.1 Phases of decommissioning

The IAEA has proposed a division of decommissioning into three stages. This definition, which is used in several countries, is based on the following classification:

Stage 1: Storage with surveillance
Consists of minimal decontamination, draining of Liquid systems, disconnection of operating systems, physical and administrative controls to assure limited access; and continued surveillance (monitoring and inspection) and maintenance for a predetermined time period. Before completing Stage 1 for nuclear reactors, spent fuel has to be removed from the facility. This stage does normally not change the Site license (depending on regulatory process).

Stage 2: Restricted site release
All equipment and buildings that can be easily dismantled are removed or are decontaminated and made available for other uses. Any remaining fluids are drained from the systems. At nuclear reactors, the biological shield is extended and sealed to completely enclose the reactor structure. At fuel cycle facilities, the primary radioactive plant and equipment will sometimes be removed. Surveillance around the barrier may be reduced, but it is desirable to continue periodic spot checks as well as surveillance of the environment.

Stage 3: Unrestricted site release
All buildings, equipment and materials that cannot be decontaminated below established clearance levels are removed and the resulting waste is handled and stored or deposited. The remaining parts of the plant and the site are released for unrestricted use. In some cases, the installation is totally dismantled and the site is re-established to "green field" conditions. No further surveillance is required.

2.2 Decommissioning strategies

Many factors must be considered when selecting the strategy for the decommissioning of a nuclear facility. These include the national nuclear policy, characteristics of the facility, health and safety, environmental protection, radioactive waste management, availability of staff, future use of the site, improvements of decommissioning technology that may be achieved in the future, cost and availability of funds for the project and various social considerations. The relative importance of these factors must be judged case by case.

[1] The European Commission's web site on Decommissioning of Nuclear Installations. http://www.sckcen.be/eccdecommissioning/

Three general types of strategy are mentioned at the European Commission's web site:

DECON (Decontamination). In DECON, all components and structures that are radioactive are cleaned or dismantled, packaged and shipped to a waste disposal site, or they are stored temporarily on site. Once this task is completed and the regulatory body terminates the plant's license, that portion of the site can be reused for other purposes.

SAFSTOR (Safe Storage). In SAFSTOR, the nuclear plant is kept intact and placed in protective storage for tens of years (20 to 150). This method, which involves locking that part of the plant containing radioactive materials and monitoring it with an on-site security force, uses time as a decontaminating agent. Once radioactivity has decayed to lower levels, the unit is taken apart similar to DECON.

ENTOMB. This option involves encasing radioactive structures, systems and components in a long-lived substance, such as concrete. The encased plant would be appropriately maintained, and surveillance would continue until the radioactivity decays to a level that permits termination of the plant's license. Most nuclear plants will have radionuclide concentrations exceeding the limits for unrestricted use even after 100 years. Therefore, special provisions will be needed for the extended monitoring period this option requires. To date, no facility owner has proposed the entombment option for any nuclear power plants undergoing decommissioning. This option is, in fact, similar to declaring the site as a shallow land burial site.

As a point of departure the present study assumes that the approach at Risø will be a combination of the first two types of strategy. In chapter 5 three scenarios are defined, two of which comprise some period of safe storage for two of the facilities.

2.3 Legal aspects in Denmark

The nuclear facilities at Risø were built and operated according to Law 170 of 16 May 1962 with later amendments. This means that changes in the facilities including eventual decommissioning have to be licensed by NUC, the Nuclear Office of the Agency of Emergency Preparedness (Ministry of the Interior) and by SIS, the Institute of Radiation Hygiene under the National Health Service of Denmark, (Ministry of Health).

The rules set out by the Danish Working Environment Authority concerning general safety and working conditions have to be followed. For certain types of work some environmental protection laws may be applicable and consent from the local community authorities may be required.

A formal EIA (environmental impact assessment) of the decommissioning project is not required in Denmark, although this is under consideration in some other countries. The establishment of a Danish repository for radioactive waste would, however, require an EIA; but the repository is not a topic for the present report.

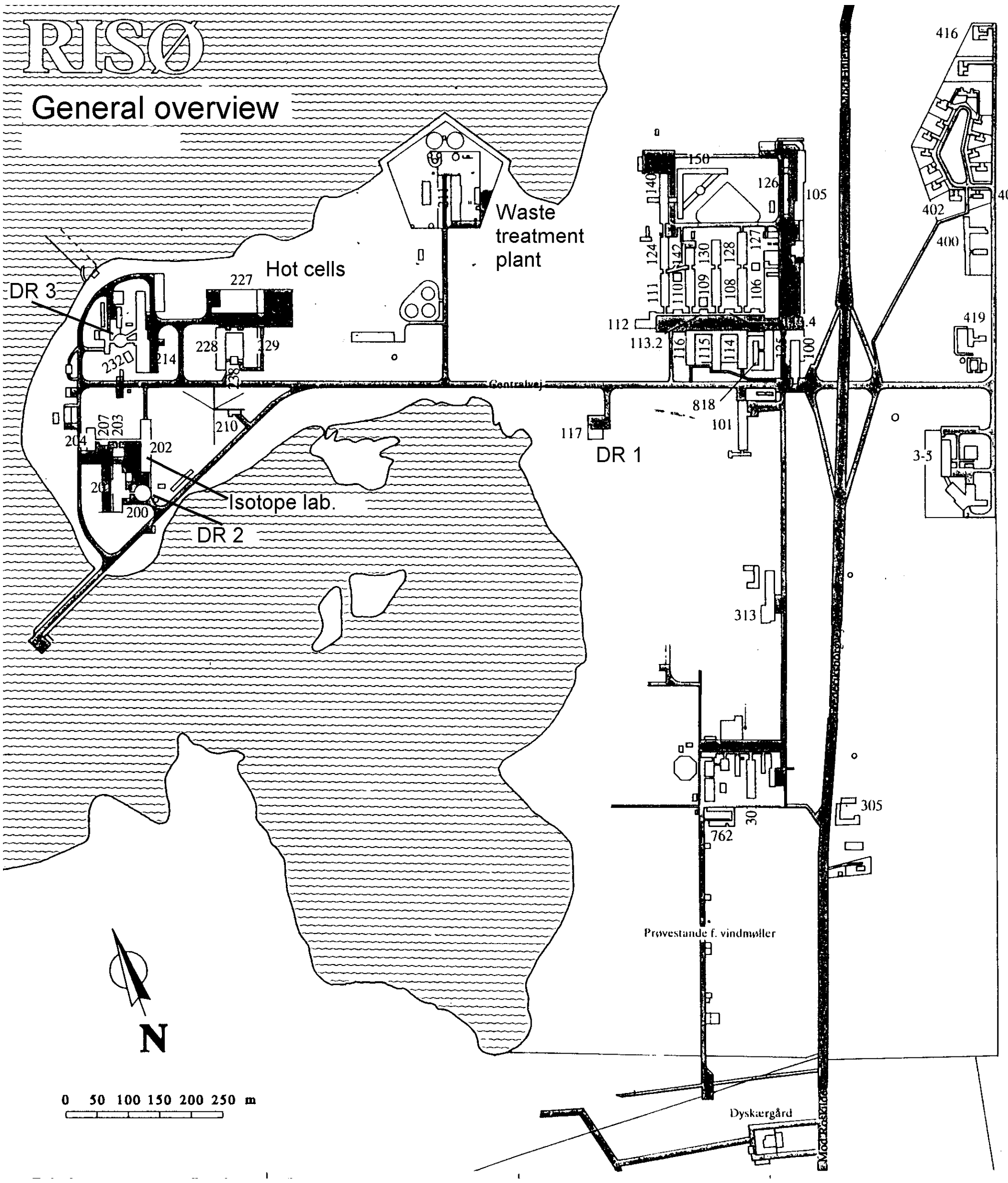
The European Union and international organisations, notably the IAEA, must be informed about the decommissioning projects.

3 Description of the nuclear facilities at Risø National Laboratory

The descriptions in this chapter serve to give an impression of the facilities to a reader who is not acquainted with them beforehand. The descriptions are not intended to be of such a detail that they themselves can form the basis for planning of the actual decommissioning. More detail in this respect can be found in chapter 5.

The descriptions of the activity inventories given below, on the other hand, are given in such a detail that they could form the basis for planning of protective measures for the decommissioning tasks. Nevertheless, surveys of radiation and contamination levels of course will have to be performed prior to and during the work.

The map below shows the locations of the nuclear facilities at the Risø area.



3.1 DR 1

3.1.1 General description

DR 1 is a 2 kW thermal homogeneous, solution-type research reactor which uses 20 % enriched uranium as fuel and light water as moderator. The American company Atomic International delivered the reactor components in 1956 and they were assembled by the Danish firm Ludvigsen & Hermann. The first criticality was obtained on August 15th, 1957. During the first 10 years of operation the reactor was heavily used for making neutron experiments, which later were transferred to the DR 2 and DR 3 reactors at Risø. Since then the reactor has mainly been used for educational purposes.

Reactor core
The reactor core consists of a spherical vessel – with an outer diameter of 31.75 cm - containing a 13.4 litres solution of uranyl sulphate in light water. The amount of ^{235}U in the solution is 984 g. The wall thickness of the vessel is 0.2 cm and it is made of corrosion resistant 347 type steel. The core is positioned at the centre of a cylindrical reflector consisting of graphite bars stacked in a steel tank, which is placed inside a shield of dense concrete, Figure 3.1.1 and Figure 3.1.2. The reactor is provided with various exposure facilities, e.g. a 1-inch diameter tube extending horizontally through the core tank. Furthermore, some of the graphite bars in the reflector can be removed, making exposure holes available. With the reactor operating at 2000 watts the neutron flux in the centre of the core is about 6.4 x 10^{10} n/cm^2/s, the corresponding flux in the exposure wholes just outside of the core is 3.5 x 10^{10} n/cm^2/s.

A stainless steel tube with an internal diameter of 0.5 cm is welded into the bottom of the core tank, see Figure 3.1.3. This tube is used for filling and draining the uranyl sulfate solution. Outside the lead shield the tube is provided with a water trap situated above the core solution level in the tank to prevent draining of the solution in case of a leak in the fuel handling system.

When the reactor is in operation hydrogen and oxygen are produced by radiolysis of the core solution. In order to recombine the produced gasses to water vapour, the reactor is provided with a tube in the top of the core vessel leading to a recombiner tank, Figure 3.1.3. The condensed water vapour returns to the core by gravity flow.

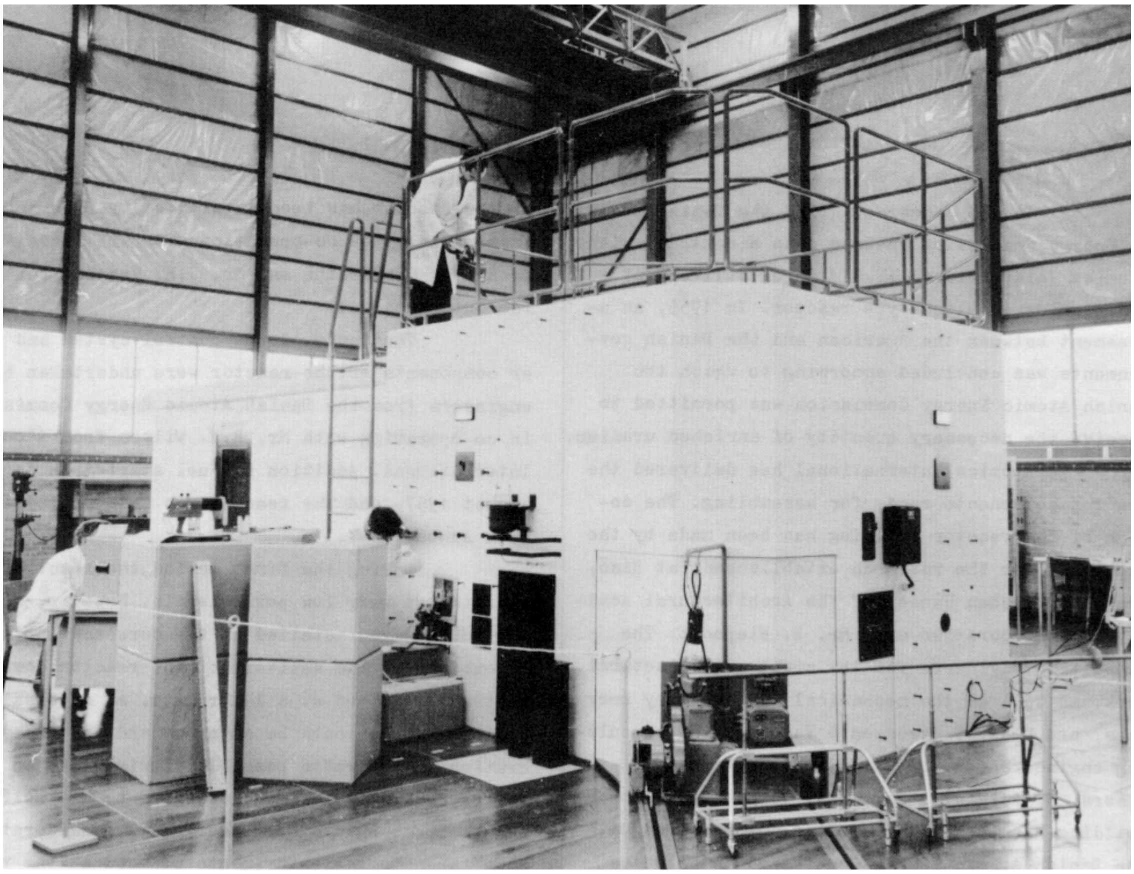


Figure 3.1.1. The DR 1 reactor.

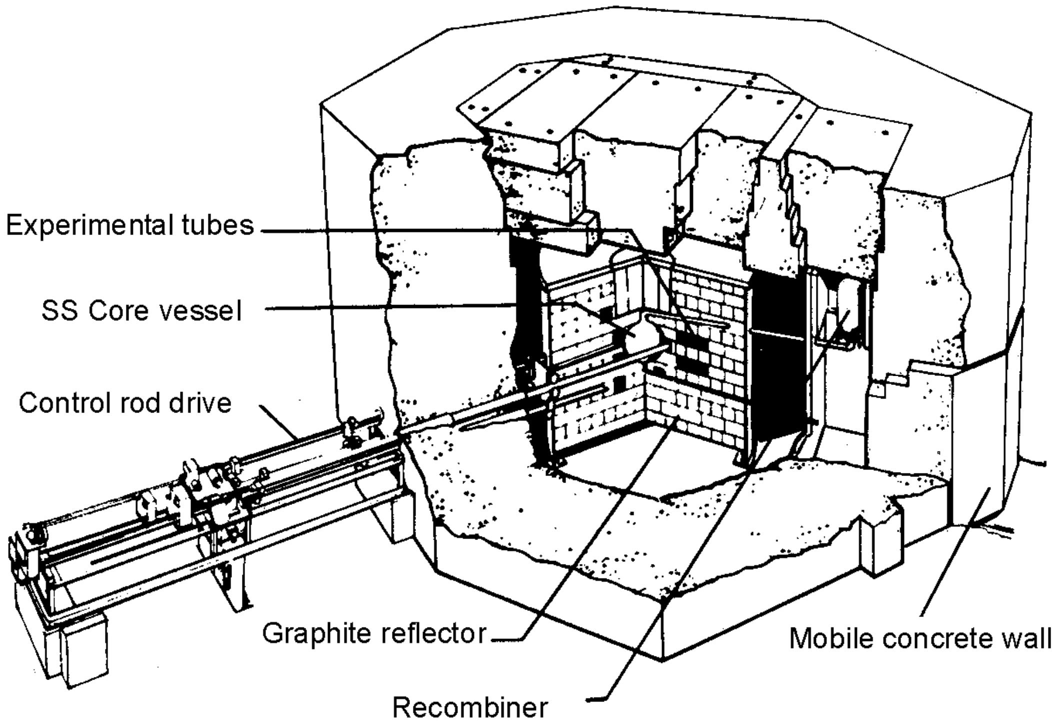


Figure 3.1.2. Cut through view of the DR 1 reactor.

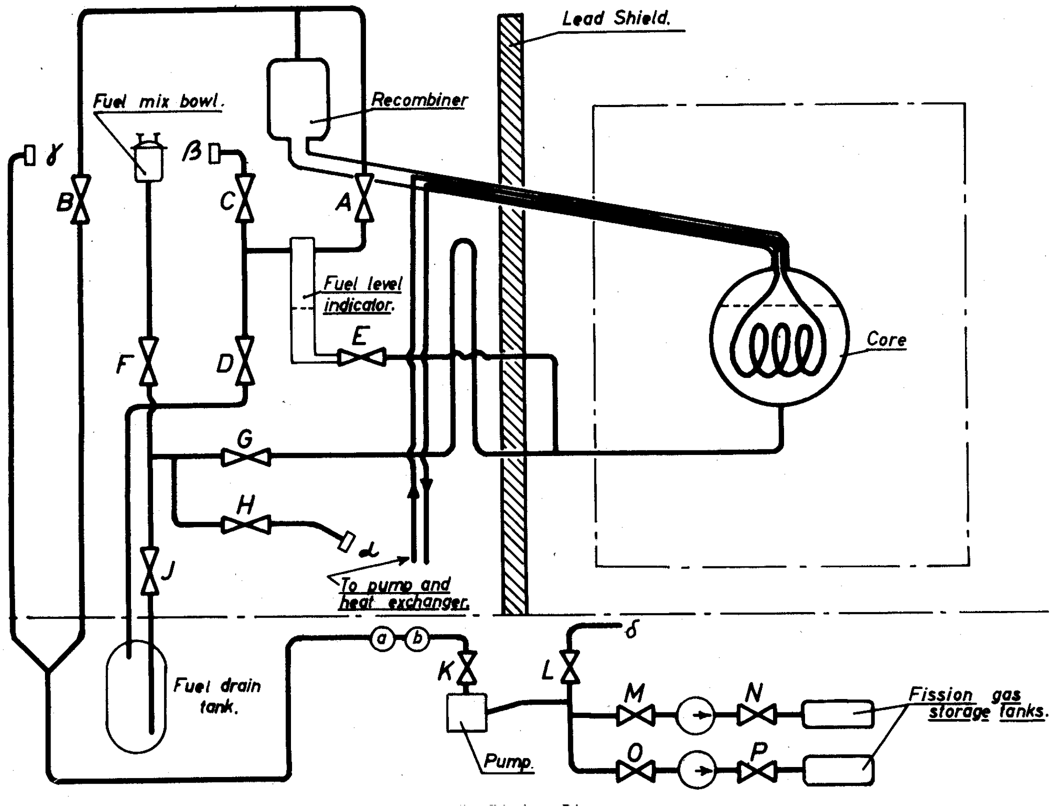


Figure 3.1.3. Piping diagram of the DR 1 reactor.

Finally there is a cooling coil made of stainless steel inside the core tank. It has 6 turns with a total length of 5 m and the outside diameter is 1 cm. The coolant lines pass through the pipe between the core and the recombiner and outside the lead shield the lines are led out and attached to the external cooling system.

Reflector
The reflector graphite bars are stacked in a cylindrical steel tank with a diameter of 152 cm and a height of 135 cm. The tank is provided with holes for the four control rods, the experimental facilities, the fuel solution fill-and-drain line and the tube to the recombiner, see Figure 3.1.3. A 0.1g Ra-Be neutron source, which is used during the start-up of the reactor, is placed in a graphite bar close to the core tank.

In case of a leak in the core tank the graphite reflector will absorb most of the outflowing fuel solution The quantity of solution which might filter down through the graphite would be caught and held by the reflector tank. It is also possible to drain the solution to the drain tank below the concrete shield and pump the fission gases to the two fission gas storage tanks, see Figure 3.1.3.

Control rods
Two safety rods and two regulating rods control the reactivity of the reactor. All rods move in horizontal slots, which are tangential to the core tank and parallel to the central exposure tube, see Figure 3.1.2. Each rod consists of a flattened stainless steel tube containing about 1 kg of boron carbide and the dimensions are 127 cm by 9.5 cm by 1.3 cm.

Shielding
The reflector tank is surrounded by an octagonal heavy-concrete shield with a wall thickness of 117 cm. The concrete contains magnetite and borocalcite and the density is 3.6 g/cm^3. Concrete shielding plugs, extending through the shield, are provided for each experimental hole, Figure 3.1.4.

There is a 9 cm thick lead gamma shield between the reflector tank and the gas recombiner vault and a 6 cm thick lead shield on top of the reflector tank. A circular part, with a diameter of 40 cm, can be removed giving access to removable graphite bars in the reflector tank. Special pure lead is used to avoid activation.
The shield above the lead shield on the reflector tank consists of some movable magnetite-concrete blocks, which are 85 cm thick. All the blocks can be removed by a 5 tons crane in the reactor hall.

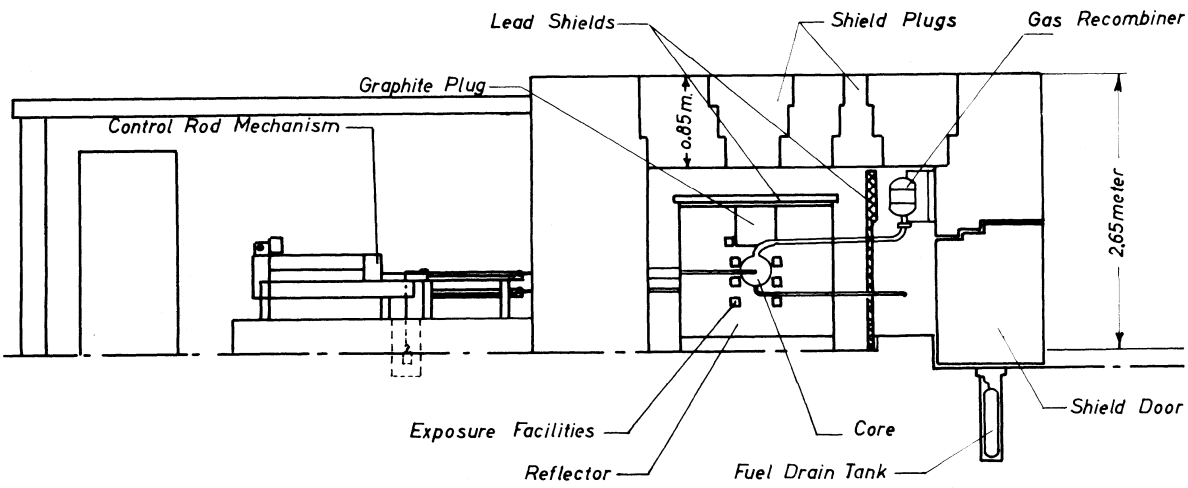
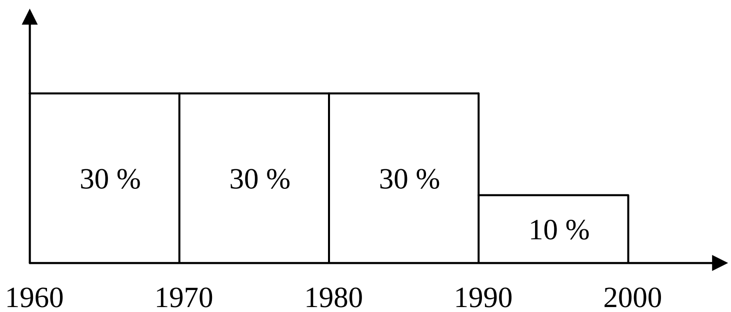


Figure 3.1.4. Vertical view of the DR 1 reactor.

3.1.2 Activity inventory

The main part of the activity is concentrated in the fuel solution. During the 40 years of operation the DR 1 reactor only has consumed about 1 g of ^{235}U of the total amount of 984 g. If the power history is assumed to be as shown below,



the total activity of the core is calculated to 200 Ci corresponding to 7.4 TBq of which 5 TBq is from fission products and 2.4 TBq from actinides. However, the fuel solution can be drained to criticality safe storage bottles and removed, thus reducing the radiation, which will ease the dismantling of the reactor components.

With the core solution removed the recombiner, together with the connecting pipes and the core tank, constitutes the most active component left due to gamma radiation from mainly ^{137}Cs deposited on the inner surfaces.

Small amounts of long-lived activation products are left in the different construction parts of the reactor, mainly in the core tank, the reflector tank and the concrete shield surrounding the graphite reflector. The assumed radionuclide activity is situated in the following parts of the reactor system.

- core tank (^{60}Co, ^{63}Ni, ^{137}Cs)
- cooling coil (^{60}Co, ^{63}Ni)
- reflector tank (^{60}Co, ^{63}Ni)
- reflector graphite (^{14}C)
- heavy concrete shield (^{133}Ba, ^{152}Eu, ^{154}Eu)
- control rods (^{60}Co, ^{63}Ni)

Since the neutron flux outside the core has been low (< 1.0 x10 ^{6} n/cm^2/s) the activation of structure materials is expected to be low, probably free-class values. Only the recombiner, the core tank and the control rods constitute components for disposal in a future waste storage.

Getting rid of the fuel solution itself may be more difficult, but it could be converted to a solid form and sent to a reprocessing plant. However, no agreement exists on this issue.

3.2 DR 2

3.2.1 General description

DR 2 was a tank-type, light-water moderated and cooled reactor with a power level of 5 MW$_t$. It was supplied by the Foster Wheeler Corporation, New York. It went critical on December 19th, 1958 and full power was for the first time achieved on August 26th, 1959. The reactor was finally closed down on October 31st, 1975 and later partially decommissioned. At present the reactor block and part of the coolant circuit remain

The Reactor Core
DR 2 was fuelled with MTR-type fuel elements. The standard fuel elements were approximately 8.0 cm by 7.6 cm in cross section and 87 cm long. Each standard element contained 18 curved fuel plates, 7.6 cm wide, 62.5 cm long and 1.5 cm thick. The fuel bearing length of the plates, i.e. the core height, was 60 cm. The plates were assembled by use of side plate and end boxes to form the complete fuel element. The fuel containing part of the fuel plates consisted of an alloy of 90% enriched uranium and aluminium with a thickness of 0.5 mm. This part was on both sides provided with a cladding layer of aluminium with a thickness of 0.5 mm. A standard element contained 186 g uranium-235.

The special fuel elements were designed to allow the insertion of the control rods into these elements, i.e. into the core. Their design was similar to that of the standard fuel elements, but they lacked the central 9 fuel plates. These were replaced by two aluminium plates with a 2.9 cm water gap between the plates. The control rods moved in this gap. The special elements contained only 93 g uranium-235.

All fuel elements were placed in holes in a 13 cm thick aluminium grid plate. The cross section of the grid plate is 83 cm by 46 cm. The grid plate is provided with 48 element positions in a 6 by 8 arrangement. Usually the 6 positions in both ends were used for reflector elements, which have roughly the same outer dimensions as the fuel elements. Initially aluminium clad graphite reflector elements were used, but they had a tendency to leak. They were, therefore, replaced with beryllium reflector elements, some of which are provided with a central channel for irradiation samples. Each element position in the grid plate is provided with a 6 mm diameter stainless steel pin to ensure correct alignment of the individual fuel elements. The grid plate, which is placed 137 cm above the bottom of the reactor tank, is fixed to the square coolant channel below the plate by stainless steel bolts.

The reactor was controlled by use of one stainless steel regulating rod and five B_4C shim-safety rods. The control rods are 5.8 cm wide and 2.2 cm thick. The absorber length of the shim-safety rods is 67 cm. The absorber length of the regulating rod is 73 cm.

The shim-safety rods were held by electromagnets, which move in stainless steel guide tubes. The guide tubes were situated on top of the special fuel elements and mounted to them by stainless steel bolts. When the reactor was scrammed, the magnets released the shim-safety rods, which then fell down into the special fuel elements by gravity. The fall of the rods was cushioned by shock absorbers attached to the top of the shim-safety rods. The magnet armature of the rods was placed on top of the shock absorber. The electromagnets and the regulating rod were, by use of long rods, connected to the control rod drive actuators, which were situated in a steel box above the top of the reactor tank. The actuators were of the rack and pinion type and could be placed above any of the 24 central fuel element positions.

The Reactor Tank

The reactor tank is made of aluminium. It has a wall thickness of 9.5 mm, a diameter of 201 cm and a height of 808 cm.

The tank is provided with a spent fuel element storage rack situated 91 cm above the top of the core. In this rack spent fuel elements could be stored to allow reduction in the activity before they were removed from the tank.

The tank is also provided with a number of penetrating beam- and irradiation tubes and a rectangular channel for the thermal column. These facilities will be discussed under Experimental Facilities.

Below the core the reactor tank is provided with 2×3 channels, which go from the reactor surface to the centre of the coolant channel below the core. These thimbles were used for the nuclear control instruments (neutron- and γ-monitors).

The top of the reactor was covered by a 6.4 mm thick steel plate, which could be removed during refuelling, inspection and maintenance. The cover plate ensured that any radioactive gases from the reactor was sucked into the air purge system, which had an intake at the top of the reactor tank.

In figures Figure 3.2.1 and Figure 3.2.2 vertical and horizontal cross sections of DR 2 are given.

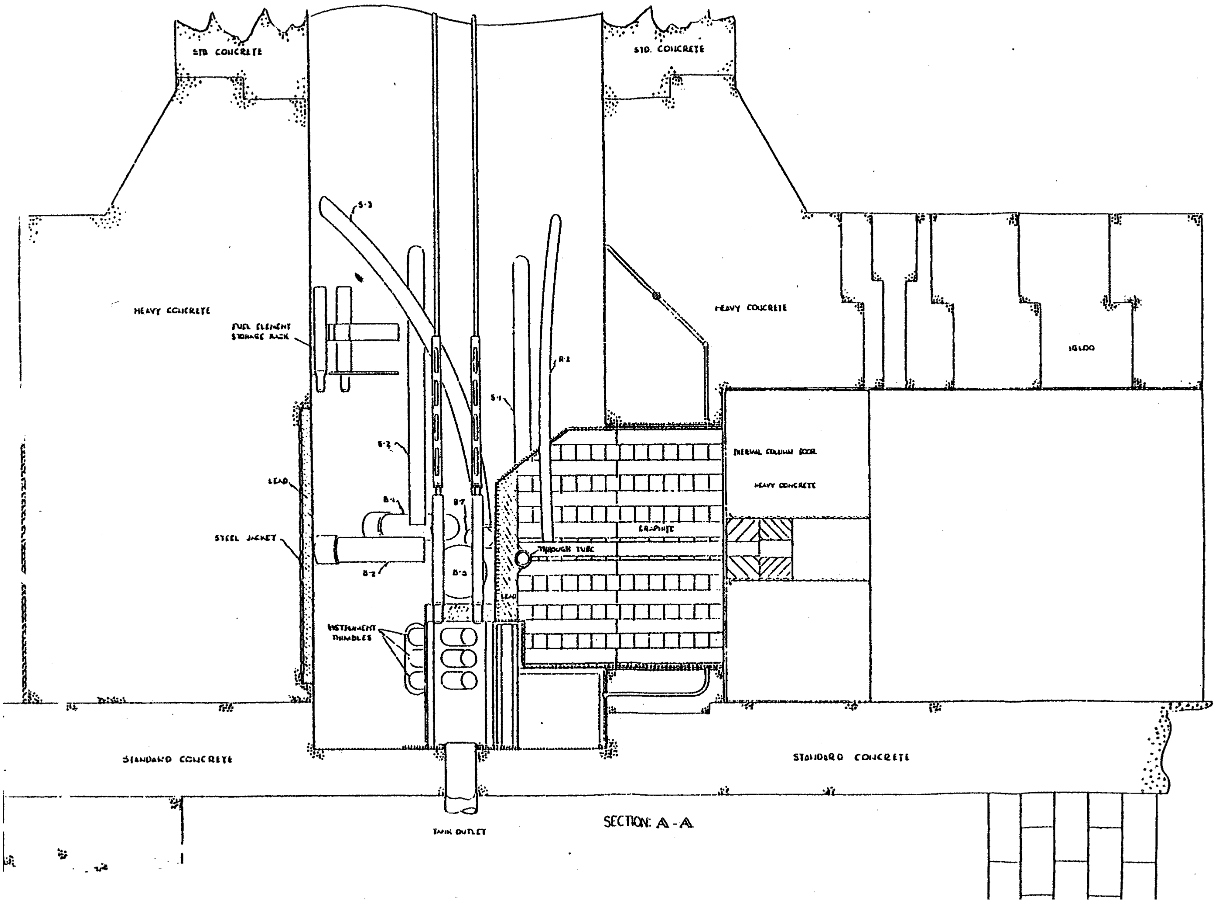


Figure 3.2.1 Vertical cross section of DR 2.

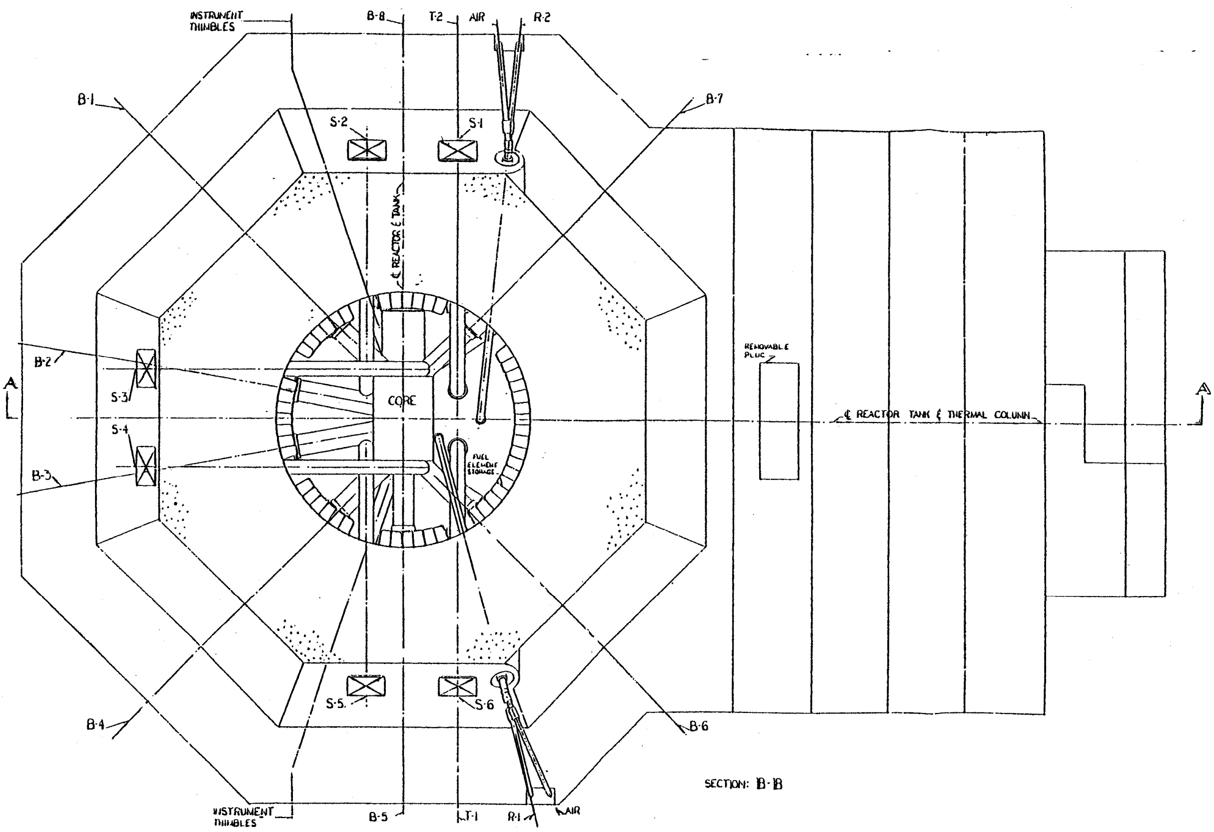


Figure 3.2.2 Horizontal cross section of DR 2.

Experimental Facilities

DR 2 is provided with a number of experimental facilities: beam tubes, a thermal column, a through tube, irradiation tubes and pneumatic tubes. These facilities are shown in Figure 3.2.1 and Figure 3.2.2.

The reactor has eight beam tubes. Five of these, B-1, B-2, B-3, B-4 and B-5, are nominal 6” aluminium tubes with a usable inside diameter of 14.6 cm. B-6 and B-7 are nominal 4” aluminium tubes with a usable diameter of 12.2 cm. The last beam tube, B-8, was nominally a 13” tube with a usable inside diameter of 32 cm. The tubes start at the outer surface of the reactor and extend all the way to the reflector elements or the fuel elements. They are removable, since they are mounted in and supported by an aluminium sleeve, which is anchored in the concrete shield. The inner part of the sleeve (from the core and until the step increase in the sleeve diameter) is made of aluminium, the outer part in the concrete shield is made of stainless steel.

Shielding plugs of similar design are inserted into all beam tubes. The plug of the 13” beam tube is provided with rollers to ease the movement of the plug. The plugs consist of two parts, a removable nose piece (water filled during operation) and a barytes concrete cylinder, which at the end closest to the core is provided with a neutron-absorbing boral layer. The concrete cylinder is encased in a chromium-plated carbon steel tube. This tube is machined so accurately that the tolerance between the steel tube and the sleeve of the beam tube is 0.5 to 1.3 mm. Each plug is provided with three spiral stainless steel conduits, which ends in a threaded hole in the nosepiece, two for water and one for experimental use.

The beam plugs described above are the plugs delivered by Foster Wheeler. Several of them have been replaced by new plugs e.g. to permit beam experiments where the beam plug has to be provided with a beam hole from the core to the experimental equipment outside the reactor.

The thermal column is situated at the east face of the core. At the core end it is 64 cm wide and 97 cm high. Along its 163 cm length its cross-section increases so that it is 132 cm wide and 157 cm high at the outer end. A 15 cm thick lead layer is situated between the core and the graphite inside the aluminium casing of the thermal column. The lead layer is bonded to the aluminium nose plate. The casing is inserted into the rectangular channel, which protrudes from the tank, and which is anchored in the concrete shield. At the outer end the casing is welded to the channel wall. Spacers between the channel and the casing permit circulation of tank water around the thermal column. The part of the aluminium casing that is inside the concrete is lined by a 6.4 mm thick boral plate to reduce the activation of the concrete.

The thermal column consists of 10.1 by 10.2 cm graphite stringers, stacked horizontally, alternatively across and along the axis of the column. The arrangement permits the removal of a central 36 by 36 cm section of the stringers. Five of these stringers may be removed independently through holes in the concrete door at the end of the column (the inner igloo door). These holes are provided with steel plugs. The thermal column was designed for a centreline thermal neutron flux of 5 10^9 n/(cm^{2}s) at a distance of 30 cm from the outer end of the column. The cadmium ratio at that point was designed to be above 1000.

Outside the thermal column the igloo, built of heavy concrete blocks, is constructed (cf. Figure 3.2.1 and Figure 3.2.2). The inner surface of the igloo is lined with boral plates. The inner igloo door is situated at the outer end of the column. It is a concrete door, mounted on rollers and movable by use of an electric motor in the door. The inner door has a thickness of 112 cm and is encased in a 1.9 cm thick steel plate. Its inner surface is lined with a boral plate. Two 61 cm thick concrete blocks to provide additional shielding close the outer end of the igloo. This outer ”door” can only be ”opened” and ”closed” by use of the crane of the reactor hall.

DR 2 is provided with a through tube (T1-T2) which runs through the thermal column, near and along the core (cf. Figure 3.2.1 and Figure 3.2.2). In the concrete shield and in the thermal column the through tube is surrounded by aluminium sleeves to allow removal of the tube. The aluminium sleeves are flanged to the carbon steel vestibule by use of rubber gaskets to allow thermal expansion and prevent leakage of tank water.

From the reactor balcony six irradiation tubes, S-1, S-2, S-3, S-4, S-5, and S-6, curve down to the reactor core through the concrete shield and the tank water (cf. Figure 3.2.1 and Figure 3.2.2). Two of these tubes terminate in the thermal column (S-1 and S-6), two along the west side of the core (S-2 and S-5), and one along the north (S-3) and the south (S-5) side. All S-tubes are of aluminium and have a diameter of 10 cm. During operation they were filled with water to improve the shielding. The S-tubes penetrations through the concrete shield are also provided with sleeves to allow the removal of the tubes.

DR 2 was provided with two pneumatic tube facilities, R-1 and R-2 (cf. Figure 3.2.1 and Figure 3.2.2), which both have a diameter of 3.8 cm. R-1 extended from the send-receive station on the northern shield surface and down into the graphite of the thermal column. R-2 extended from its station on the southern shield surface into the lead layer of the thermal column. In the concrete aluminium sleeves surround the R-tubes so that they can readily be removed. R-1 could be extended to an adjoining laboratory building. The R-2 tube never came into operation and has been removed. Its aluminium sleeve was used for insertion of a tank water level meter.

Reactor Shield

The radial radiation shield of DR 2 consists from the core and outwards of three shielding layers. The first is 68 to 77 cm of tank water (neglecting the reflector elements). The second is a 6.4 cm thick and 183 cm high lead layer in an aluminium jacket surrounding the reactor tank at the level of the core and above and below it. The third layer is about 2 m of barytes concrete with a density of 3.4 g/cm^3. The barytes concrete reaches from the hall floor and 4.1 m upward to the reactor balcony. The remaining 3.6 m of the reactor block is made of ordinary concrete. The lower part of the outer surface of the shield is lined by 19 mm steel plates. The radial shield is of course perturbed by the experimental facilities such as beam tubes, S-tubes and the thermal column.

Above the core the shielding consists essentially of 6 m of water. The shielding below the core consists of the 13 cm thick grid plate, 85 cm of water and the floor of the reactor hall, which below the reactor tank is about 25 cm thick. This floor is at the same time the roof of the hold-up tank room.

Primary Circuit

During reactor operation the cooling water flowed down through the fuel elements to the hold-up tank in the basement where the ^{16}N and ^{19}O activities, produced by neutron capture in the oxygen of the coolant during passage of the core, were allowed to decay. From here the coolant flowed through two parallel primary pumps and two parallel tube-and-shell heat exchangers after which it returned to the bottom of the reactor tank (see Figure 3.2.3). All components of the primary circuit are made of aluminium except for the circulation pumps and the valves, which are made of steel. Isolation sections are inserted at appropriate points in the circuit to prevent galvanic corrosion between the iron and the aluminium.

To keep the demineralized water of the primary circuit clean, a by-pass stream was continuously withdrawn from the primary circuit and sent through an ion exchange demineralizer and filter unit. The unit contained a separate cation exchanger and a mixed-bed demineralizer. The bottom of the reactor tank was connected to a 38,000 litre dump tank in the basement. The drainage was done by gravity. The primary circuit was also coupled to two 5,500 litre liquid waste tanks.

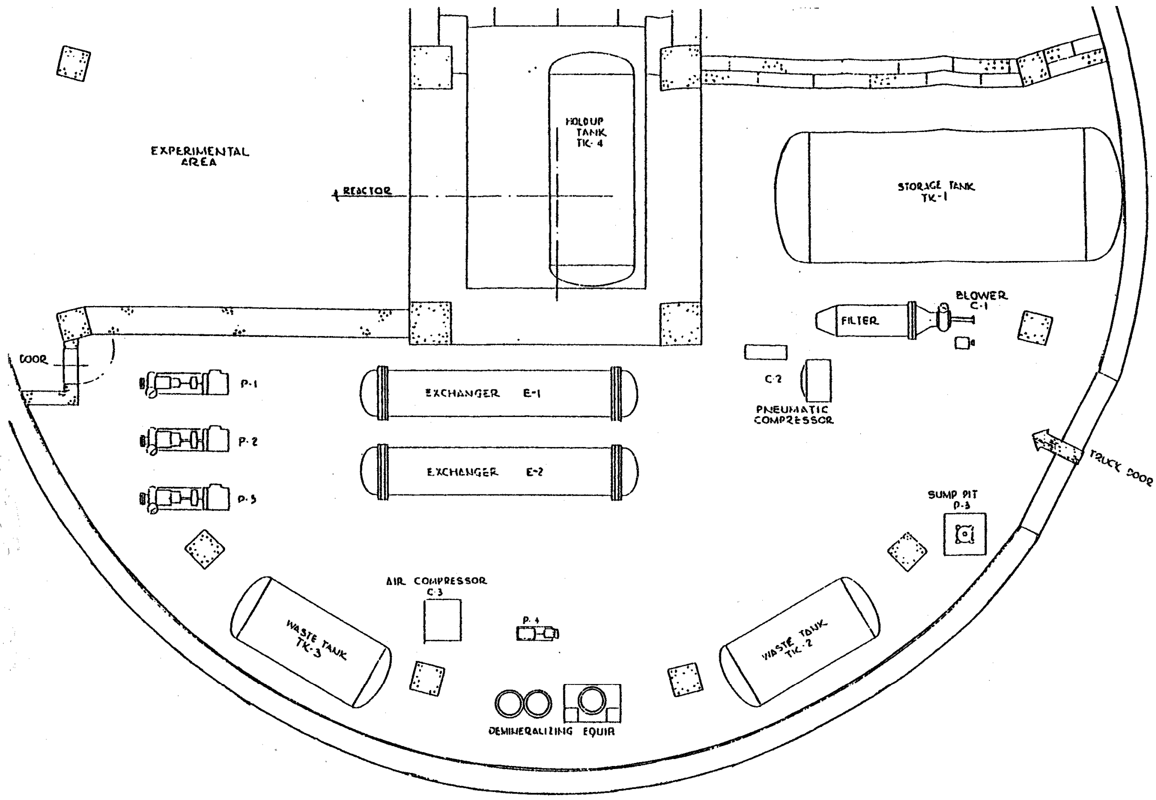


Figure 3.2.3 Arrangement of basement equipment at DR 2.

Operational History 1959-1975

As mentioned above, the DR 2 started to operate on full power, 5 MW_t, on August 26th, 1959. It operated with three shifts per day, five days per week until the end of October 1963, i.e. for 1525 days. During this period the integrated power was 2450 MW_td. From November 1st, 1963 and until the final closedown at the end of October 1975 the operation was reduced to one shift per day, five days per week. During this period of 4380 days the integrated power was 5488 MW_td.

The average thermal neutron flux in the beryllium elements has been estimated to be 3.9 10^{13} n/(cm^{2}s) and the epithermal flux to be 1.75 10^{12} n/(cm^{2}s) at full reactor power.

Close-down Period 1975-1997

The reason for closing DR 2 down at the end of October 1975 was that it seemed that the DR 3 could handle all Danish research reactor needs. However, there was some doubt whether this was correct, and for this reason it was decided that the DR 2 should be closed down in such a way that it could be re-started without too much difficulty, i.e. DR 2 was mothballed rather than finally closed down.

Immediately after the DR 2-close down, the following actions were taken. Before the end of 1975 all fuel had been transferred to a storage facility at DR 3 from where it was later sent to USA. The beryllium reflector elements, the V-tube facility etc were left in the grid plate. The guide tubes and corresponding control rods as well as plugs for the lattice plate were placed in the storage rack. The regulating rod, the magnet rods and other rods were suspended from the beams, which carried the actuator box. Shortly after the fuel had been removed, the primary circuit, which contained about 40 m^3, was drained and the reactor water transferred to the liquid waste treatment plant at Risø where it was cleaned by distillation. The secondary circuit was also drained. It was initially planned to blow dry air through the primary circuit, but this plan was abandoned.

To provide the necessary radiation shielding after the removal of the tank water, the reactor tank was provided with a new top plate of steel and a five cm thick layer of lead bricks. The beam plugs were left in the beam tubes. If they were provided with a beam hole, it was closed with an iron rod. In addition the vestibule boxes were, where needed, filled with lead bricks and sealed with tape. The six S-tubes were emptied for water and dried. Their vestibule boxes were also filled with lead and sealed with tape.

All equipment for neutron experiments in the reactor hall was removed. The GAKO valve was drained for water and blocked in the open position so that the hall is directly connected to the surroundings. The emergency exit was drained for water and left open. Various alarm systems were deactivated. The personnel air lock was de-activated so that the two doors of the lock could be opened simultaneously.

When in 1978/79 it was decided to use the reactor hall for chemical engineering experiments and to abandon the possibility of future restart of the DR 2, the box with the control rod actuators on top of the reactor tank, related cables and other equipment were removed. The thickness of the lead layer was increased to 10 cm, and a 40 cm thick reinforced concrete lid was placed above the lead layer. The concrete lid was sealed to the reactor top.

The igloo in front of the thermal column was used in a somewhat unorganised way for storage of slightly radioactive components, e.g. ion chambers, television camera, tubes used in the V-tube facility etc., with the motor driven door in its inner position. The two concrete “doors” were placed in front of the igloo opening and the igloo was sealed.

All vestibule boxes of the beam tubes were covered by steel plates, which were welded to the steel faceplate on the outer surface of the reactor shield. The vestibules of the S-tubes were also closed with steel plates, fixed to the faceplate by screws or welding. The pneumatic system was taken apart and all tubes closed.

The most if not all of the ionisation chambers of the instrument thimbles were removed and transferred to the igloo. The shielding plugs of the thimbles were left in place and sealed.

The control room panel and other electronic equipment was removed.

Much of the remaining tubing and instrumentation was removed. Free tube ends were plugged and so were a number of experimental holes in the floor. The reactor hall was provided with windows.

The basement under DR 2 is divided into three areas: The experimental area, the operational area, and the hold-up tank room. The experimental area was cleared for all equipment. Most of the equipment of the operational area was left intact. Some of the equipment was transferred to the igloo in the reactor hall, some to the Risø waste treatment plant. The area, which remains locked, has been used for storage of reactor equipment, “fishing tools”, spare parts and for some time of one of the two DR 2 archives. The hold-up tank room was used to store slightly radioactive components. This room was originally provided with a door, but it was removed and the concrete wall was extended by use on concrete blocks.

Decommissioning Study 1997-2001

The chemical engineering experiments were terminated in 1995, and after that the reactor hall was used for storage of various equipment. In 1997 it was decided to start a decommissioning study, while some of the people with experience of DR 2 were still at Risø. The project had two aims: 1) To review the status of the DR 2, in particular to obtain information of the radioactivity of the various parts of the DR 2, in order to plan its final decommissioning. 2) To use DR 2 as a research facility to open the possibility of Risø acting as consultant for decommissioning activities elsewhere.

The first step in this process was to clean in particular the reactor hall after the chemical engineering experiments, to re-establish the access to the reactor top and to provide the necessary working area at the reactor top. This task had been accomplished in March 1999. Parallel to this work a revised version of the safety documentation for the DR 2, a revised proposal for Conditions for Operation and a project plan was prepared for submission to the Danish Authorities. The documents were submitted during the summer of 1999 and approved by the Authorities in December 1999.

The practical execution of the project was planned to start in January 2000. It was delayed due to the leak that was detected in the DR 3 in December 1999 and the following events, because the personnel who were to carry out the practical work on DR 2 was DR 3 personnel.

In May 2000 the concrete lid, the lead bricks and the steel lid were removed from the top of the DR 2. Air samples were taken to test the tank air for radioactivity and beryllium dust. The radioactivity of the air was at background level, and the beryllium content was a factor of thousand below the permissible level. The visual observation of the tank interior revealed that fewer components were stored in the tank than had been expected.

In June smear tests were taken at the surface of the tank wall and of the beryllium reflector elements. The radioactivity of the samples were at background level, while the beryllium content at the “no action required” level. The igloo in front of the thermal column was opened and it turned out, that the amount of components stored here was more than expected.

In July smear tests were taken at the tank bottom and at the top of reflector. The radioactivity was at background level and the beryllium content was at the “no action required” level. A guide tube was lifted up to the top of the tank and at a distance of about 30 cm the radiation level was 30 mSv/hr. A reflector element was lifted up from the grid plate without any difficulty. An attempt was made to loosen one of the steel bolts at the top of a reflector element with a tool, but without success.

It is planned to take all loose parts out of the reactor tank, to measure their induced activity in a facility built for this purpose and to cut them into pieces, put the pieces in waste drums and ship them to the Risø waste treatment plant. The same will be done with radioactive components in the igloo. Non-radioactive components will be stored in the experimental area of the DR 2 basement. Bulky, slightly radioactive components may be decontaminated. The hold-up tank room will be opened and the radioactivity of the stored components measured. Some beam plugs will be extracted and their radioactivity measured. The radioactivity through the beam tubes and in S-tubes will be measured. The primary circuit will be opened and smear tests taken. Cored holes will be drilled through the concrete shield at the level of the reactor core and the radioactivity of the bore hole core will be measured.

It is expected that the major part of the radioactivity remaining in the reactor is situated in stainless steel components. When these components are removed to the extent possible the γ-radiation of the reactor will be significantly reduced. This will ease the decommissioning of the DR 2.

3.2.2 Activity inventory

The DR 2 reactor was started in 1959 and was finally shut down in 1975. Small amounts of long-lived activation products are left in the different construction parts of the reactor, mainly in the reactor tank and the concrete shield surrounding the tank. The assumed radionuclide activity is situated in the following parts of the reactor system.

- reactor tank (^{60}Co, ^{63}Ni)
- heavy concrete shield (^{133}Ba, ^{152}Eu, ^{154}Eu)
- beryllium reflector elements (^{10}Be)
- thermal column graphite (^{14}C)
- beam plugs (^{60}Co, ^{63}Ni)
- guide tubes and S-tubes (^{60}Co, ^{63}Ni)
- primary cooling circuit (^{60}Co, ^{137}Cs)

The major part of the total activity will be found in steel components (screws and pins), beam plugs and heavy concrete shield.

A γ-spectrum was measured at the S1-tube and ^{152}Eu were clearly identified. The result of the two-hour measurement is shown in Table 3.2.1.

Table 3.2.1 Result of γ-spectrometric measurement at the S1-tube on 4 July 1997. The only radionuclides identified were ^{60}Co and ^{152}Eu.

Photon energy [keV]	Photon yield [%]	Net count rate [counts]	Radionuclide
344	26.6	156	^{152}Eu
779	13.0	66	^{152}Eu
964	14.5	59	^{152}Eu
1086	10.2	25	^{152}Eu
1112	13.6	88	^{152}Eu
1408	20.9	52	^{152}Eu
1173	100	285	^{60}Co
1332	100	278	^{60}Co

Since 1980 the exposure rate has been measured annually at 16 different points in the reactor hall, including the six S-tubes. Over many years the decrease in exposure rate correspond to the half-life of ^{60}Co. In later years more long-lived components have contributed to the exposure rate at the S-tubes. The measured annual exposure rate has been fitted to a two-term exponential function:

$$\dot{X} = a_1 \exp(-\lambda_1 t) + a_2 \exp(-\lambda_2 t)$$

The annual measurements and the fitted function are shown in Figure 3.2.4. The fitted value of the decay constants λ_1 and λ_2 was determined to 0.13 y^{-1} and 0.047 y^{-1}, respectively. This corresponds to a half-life of 5.3 years and 14.7 years, respectively, indicating that the radionuclides dominating the exposure rate at the S-tubes are ^{60}Co and ^{152}Eu. Europium and other rare earth materials are commonly found impurities in all types of concrete.

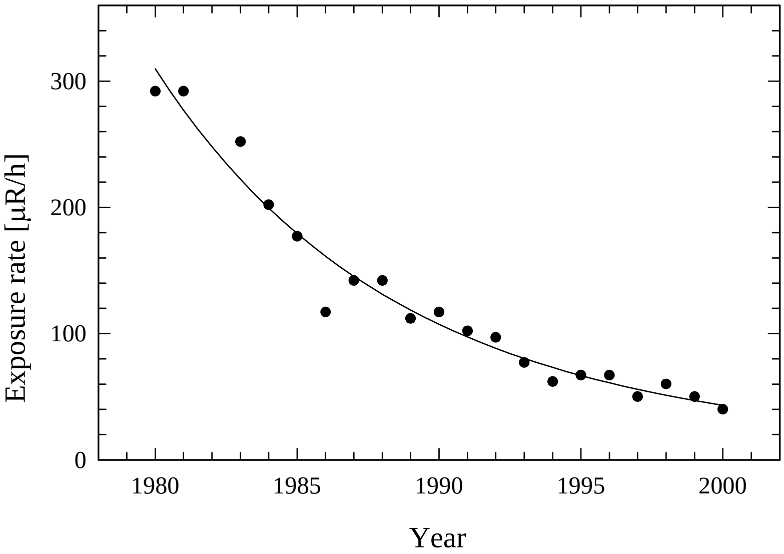


Figure 3.2.4 Measured annual exposure rate (•) at the S1-tube and the double exponential fit to the measured exposure rate.

Another fit of the measured exposure rates has been made where the decay constant were fixed at the exact values for ^{60}Co and ^{152}Eu (0.1315 y^{-1} and 0.052 y^{-1}, respectively). Figure 3.2.5 shows the fitted curve as well as separate plots of the two components.

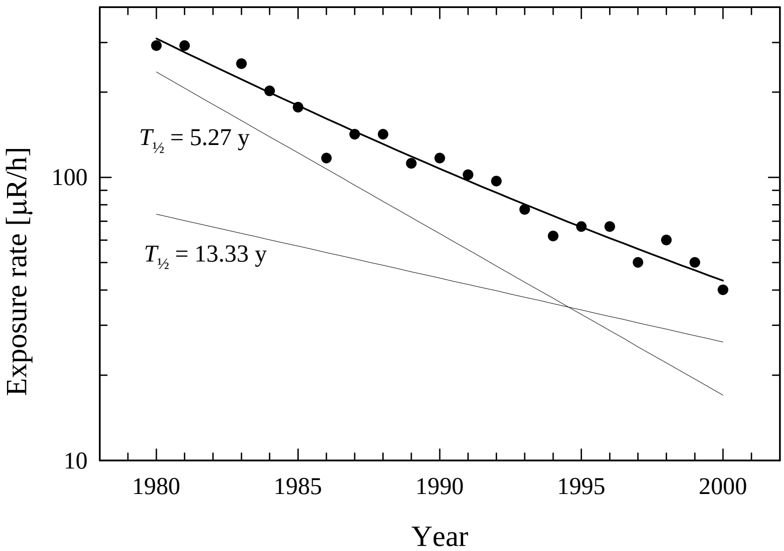


Figure 3.2.5 Measured annual exposure rate (•) at the S1-tube and the double exponential fit to the measured exposure rate. The two components contribute equally to the exposure rate around 1995. After 1995 ^{152}Eu dominates the exposure rate at the S-tubes.

After the top shield was removed the dose rate above the open reactor tank was measured to 400 $\mu Sv \cdot h^{-1}$. If this dose rate is caused by ^{60}Co alone situated in the central plane of the reactor core (distance to point of measurement about 7 metres) the activity can be calculated to about 60 GBq. If the dose rate is caused by equal amounts of ^{60}Co and ^{152}Eu the activity of each of these nuclides can be calculated to about 40 GBq. If the activity is situated closer to the measurement point these activity levels will be lower.

The stainless steel components in the reactor tank have been activated to ^{60}Co by thermal neutron capture in ^{58}Fe and impurities of ^{59}Co. Double neutron capture in ^{58}Fe ($^{58}Fe + n \rightarrow {}^{59}Fe \rightarrow {}^{59}Co \rightarrow {}^{60}Co$) will be the dominating process for production of ^{60}Co for long irradiation times compared to single neutron capture in ^{59}Co-impurities of a few ppm. Neutron activation calculations have been made for the following steel components in the DR 2 tank:

• bolts fixing guide tubes to special fuel elements	230 g SS304
• guide tubes	1,106 g SS304
• bolts fixing flange piece assembly to Be-reflector elements	480 g SS304
• shim safety rods	2,625 g SS304
• regulating rod	2,250 g SS304
• steel pins in lattice plate	300 g SS304

The activation calculations indicate that the total ^{60}Co-activity in steel components in the reactor tank is about 3.3 GBq. This is about an order of magnitude lower than the activity estimated from the dose rate measurements above the open reactor tank. At least two reasons can explain this difference. Firstly, the ^{60}Co-activity is distributed on several items in the reactor tank some of which are closer to the point of measurement than the assumed 7 metres. This will give an overestimation on the ^{60}Co-activity, as the variation of dose rate with distance is given by the inverse square law, *i.e.* if the distance is halved, the dose rate will increase four times. Secondly, the measured dose rate is a total dose rate from all radionuclides present in the reactor tank and in the surrounding concrete shield, not only from ^{60}Co.

The primary cooling system in the cellar includes heat exchangers and large pipes. Radiation measurements directly on the tubes and heat exchangers showed no increased radiation above the natural background. A γ-spectrometric analysis was made on a large smear sample taken from the inner side of a tube bending showed a contamination level of 0.1 - 6 mBq·cm^{-2} of the nuclides ^{60}Co, ^{137}Cs, ^{152}Eu, ^{154}Eu and ^{155}Eu, dominated by ^{60}Co (around 70% of the total surface contamination density). Some γ-spectrometric measurements were made in the cellar with a mobile germanium detector. All spectra showed the presence of particularly ^{60}Co but also ^{137}Cs were detected. The γ-spectrum measured close to the demineralising equipment is shown in Figure 3.2.6.

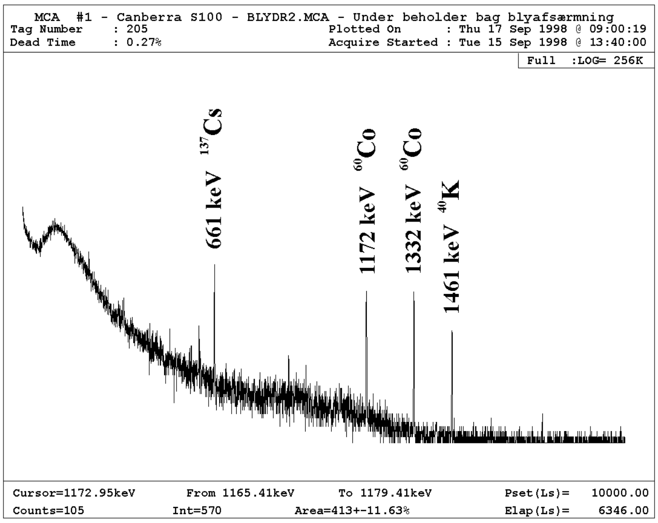


Figure 3.2.6 Gamma spectrum measured in the cellar close to the demineralising equipment.

The total activity inventory in the primary cooling system is extremely low and all parts of this system can probably be released as non-radioactive waste after approval by the authorities.

Before decommissioning can proceed, an estimate has to be made of the ^{14}C-activity concentration in the thermal column graphite as well as an estimate of the ^{133}Ba-, ^{152}Eu- and ^{154}Eu-activity in the concrete shield.

3.3 DR 3

3.3.1 General description

DR 3 was acquired for the Danish Atomic Energy Commission (AEK) in accordance with a contract between AEK and the English company Head Wrightson Processes (HWP). The buildings and technical installations were designed by Steensen & Varming working together with Professor Preben Hansen and Mogens Balslev, partly on the basis of the English projects for corresponding work on Pluto.

Danish reactor no. 3, DR 3 in short, is a research reactor built to test materials and new components for power reactors. It has been designed in accordance with drawings used during the construction of the two English reactors Pluto at Harwell and DMTR at Dounreay.

The reactor uses 19.75% enriched uranium in the fuel elements, and is moderated and cooled using heavy water. The reactor core consists of 26 fuel elements, which together contains approx. 3.3 kg of U-235, on average during routine operation at 10 MW. These elements are arranged in a square grid on the bottom of a D_2O-filled aluminium tank (see drawing no. 00-A4-094). Around the core, part of the heavy water acts as a reflector, and outside the aluminium tank, on the sides and under the bottom of this, is a graphite reflector approx. 30 cm thick. All this is contained in a steel tank, the walls of which are clad with boral and lead, to reduce the thermal neutron and gamma radiation from the core. On the outside is a biological shield of barytes concrete.

A total of 18 experimental tubes are located around the reactor core. Four of these are horizontal tangential tubes with a diameter of 7". The remaining 14 facilities are all vertical tubes, of which 4 x 7" tubes and 4 x 4" tubes are immersed in the heavy water in the aluminium tank, while the remaining 6 tubes are 4" tubes in the graphite reflector. It is also possible to place experimental materials in the 2" tubes located in the middle of the fuel elements. At 10 MW, the maximum thermal and fast flux is approx. 1.5×10^{14} n/cm^{2}s and approx. 4.5×10^{13} n/cm^{2}s respectively.

Position of buildings

The reactor complex, shown in Figure 3.3.1, is located in the western part of the Risø peninsula on a filled-in area north of the central road. Around the central reactor hall, building 213, are a number of buildings housing the necessary auxiliary machinery for the reactor. In a long wing running south-north in building 214 immediately to the east of the reactor hall, there is a “red” store furthest south, i.e. a store for low level radioactive equipment. Following this is a hall, the AH hall (Active Handling) with storage facilities for spent fuel elements and irradiated material, together with workshop areas for the maintenance of radioactive equipment. On the south wall of the hall is a distillation plant for processing degraded heavy water. After this comes a workshop hall (machine shop), and to the north of this are a number of offices intended for operating personnel, along with several changing rooms connected directly to the staff entry lock to the reactor hall.

Lying due north of the reactor hall is building 215, the so-called operations building, which houses the air-conditioning and cooling systems for experiments, compressed air plants, transformers, the main electrical distribution panel and a diesel plant for the stand-by power supply. Two emergency power batteries and electrical inverters are each located in their fire section in a side wing to the storage building north of building 215. To the west of the reactor hall and running south-north is building 217 (pump house), which as well as housing the secondary main and shut-down pumps accommodates part of the ventilation equipment and a water treatment plant. This plant is to provide the reactor with make-up water for the secondary cooling circuits and supply demineralized water to the entire Risø area. Lying due south of the reactor hall, at a distance of approx. 50 m from it, is building 218, which houses an emergency control room with remote indicating instruments for a number of parameters.

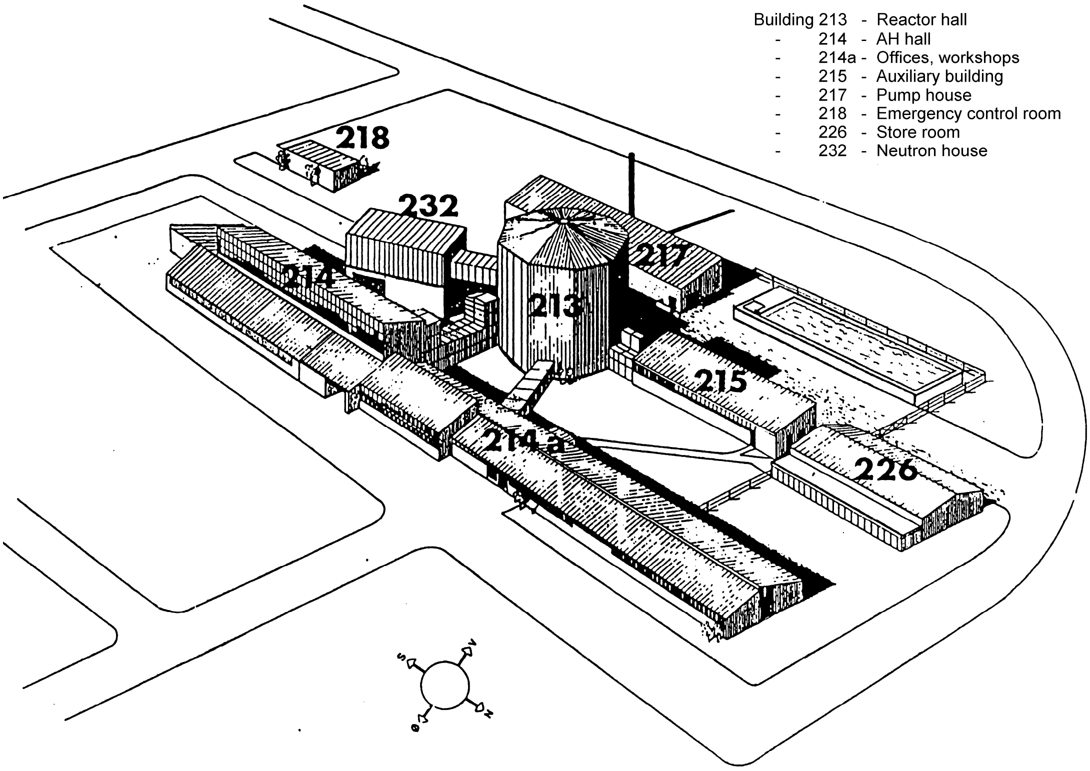


Figure 3.3.1 The DR 3 complex

Reactor hall

The reactor itself appears as a square, iron-clad concrete block with sides measuring 6.1 m in length and a height of 10.53 m, see Figure 3.3.2 and Figure 3.3.3. The four façades are to all intents and purposes oriented to the points of the compass, and are numbered so that façades 1, 2, 3 and 4 face north, east, south and west respectively.

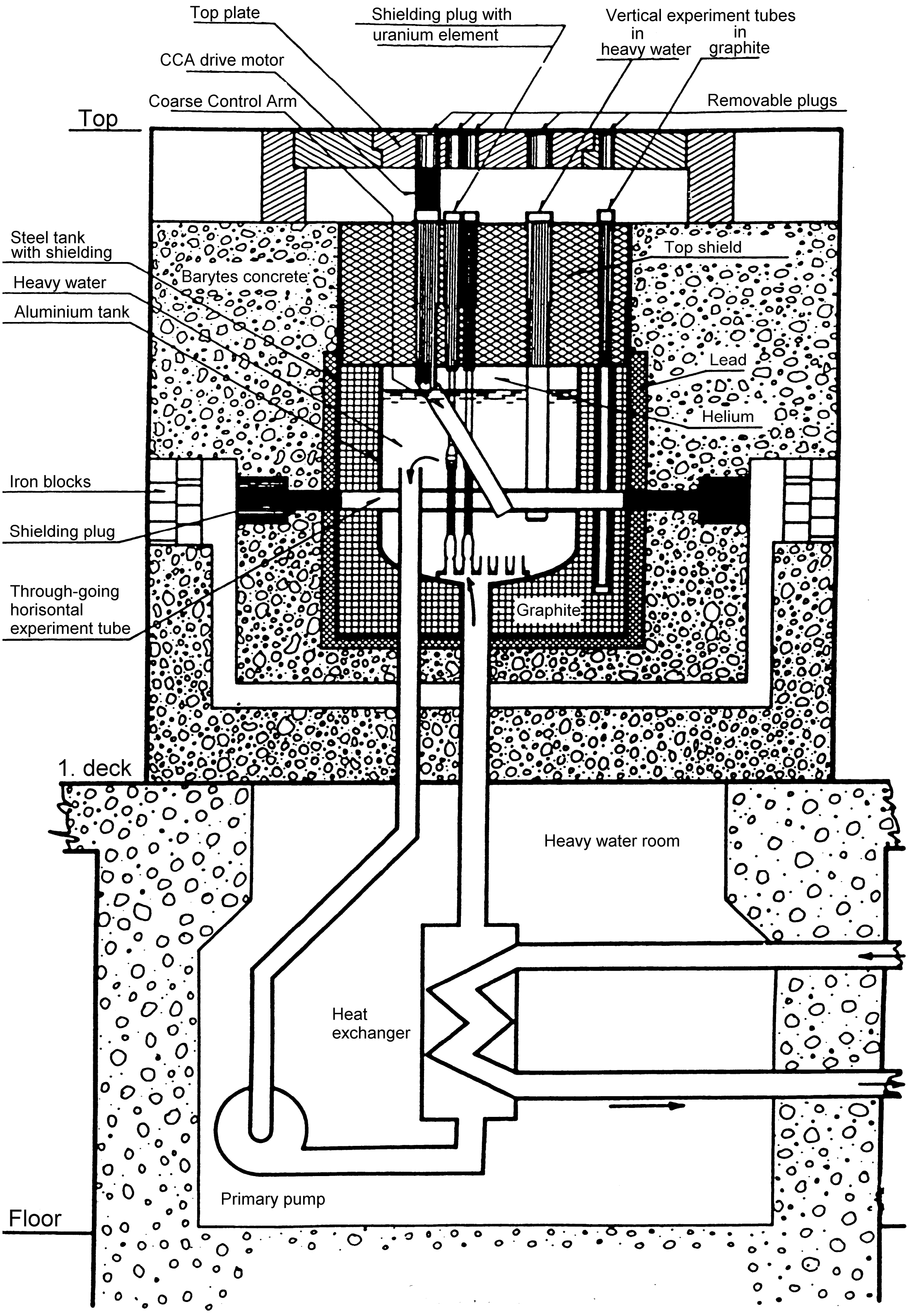


Figure 3.3.2 Sketch of the principle in DR 3

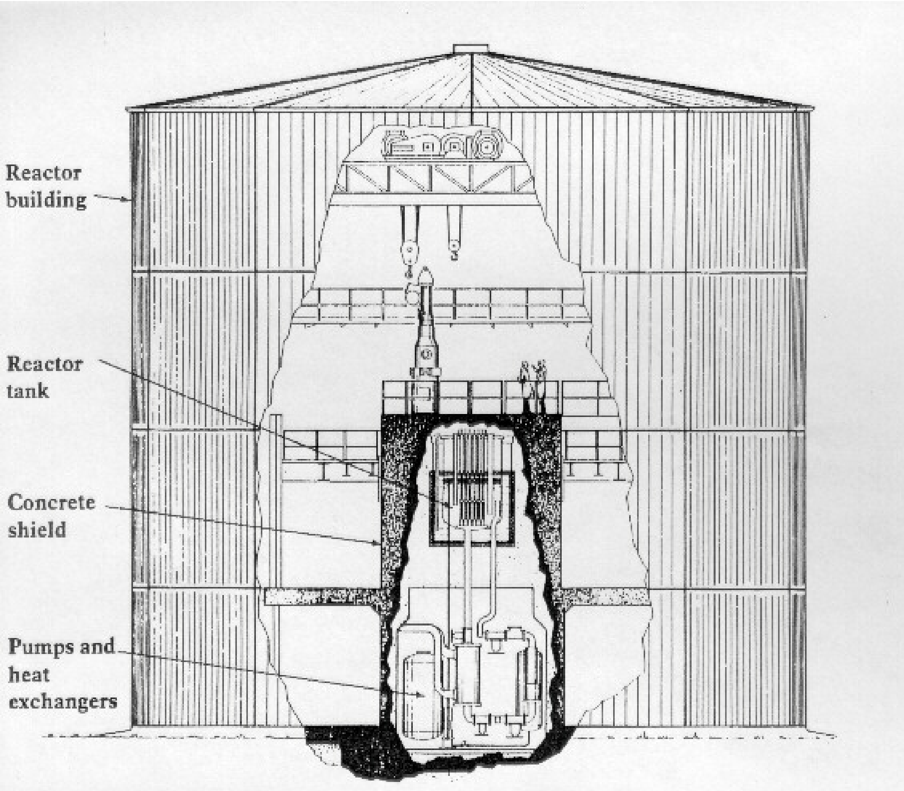


Figure 3.3.3 "X-ray picture" of the reactor building, showing size and position of the reactor.

The reactor block is positioned centrally in a hall created by an airtight cylinder with an internal diameter of 21 m and an internal height of 22.4 m at the centre. This produces a total volume of approx. 8000 m^3. The structural elements are positioned externally together with the insulation, so that seen from the outside the hall is 10-sided and clad with profiled aluminium sheets.

The hall is served by a ring crane with a lifting capacity of 25/10 tonnes. Its free lifting height above the reactor top is about 7 m.

The hall is divided into three levels: the basement, 1st level and reactor top with movable floor. These are connected on the one hand by stairs and on the other by a combined passenger and goods lift with a capacity of 550 kg.

The hall is entered via a staff entry lock and a vehicle entry lock. There is also an emergency entry lock on the 1st level, which is a marked escape route for persons coming from the reactor top and 1st level.

Reactor basement

The staff entry lock opens into the reactor basement opposite façade 2, as shown in Figure 3.3.4. Sited along this façade is the control room, from where operation of the reactor is controlled and monitored. The room measures 15 m^2 and has exits to both north and south. It houses most of the reactor's control - equipment, including a centrally located control desk with automatic and manual control of the reactor's control elements.

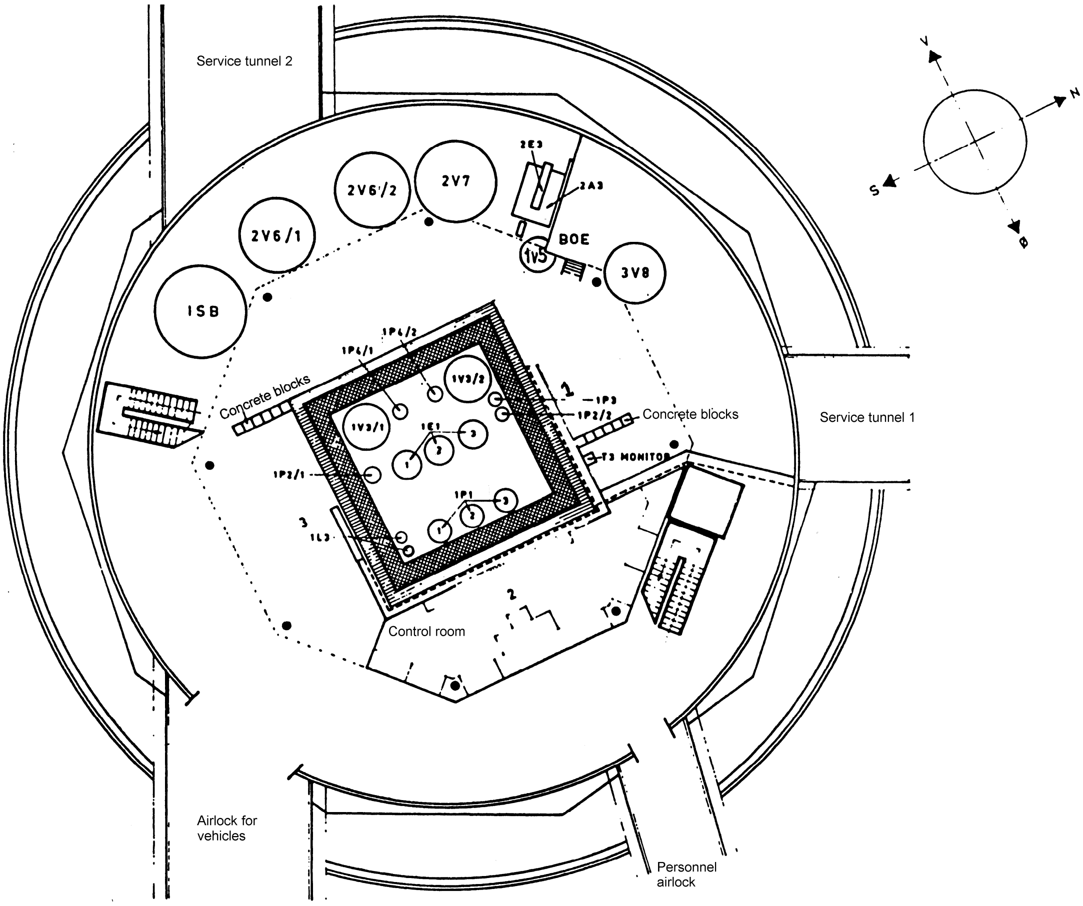


Figure 3.3.4 Horizontal cross section, basement level

Apart from the opening to the staff entry lock, the hall wall outside façade 2 is covered by electrical terminal boards (panel 5).

Outside façade 3 there is an open area connecting to the vehicle entry lock. This is used for transportation into and out of the hall, and for crane traffic between the three levels, the floor above having an opening roughly 40 m^2 in size. Part of this opening is blocked up at present by a gangway holding a neutron guide tube. From façades 3 and 4 there is access to a room (D_2O room) in the reactor block. The dimensions of this room are 5 x 5 m^2 and its height is 5.5 m, as the floor is 94 cm lower than the basement floor. The room contains pumps, heat exchangers and various tanks for the primary circuit. Access to this room is blocked during normal operation, and the high-density concrete walls of the reactor block shield the remaining part of the radiation coming from the primary water during operation.

A concrete cylinder ISB (Internal Storage Block) is located against the hall wall at the corner between façades 3 and 4. It houses irradiated fuel elements and experiments.

Opposite façade 4 are two expansion tanks and a gas tank for the helium used as cover gas above all heavy water surfaces. A heat exchanger for the shield cooling system is located between the gas tank and the hall wall. There is a purging and drying system for helium outside the corner between façades 4 and 1, and on a platform above, a helium cooler along with measuring instruments and recombiners for the D_2 and O_2 content of the helium system. In addition, there is a vessel here that is used for draining and transfers of D_2O (1V5).

All branch connections for the shield cooling system exit under the ceiling at façade 1, close to the corner with façade 4. The helium circulators are situated opposite and adjacent to these is a gas tank for CO_2 gas for the graphite reflector, behind which are the pumps for the shield cooling system. The hall wall opposite façade 1 is taken up by electrical panels (panel 4) supplying most of the motors in the reactor cooling systems and by an alarm panel for shield cooling and gas.

The lift is located outside the corner between façades 1 and 2 and adjacent to this are stairs leading up to the 1st level and the reactor top.

Building 214/214a
Normal access to the DR3 plant is through the entrance hall of the office wing. An electronic system ensures that only those with permission to enter can do so unaccompanied.

From the entrance hall, access is provided via the central aisle on the one hand to various offices etc., and on the other to the staff entry lock of the reactor hall and the workshop areas. The area in front of the staff entry lock and the workshops at the southern end of the building is a blue classified zone, in which personnel must wear overalls.

Adjoining the entrance area to the reactor hall, in addition to the washbasins and radiation monitors for checking staff, there are also showers for decontaminating personnel. Entry to the decontamination showers is exclusively via the health physics laboratory, immediately adjacent to the staff entry lock. These showers and washbasins by the lock entrance are connected to the active drainage system, whereas the bathing area in the white zone is connected to the normal drainage system.

AH hall
The AH (Active Handling) hall measures 12 x 39 m^2. It is served over its entire length by a travelling crane, the double lift of which has a capacity of 25/5 tonnes and a maximum lifting height of 7.6 m. The entrance to the hall from the changing room is at the north gable. At the northern end, to the west, is the entrance to the vehicle lock and access to the open air, while to the east there is a portal opening onto the outside. In the southern gable there is a so-called “red” store for active objects. A red classified zone has been established in the western and southern part of the hall (higher permitted contamination/ radiation level than in the blue zone), and along the wall there are a number of service stations with compressed air, water, drainage, electricity and a connection to a ventilation system with a separate exhaust. In the floor of the zone, close to the entrance to the vehicle lock, is a storage facility (EFESB = External Fuel Element Storage Block) with 80 positions, intended for the storage of spent fuel elements which no longer have to be cooled. A pit (VEFSB = Vertical Experimental Facilities Storage Block) located to the east of this has 24 storage spaces (229 mm Ø tubes) for irradiated experiments from the vertical experimental facilities. Both these pits are ventilated. Another pit, 1.83 x 1.83 m^2 and 1.07 m deep, is situated south of the EFESB. This is used for storing irradiated fuel element plugs and other radioactive equipment. South of this pit, a shielded facility has been erected for the connection of new fuel elements with reused shielding plugs. In the hall floor south of the VEFSB there are two apertures of 250 mm Ø and 3.55 m deep, which are used in connection with operations in the Boucher cell. Further south is an area intended for the inspection of the heavy water pumps, although part of this work is carried out in a special decontamination room. To the south of this is an area for inspection and maintenance of the control equipment. The area contains a lead-shielded cell, test stand and workroom. Further south is an area used for parking the horizontal flask 9R10 and the fuel tube flask 9R4. Four blocks in the floor each containing 12 holes are for the intermediate storage of released fuel tubes. At the southernmost end of the AH hall is a fourth zone around the cutting basin for fuel elements. The basin is 4 x 5 m^2 and 4.5 m deep and is the place where the fissile zone is cut free from the rest of the fuel elements. The fissile zones (tubes) are stored in racks. The basin has a separate cleaning circuit with filter and ion exchanger located in a pit to the west of the basin.

A distillation column for processing heavy water has been set up along the south wall.

The handling equipment for the cutting basin is suspended on the southern part of the east wall, and on the south wall itself is a system for filling and emptying the heavy water ion exchangers. Further north, by the east wall, is a pump stand for the heavy water pumps and an evaporation plant for heavy water rags. There is an annex of 3.7 x 7.5 m^2, height 3.5 m, in the northern part of the east wall. The annex consists of concrete walls around 30 horizontal 245 mm Ø steel tubes, intended to hold irradiated experiments and plugs from the horizontal experimental facilities. Situated behind this storage block is a smaller room, which is used for cleaning slightly contaminated tools and equipment. A further (large) decontamination room is located adjacent to this and is laid out as a wet room, where larger items can be handled and decontaminated. This room is used, for example, when working on the heavy water pumps.

The premises of the isotope department, where silicon irradiation is handled, can also be accessed from the AH hall.

Machine shop
The entrance to the workshop is located in the north gable of the AH hall. The workshop hall is 9 x 18 m^2 and is served by a lifting crane with a max. capacity of 1 t and a lifting height of 4 m. The workshop has supply lines for compressed air and electricity. A portal in the south wall offers access to outside; the passage is intended for the transportation of large objects to and from the workshop and is normally closed off. The workshop is located in the classified zone, so that work can be undertaken on contaminated/active equipment.

Auxiliary building, Building 215
In the auxiliary building running north-south and situated north of the reactor hall, there is a machine hall measuring 28 x 9 m^2 and below it a basement of a similar floor area. The northernmost end of the building contains two transformer rooms, one of which houses a 1600 KVA transformer for operating the reactor. The other transformer room holds a transformer for operating Risø’s heat pump plant. The north wall of the hall is taken up by the main electrical distribution panel.

The northern part of the building is taken up by 2 diesel generators for the stand-by supply (DG1, 120 KVA and DG2, 250 KVA). In the centre of the building is an air compressor, which acts as a stand-by for the compressed air supply from the boiler house. Located in the southern part of the building is the air inlet system for the reactor hall’s ventilation plant, and cooling machinery for the air conditioning system (and for experiment cooling).

The central part of the basement accommodates tanks and distribution system for compressed air. The southern part constitutes a separate room, which contains pumps, valves and an electrical panel for experiment cooling. The room also contains the pump arrangement for emptying the lift sump (following use of the sprinkler system in the hall), and the GAKO valve for closing off the hall’s air inlet connection in the event of a major release of activity in the hall.

Pump house, Building 217
The pump house, running north-south and situated west of the reactor hall, is divided up into three rooms. In the northernmost of these, measuring 9 x 15 m^2, are the two main pumps and the two shut-down cooling pumps, which circulate the secondary cooling water between the heat exchangers under the reactor and the fjord cooling plant. The main pipes and branches with shut-off valves for the pumps are located in a pit along the western side of the room. The basement contains two 10 m^3 tanks for non-active discharges and adjacent to these is a tank pit for 2 x 10 m^3 tanks for active discharge.

The centre room, measuring 9 x 12 m^2, houses extractor fans and filters for the ventilation system and in the basement room beneath this are 3 GAKO valves with related water tanks. One is situated in the outlet from the ventilation system to the absolute filters and extractor fans to a 22 m high chimney situated immediately to the west of the building. The other two are located in a recirculation branch with filter and fans for the ISB. There are also two shut-off valves for the active ventilation system in service tunnel II. The two 5 m high air traps for wastewater from the reactor hall are also to be found here. The air discharged from the system is monitored before being discharged via the chimney, the monitor effecting an isolation of the hall from the surroundings (building seal) if a certain level of activity is exceeded. The overall activity at the main filters and the active ventilation filters is also monitored.

The southernmost room houses the water treatment plant for the production of demineralized water for Risø. This only occupies a small part of the room, with the rest being used for storage purposes.

Storeroom, Building 226
To the north of the operations building is a building running north-south, which holds spares for the various reactor systems and consumer stores.

Battery room
Along the eastern side of the storeroom is a building containing the batteries for the guaranteed supply, the static current inverters, which supply emergency power to the primary shut-down pumps and the emergency cooling pumps.

Emergency control room, Building 218
Situated to the south of the reactor hall is the so-called emergency control room, which is constructed as a shelter, with walls 90-125 cm thick and a concrete ceiling 100 cm thick. The room, which measures 4.3 x 7.8 m^2 and is 2.4 m high, contains an emergency control console, from which certain functions in the reactor hall can be controlled remotely and monitored.

The room is ventilated via absolute filters to guarantee clean air even if the outside air is contaminated following an accident.

Neutron house, Building 232
Through a guide tube cold neutrons were led to spectrometers placed in the neutron house, south of the reactor building. In addition to the experimental facilities, there are offices and workshops in the neutron house.

3.3.2 Activity inventory

The major activity in DR 3 is the fission products situated in the irradiated fuel elements. During operation of the reactor the fission product activity in the 26 elements was about 10^{18} Bq, mainly short-lived and semi short-lived radionuclides. In the heavy water the tritium activity is about $3 \cdot 10^{15}$ Bq with a half-life of 12.3 years. Both fuel elements and heavy water will be removed before a decommissioning operation will start. Without fuel elements and heavy water the major activity will be found in the following parts of the reactor:

- reactor aluminium tank
- graphite reflector
- reactor steel tank
- top shield
- annular shield
- biological shield
- coarse control arms (CCAs)
- irradiation rigs and thimbles
- physics experimental facilities (TAS)

The upper six parts constitute the major structural materials in DR 3. They have a total weight of about 10^6 kg and nearly all the residual activity will be found here, approximately $2 \cdot 10^{14}$ Bq of semi long-lived and long-lived radionuclides a few years after the final shut down of DR 3. The activity contents in rigs and thimbles and in the structural materials can be estimated from calculated activities for the DIDO reactor at the Harwell site in the UK[2]. DR3 was run at 10 MW and DIDO at 25 MW and the DR3 activities are, therefore, roughly a factor of two lower than those in DIDO. The radionuclide activity

[2] *Possible Methods and Timings for Stage 3. Decommissioning of DIDO and PLUTO.* DIDO Operators' Meeting. FZK Jülich, 2 - 5 June 1997. N.W. Crick, Manorcroft Consultancy Ltd.

inventory in the major structural components of DIDO: (1) top shield, (2) reactor tank, (3) graphite, (4) steel tank, (5) annular shield and (6) biological shield are shown in Table 3.3.1. The average activity of γ-emitting radionuclides in rigs/thimbles is about 1 - 2% of the activity in the major structural materials having a half-life of about five years over the next 40 years as shown in Figure 3.3.5.

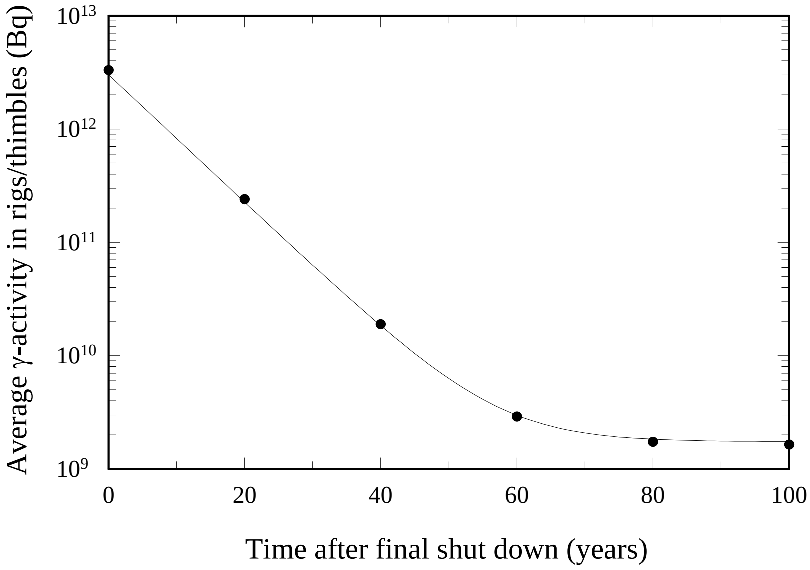


Figure 3.3.5 Average content of γ-emitting radionuclides in DIDO reactor rigs/thimbles as a function of time after shut down.

Table 3.3.1 Activity inventories in DR3 structure material at 10, 25 and 40 years after final shut down of DR3 can be estimated from calculated activities for the similar DIDO reactor at Harwell. DR3 was run at 10 MW and DIDO at 25 MW and the DR3 activities are, therefore, roughly a factor two lower than indicated in the table.

ACTIVITY INVENTORY IN DR 3 STRUCTURAL MATERIALS (Bq)

Component	Mass (kg)	H-3			C-14			Fe-55			Co-60			Ni-63			Zn-65			Cd-113m			Ba-133			Eu-152			Eu-154			Sum
		10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	10 y	25 y	40 y	(20 y)
Al tank	5.3E+02							7.0E+11	1.5E+10	3.4E+08	2.5E+12	3.4E+11	4.7E+10				3.1E+07	5.5E+00	9.8E-07													
FE header plate	6.5E+01							1.2E+11	2.6E+09	5.8E+07	4.1E+11	5.8E+10	8.0E+09				5.2E+06	9.2E-01	1.7E-07													
FE nozzles	1.9E+02							4.3E+10	9.6E+08	2.1E+07	1.5E+11	2.1E+10	2.9E+09				1.9E+06	3.4E-01	6.1E-08													
Outlet pipes	4.9E+01							9.6E+10	2.1E+09	4.7E+07	3.4E+11	4.7E+10	6.6E+09				4.2E+06	7.5E-01	1.3E-07													
Overflow pipe	1.7E+00							3.2E+09	7.2E+07	1.6E+06	1.1E+10	1.6E+09	2.2E+08				1.4E+05	2.5E-02	4.5E-09													
Inlet pipes	6.2E+01							9.3E+09	2.1E+08	4.6E+06	3.3E+10	4.6E+09	6.4E+08				4.1E+05	7.3E-02	1.3E-08													
Total RAT	**9.0E+02**		**0.0E+00**			**0.0E+00**		**9.7E+11**	**2.1E+10**	**4.8E+08**	**3.4E+12**	**4.7E+11**	**6.6E+10**		**0.0E+00**		**4.2E+07**	**7.6E+00**	**1.4E-06**		**0.0E+00**			**0.0E+00**			**0.0E+00**			**0.0E+00**		**9.9E+11**
Graphite reflector	**2.6E+04**	**3.1E+12**	**1.3E+12**	**5.7E+11**	**2.0E+11**	**2.0E+11**	**2.0E+11**	**1.5E+10**	**3.4E+08**	**7.5E+06**	**1.8E+11**	**2.6E+10**	**3.6E+09**	**1.0E+11**	**9.4E+10**	**8.5E+10**	**1.1E+05**	**1.9E-02**	**3.4E-09**	**1.4E+04**	**6.9E+03**	**3.3E+03**	**8.8E+08**	**3.3E+08**	**1.2E+08**		**0.0E+00**			**0.0E+00**		**2.1E+12**
Reactor steel tank	8.5E+03							6.3E+11	1.4E+10	3.1E+08	2.2E+11	3.0E+10	4.2E+09	3.7E+10	3.3E+10	3.0E+10	6.2E+05	1.1E-01	2.0E-08	1.0E+04	4.8E+03	2.3E+03										
Boral liner	6.9E+02							1.3E+09	3.0E+07	6.6E+05	4.6E+08	6.5E+07	9.0E+06	7.9E+07	7.1E+07	6.4E+07	1.3E+03	2.4E-04	4.2E-11	2.2E+01	1.0E+01	5.0E+00										
Lead gamma shield	4.5E+04							9.4E+08	2.1E+07	4.6E+05	3.2E+08	4.5E+07	6.3E+06	5.5E+07	5.0E+07	4.5E+07	9.3E+02	1.7E-04	3.0E-11	1.5E+01	7.2E+00	3.5E+00										
Steel outer sheath	2.5E+03							5.4E+10	1.2E+09	2.7E+07	1.9E+10	2.6E+09	3.6E+08	3.2E+09	2.9E+09	2.6E+09	5.3E+04	9.5E-03	1.7E-09	8.7E+02	4.1E+02	2.0E+02										
Thermal bond	5.7E+03							6.3E+10	1.4E+09	3.1E+07	2.2E+10	3.0E+09	4.2E+08	3.7E+09	3.3E+09	3.0E+09	6.2E+04	1.1E-02	2.0E-09	1.0E+03	4.8E+02	2.3E+02										
D2O inlet pipes	2.4E+01							2.0E+10	4.5E+08	9.9E+06	7.0E+09	9.7E+08	1.3E+08	1.2E+09	1.1E+09	9.7E+08	2.0E+04	3.5E-03	6.3E-10	3.2E+02	1.6E+02	7.4E+01										
D2O outlet pipes	2.8E+01							2.4E+10	5.3E+08	1.2E+07	8.2E+09	1.1E+09	1.6E+08	1.4E+09	1.3E+09	1.1E+09	2.3E+04	4.2E-03	7.5E-10	3.8E+02	1.8E+02	8.7E+01										
Horizontal beam holes	7.8E+01							4.9E+10	1.1E+09	2.4E+07	1.7E+10	2.4E+09	3.3E+08	2.9E+09	2.6E+09	2.4E+09	4.9E+04	8.7E-03	1.5E-09	7.9E+02	3.8E+02	1.8E+02										
Vertical exp. thimble	1.0E+02							7.2E+10	1.6E+09	3.5E+07	2.5E+10	3.4E+09	4.8E+08	4.2E+09	3.8E+09	3.4E+09	7.1E+04	1.3E-02	2.3E-09	1.2E+03	5.5E+02	2.6E+02										
Total steel tank	**6.3E+04**		**0.0E+00**			**0.0E+00**		**9.1E+11**	**2.0E+10**	**4.5E+08**	**3.1E+11**	**4.4E+10**	**6.1E+09**	**5.4E+10**	**4.8E+10**	**4.4E+10**	**9.0E+05**	**1.6E-01**	**2.9E-08**	**1.5E+04**	**7.0E+03**	**3.4E+03**		**0.0E+00**			**0.0E+00**			**0.0E+00**		**2.1E+11**
Bottom SS disc	3.4E+02	1.6E+12	7.0E+11	3.0E+11	2.0E+05	2.0E+05	2.0E+05	3.6E+12	8.0E+10	1.8E+09	5.3E+12	7.4E+11	1.0E+11	4.8E+12	4.3E+12	3.9E+12	1.5E+04	2.8E-03	4.9E-10	6.0E+10	2.9E+10	1.4E+10	9.0E+06	3.4E+06	1.2E+06	7.7E+10	3.6E+10	1.7E+10	7.5E+09	2.2E+09	6.6E+08	
Cadium layer	4.0E+01	1.2E+10	5.1E+09	2.2E+09	1.4E+03	1.4E+03	1.4E+03	2.6E+10	5.9E+08	1.3E+07	3.9E+10	5.4E+09	7.6E+08	3.5E+10	3.2E+10	2.9E+10	1.1E+02	2.0E-05	3.6E-12	4.4E+08	2.1E+08	1.0E+08	6.6E+04	2.5E+04	9.1E+03	5.6E+08	2.6E+08	1.2E+08	5.5E+07	1.6E+07	4.9E+06	
Lead gamma shield	2.6E+03	3.9E+08	1.7E+08	7.2E+07	4.7E+01	4.7E+01	4.7E+01	8.6E+08	1.9E+07	4.3E+05	1.3E+09	1.8E+08	2.5E+07	1.2E+09	1.0E+09	9.4E+08	3.7E+00	6.6E-07	1.2E-13	1.4E+07	6.9E+06	3.3E+06	2.2E+03	8.0E+02	3.0E+02	1.8E+07	8.5E+06	4.0E+06	1.8E+06	5.3E+05	1.6E+05	
Mild steel plate	6.8E+02	6.4E+09	2.8E+09	1.2E+09	7.8E+02	7.8E+02	7.7E+02	1.4E+10	3.2E+08	7.0E+06	2.1E+10	2.9E+09	4.1E+08	1.9E+10	1.7E+10	1.5E+10	6.1E+01	1.1E-05	1.9E-12	2.4E+08	1.1E+08	5.4E+07	3.6E+04	1.3E+04	4.9E+03	3.0E+08	1.4E+08	6.5E+07	2.9E+07	8.8E+06	2.6E+06	
Iron shot concrete	1.4E+04	3.4E+11	1.5E+11	6.4E+10	4.2E+04	4.2E+04	4.1E+04	7.6E+11	1.7E+10	3.8E+08	1.1E+12	1.6E+11	2.2E+10	1.0E+12	9.2E+11	8.3E+11	3.3E+03	5.8E-04	1.0E-10	1.3E+10	6.1E+09	2.9E+09	1.9E+06	7.1E+05	2.6E+05	1.6E+10	7.5E+09	3.5E+09	1.6E+09	4.7E+08	1.4E+08	
SS master plate	6.1E+02	1.3E+05	5.5E+04	2.4E+04	1.6E-02	1.6E-02	1.5E-02	2.8E+05	6.3E+03	1.4E+02	4.2E+05	5.9E+04	8.2E+03	3.8E+05	3.4E+05	3.1E+05	1.2E-03	2.2E-10	3.9E-17	4.8E+03	2.3E+03	1.1E+03	7.1E-01	2.6E-01	9.8E-02	6.0E+03	2.8E+03	1.3E+03	5.9E+02	1.8E+02	5.2E+01	
SS cylinder	7.7E+02	4.1E+10	1.8E+10	7.7E+09	5.0E+03	5.0E+03	5.0E+03	9.1E+10	2.0E+09	4.5E+07	1.4E+11	1.9E+10	2.6E+09	1.2E+11	1.1E+11	9.9E+10	3.9E+02	7.0E-05	1.2E-11	1.5E+09	7.3E+08	3.5E+08	2.3E+05	8.5E+04	3.2E+04	1.9E+09	9.0E+08	4.2E+08	1.9E+08	5.7E+07	1.7E+07	
FE hole liners	8.9E+02	2.1E+11	9.1E+10	3.9E+10	2.6E+04	2.5E+04	2.5E+04	4.7E+11	1.0E+10	2.3E+08	6.9E+11	9.6E+10	1.3E+10	6.2E+11	5.6E+11	5.1E+11	2.0E+03	3.6E-04	6.4E-11	7.8E+09	3.7E+09	1.8E+09	1.2E+06	4.3E+05	1.6E+05	9.9E+09	4.6E+09	2.1E+09	9.7E+08	2.9E+08	8.6E+07	
6" hole liners	1.7E+02	3.0E+10	1.3E+10	5.5E+09	3.6E+03	3.6E+03	3.6E+03	6.6E+10	1.5E+09	3.3E+07	9.8E+10	1.4E+10	1.9E+09	8.8E+10	7.9E+10	7.2E+10	2.8E+02	5.1E-05	9.0E-12	1.1E+09	5.3E+08	2.5E+08	1.7E+05	6.1E+04	2.3E+04	1.4E+09	6.5E+08	3.0E+08	1.4E+08	4.1E+07	1.2E+07	
4" hole liners	1.7E+02	1.0E+10	4.4E+09	1.9E+09	1.2E+03	1.2E+03	1.2E+03	2.2E+10	5.0E+08	1.1E+07	3.3E+10	4.6E+09	6.4E+08	3.0E+10	2.7E+10	2.4E+10	9.6E+01	1.7E-05	3.1E-12	3.7E+08	1.8E+08	8.5E+07	5.6E+04	2.1E+04	7.7E+03	4.7E+08	2.2E+08	1.0E+08	4.6E+07	1.4E+07	4.1E+06	
2" hole liners	1.7E+02	1.7E+10	7.2E+09	3.1E+09	2.0E+03	2.0E+03	2.0E+03	3.7E+10	8.2E+08	1.8E+07	5.5E+10	7.6E+09	1.1E+09	4.9E+10	4.5E+10	4.0E+10	1.6E+02	2.8E-05	5.1E-12	6.2E+08	3.0E+08	1.4E+08	9.3E+04	3.4E+04	1.3E+04	7.9E+08	3.7E+08	1.7E+08	7.7E+07	2.3E+07	6.8E+06	
Vertical hole plugs	5.8E+02	2.2E+10	9.6E+09	4.1E+09	2.7E+03	2.7E+03	2.7E+03	4.9E+10	1.1E+09	2.4E+07	7.3E+10	1.0E+10	1.4E+09	6.6E+10	5.9E+10	5.3E+10	2.1E+02	3.8E-05	6.7E-12	8.2E+08	3.9E+08	1.9E+08	1.2E+05	4.6E+04	1.7E+04	1.0E+09	4.9E+08	2.3E+08	1.0E+08	3.0E+07	9.1E+06	
Total top shield	**2.1E+04**	**2.3E+12**	**1.0E+12**	**4.3E+11**	**2.8E+05**	**2.8E+05**	**2.8E+05**	**5.1E+12**	**1.1E+11**	**2.5E+09**	**7.6E+12**	**1.1E+12**	**1.5E+11**	**6.9E+12**	**6.2E+12**	**5.6E+12**	**2.2E+04**	**3.9E-03**	**7.0E-10**	**8.6E+10**	**4.1E+10**	**2.0E+10**	**1.3E+07**	**4.8E+06**	**1.8E+06**	**1.1E+11**	**5.1E+10**	**2.4E+10**	**1.1E+10**	**3.2E+09**	**9.5E+08**	**1.0E+13**
Mild steel outer	2.8E+03	6.2E+10	2.7E+10	1.1E+10	1.1E+04	1.1E+04	1.1E+04	8.9E+10	2.0E+09	4.4E+07	1.6E+10	2.2E+09	3.0E+08	3.1E+09	2.8E+09	2.5E+09	5.9E+02	1.0E-04	1.9E-11	2.3E+09	1.1E+09	5.3E+08	3.4E+05	1.3E+05	4.7E+04	2.9E+09	1.3E+09	6.2E+08	2.8E+08	8.4E+07	2.5E+07	
Mild steel base	2.7E+02	3.2E+10	1.4E+10	6.0E+09	6.0E+03	6.0E+03	5.9E+03	4.7E+10	1.0E+09	2.3E+07	8.1E+09	1.1E+09	1.6E+08	1.6E+09	1.5E+09	1.3E+09	3.1E+02	5.5E-05	9.8E-12	1.2E+09	5.8E+08	2.8E+08	1.8E+05	6.6E+04	2.5E+04	1.5E+09	7.0E+08	3.3E+08	1.5E+08	4.4E+07	1.3E+07	
Cadmium layer	7.0E+01	1.7E+10	7.4E+09	3.2E+09	3.2E+03	3.2E+03	3.2E+03	2.5E+10	5.6E+08	1.2E+07	4.4E+09	6.1E+08	8.4E+07	8.7E+08	7.8E+08	7.0E+08	1.6E+02	2.9E-05	5.3E-12	6.5E+08	3.1E+08	1.5E+08	9.5E+04	3.5E+04	1.3E+04	8.1E+08	3.8E+08	1.8E+08	7.9E+07	2.4E+07	7.0E+06	
Lead gamma shield	3.1E+03	1.8E+08	7.6E+07	3.3E+07	3.3E+01	3.3E+01	3.3E+01	2.6E+08	5.7E+06	1.3E+05	4.4E+07	6.2E+06	8.6E+05	8.9E+06	8.0E+06	7.2E+06	1.7E+00	3.0E-07	5.4E-14	6.6E+06	3.2E+06	1.5E+06	9.7E+02	3.6E+02	1.3E+02	8.3E+06	3.8E+06	1.8E+06	8.0E+05	2.4E+05	7.2E+04	
3/4 inch mild steel plate	4.1E+02	1.0E+10	4.5E+09	1.9E+09	1.9E+03	1.9E+03	1.9E+03	1.5E+10	3.3E+08	7.4E+06	2.6E+09	3.6E+08	5.0E+07	5.2E+08	4.7E+08	4.2E+08	9.9E+01	1.8E-05	3.1E-12	3.9E+08	1.9E+08	8.9E+07	5.7E+04	2.1E+04	7.9E+03	4.9E+08	2.3E+08	1.0E+08	4.7E+07	1.4E+07	4.2E+06	
Fe shot concrete	1.5E+04	4.9E+11	2.1E+11	9.1E+10	9.1E+04	9.1E+04	9.1E+04	7.1E+11	1.6E+10	3.5E+08	1.2E+11	1.7E+10	2.4E+09	2.5E+10	2.2E+10	2.0E+10	4.7E+03	8.4E-04	1.5E-10	1.8E+10	8.9E+09	4.2E+09	2.7E+06	1.0E+06	3.7E+05	2.3E+10	1.1E+10	5.0E+09	2.2E+09	6.7E+08	2.0E+08	
Vertical hole liners	1.7E+02	2.5E+09	1.1E+09	4.6E+08	4.5E+02	4.5E+02	4.5E+02	3.6E+09	7.9E+07	1.8E+06	6.2E+08	8.6E+07	1.2E+07	1.2E+08	1.1E+08	1.0E+08	2.3E+01	4.2E-06	7.5E-13	9.2E+07	4.4E+07	2.1E+07	1.4E+04	5.0E+03	1.9E+03	1.2E+08	5.4E+07	2.5E+07	1.1E+07	3.3E+06	1.0E+06	
Vertical hole plugs	1.6E+03	2.2E+10	9.4E+09	4.0E+09	4.0E+03	4.0E+03	4.0E+03	3.2E+10	7.0E+08	1.6E+07	5.5E+09	7.6E+08	1.1E+08	1.1E+09	9.8E+08	8.9E+08	2.1E+02	3.7E-05	6.6E-12	8.2E+08	3.9E+08	1.9E+08	1.2E+05	4.5E+04	1.7E+04	1.0E+09	4.7E+08	2.2E+08	9.9E+07	3.0E+07	8.8E+06	
Total annular shield	**2.3E+04**	**6.4E+11**	**2.7E+11**	**1.2E+11**	**1.2E+05**	**1.2E+05**	**1.2E+05**	**9.2E+11**	**2.1E+10**	**4.6E+08**	**1.6E+11**	**2.2E+10**	**3.1E+09**	**3.2E+10**	**2.9E+10**	**2.6E+10**	**6.1E+03**	**1.1E-03**	**1.9E-10**	**2.4E+10**	**1.1E+10**	**5.5E+09**	**3.5E+06**	**1.3E+06**	**4.9E+05**	**3.0E+10**	**1.4E+10**	**6.5E+09**	**2.9E+09**	**8.7E+08**	**2.6E+08**	**5.4E+11**
Barytes 20 cm annulus	3.4E+04	1.1E+13	4.7E+12	2.0E+12	1.3E+03	1.3E+03	1.3E+03	1.1E+11	2.4E+09	5.3E+07	6.0E+10	8.3E+09	1.2E+09	3.2E+10	2.9E+10	2.6E+10	1.0E+05	1.8E-02	3.3E-09				7.2E+11	2.7E+11	1.0E+11	5.1E+11	2.4E+11	1.1E+11	5.0E+10	1.5E+10	4.4E+09	
Barytes 20 cm annulus	5.0E+04	2.4E+12	1.0E+12	4.4E+11	2.8E+02	2.8E+02	2.8E+02	2.3E+10	5.2E+08	1.2E+07	1.3E+10	1.8E+09	2.5E+08	7.1E+09	6.4E+09	5.7E+09	2.2E+04	4.0E-03	7.2E-10				1.6E+11	5.8E+10	2.2E+10	1.1E+11	5.1E+10	2.4E+10	1.1E+10	3.2E+09	9.6E+08	
Barytes 20 cm annulus	6.1E+04	4.2E+11	1.8E+11	7.7E+10	5.0E+01	5.0E+01	5.0E+01	4.1E+09	9.2E+07	2.0E+06	2.3E+09	3.2E+08	4.4E+07	1.2E+09	1.1E+09	1.0E+09	4.0E+03	7.1E-04	1.3E-10				2.8E+10	1.0E+10	3.8E+09	2.0E+10	9.1E+09	4.2E+09	1.9E+09	5.7E+08	1.7E+08	
Barytes remainder	5.9E+05	2.8E+11	1.2E+11	5.1E+10	3.3E+01	3.3E+01	3.3E+01	2.7E+09	6.1E+07	1.3E+06	1.5E+09	2.1E+08	2.9E+07	8.3E+08	7.5E+08	6.7E+08	2.6E+03	4.7E-04	8.4E-11				1.8E+10	6.9E+09	2.6E+09	1.3E+10	6.0E+09	2.8E+09	1.3E+09	3.8E+08	1.1E+08	
Horizontal hole liners	4.6E+02	6.1E+10	2.6E+10	1.1E+10	7.4E+00	7.4E+00	7.4E+00	6.1E+08	1.4E+07	3.0E+05	3.4E+08	4.7E+07	6.5E+06	1.8E+08	1.7E+08	1.5E+08	5.9E+02	1.0E-04	1.9E-11				4.1E+09	1.5E+09	5.7E+08	2.9E+09	1.3E+09	6.2E+08	2.8E+08	8.4E+07	2.5E+07	
Horizontal hole plugs	3.6E+02	2.5E+10	1.1E+10	4.6E+09	3.0E+00	3.0E+00	3.0E+00	2.5E+08	5.5E+06	1.2E+05	1.4E+08	1.9E+07	2.6E+06	7.4E+07	6.7E+07	6.0E+07	2.4E+02	4.2E-05	7.5E-12				1.7E+09	6.2E+08	2.3E+08	1.2E+09	5.4E+08	2.5E+08	1.1E+08	3.4E+07	1.0E+07	
Cast iron ring	2.6E+04	2.8E+04	1.2E+04	5.2E+03	3.4E-06	3.4E-06	3.4E-06	2.8E+02	6.1E+00	1.4E-01	1.5E+02	2.1E+01	3.0E+00	8.4E+01	7.6E+01	6.8E+01	2.7E-04	4.8E-11	8.5E-18				1.9E+03	6.9E+02	2.6E+02	1.3E+03	6.1E+02	2.8E+02	1.3E+02	3.8E+01	1.1E+01	
Reactor top plate	6.8E+03	1.6E+04	7.0E+03	3.0E+03	2.0E-06	2.0E-06	2.0E-06	1.6E+02	3.6E+00	8.0E-02	9.0E+01	1.3E+01	1.7E+00	4.9E+01	4.4E+01	4.0E+01	1.6E-04	2.8E-11	5.0E-18				1.1E+03	4.1E+02	1.5E+02	7.7E+02	3.6E+02	1.7E+02	7.5E+01	2.2E+01	6.7E+00	
Top plate outer ring	1.8E+04	2.0E+03	8.8E+02	3.8E+02	2.5E-07	2.5E-07	2.5E-07	2.0E+01	4.5E-01	1.0E-02	1.1E+01	1.6E+00	2.2E-01	6.1E+00	5.5E+00	5.0E+00	1.9E-05	3.5E-12	6.2E-19				1.4E+02	5.1E+01	1.9E+01	9.6E+01	4.4E+01	2.1E+01	9.4E+00	2.8E+00	8.3E-01	
Outer casting	2.9E+04	1.8E+04	7.9E+03	3.4E+03	2.2E-06	2.2E-06	2.2E-06	1.8E+02	4.0E+00	9.0E-02	1.0E+02	1.4E+01	2.0E+00	5.5E+01	5.0E+01	4.5E+01	1.8E-04	3.1E-11	5.6E-18				1.2E+03	4.6E+02	1.7E+02	8.6E+02	4.0E+02	1.9E+02	8.5E+01	2.5E+01	7.5E+00	
Total biological shield	**8.2E+05**	**1.4E+13**	**6.0E+12**	**2.6E+12**	**1.7E+03**	**1.7E+03**	**1.7E+03**	**1.4E+11**	**3.1E+09**	**6.8E+07**	**7.7E+10**	**1.1E+10**	**1.5E+09**	**4.2E+10**	**3.8E+10**	**3.4E+10**	**1.3E+05**	**2.4E-02**	**4.2E-09**		**0.0E+00**		**9.3E+11**	**3.5E+11**	**1.3E+11**	**6.5E+11**	**3.0E+11**	**1.4E+11**	**6.4E+10**	**1.9E+10**	**5.7E+09**	**8.9E+12**

Table 3.3.2 Simplified activity inventories in DR3 structure materials at 10 years after final shut down. Excerpts from Table 3.3.1, showing calculated activities for the similar British DIDO reactor at Harwell. DR3 was run at 10 MW while was run DIDO at 25 MW, but for somewhat shorter time. The expected DR3 activities are therefore roughly a factor two lower than indicated in the table.

Half life, years		12.3	5736	2.6	5.3	93	0.7	14	10.5	12.4	8.5
	mass	^{3}H	^{14}C	^{55}Fe	^{60}Co	^{63}Ni	^{65}Zn	^{113m}Cd	^{133}Ba	^{152}Eu	^{154}Eu
	t	**GBq at 10 years after shut down**									
Reactor tank (aluminium)	0.9	-	-	970	3400	-	0.04	-	-	-	-
Graphite	26	3100	200	15	180	100	low	low	0.9	-	-
Steel tank with 45 t Pb	63	-	-	910	310	54	0.001	low	-	-	-
Top shield with 2.6 t Pb and 0.04 Cd	21	2300	low	5100	7600	690	low	86	0.013	110	11
Annular shield with 3.1 t Pb and 0.07 Cd	23	640	low	920	160	32	low	24	0.004	30	2.9
Biological shield (concrete)	816	14000	low	140	77	42	low	-	930	650	64
Sum:	**950**	**20 000**	**200**	**8000**	**11700**	**920**	**<0.1**	**110**	**931**	**790**	**78**

The power level of the PLUTO and DIDO reactors during operation was about a factor of two higher than the power level of 10 MW of DR 3. Therefore the content of long-lived activation products in the structure materials of the PLUTO/DIDO reactors is higher (up to a factor of two) than the content in similar materials in DR 3.

A leak in the drainpipe at the bottom of the reactor tank occurred in 1999 and the draining system has been modified by inserting a conical plug into the draining hole. A conical hole was produced to adapt the plug and aluminium chips were produced during this operation. The activity content in these chips has been determined by gamma-spectroscopy and a crude estimate of the total content of activity in the tank has been made. The measured gamma-spectrum is shown in Figure 3.3.6 below.

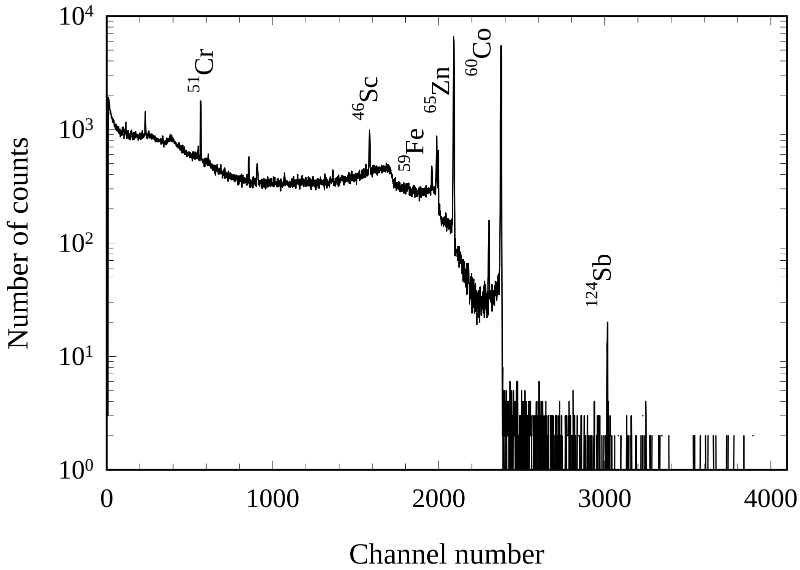


Figure 3.3.6 Gamma-spectrum for 0.8 grams of aluminium chips from the bottom of the reactor tank measured over 3000 seconds.

The dominating radionuclides are ^{60}Co, ^{51}Cr and ^{65}Zn being produced by the following activation of impurities in aluminium, mainly iron but also antimony and chromium, by the following processes:

$$^{50}\mathrm{Cr}(n, \gamma)^{51}\mathrm{Cr} \text{ and } ^{54}\mathrm{Fe}(n, \alpha)^{51}\mathrm{Cr}$$

$$^{58}\mathrm{Fe}(n, \gamma)^{59}\mathrm{Fe} \rightarrow {}^{59}\mathrm{Co}(n, \gamma)^{60}\mathrm{Co}$$

$$^{59}\mathrm{Co}(n, \gamma)^{60}\mathrm{Co} \text{ and } ^{64}\mathrm{Zn}(n, \gamma)^{65}\mathrm{Zn}$$

Assuming that the measured activity concentration in the aluminium chips is a representative average for the whole tank, the total activity content in the tank has been calculated as shown in Table 3.3.3.

Table 3.3.3 Calculated activity content in the aluminium reactor tank on 10 March 2000, 106 days after the shut down of the reactor on 26 November 1999.

Radionuclide	Activity content (Bq)
^{46}Sc	$1.9 \cdot 10^{10}$
^{51}Cr	$1.0 \cdot 10^{11}$
^{59}Fe	$1.4 \cdot 10^{10}$
^{60}Co	$3.0 \cdot 10^{11}$
^{65}Zn	$5.5 \cdot 10^{10}$
^{124}Sb	$1.3 \cdot 10^{9}$

The calculated content of $3.0 \cdot 10^{11}$ Bq ^{60}Co is around a factor of 30 lower than the content in the DIDO reactor tank given in Table 6.3.1. The reason might be that the total neutron fluence to the chips over 40 years is not a representative average fluence to the aluminium tank. If possible, neutron activation analyses (using DR 1) on (existing) representative samples of the tank material should be made.

3.4 Fuel fabrication facility

3.4.1 General description

The fuel fabrication facility is a part of the Technology Hall. Within the facility fuel elements for the DR 3 reactor have been produced and handled for more than 35 years. Up to 1988 the fabrication was based on highly enriched (93 % U^{235}) metallic uranium; but from then on the elements have been made from low enriched (< 20%U^{235}) U_3Si_2 powder.

Today the fabrication is concentrated in a minor part of the Technology Hall, but earlier it was spread all over the building's first floor. Those areas not used for the production anymore have been declassified to white areas without any difficulties.

During most of the fabrication process the uranium is handled – and has always been - encapsulated without any risk for contamination of personnel or tools, equipment and working area. Today unclad material is handled in one room only, which is classified as red contamination area. Due to the pyroforic nature of the U_3Si_2 powder it is handled only under argon atmosphere in closed containers or in gloveboxes. This limits the necessary cleanup mainly to the equipment in this room and the connected ventilation channels.

Additionally, the whole building (and building 228, as well) has a separate drain system for active water, which is picked up in tanks placed underground north of the Technology Hall, from where it has been transported to the waste treatment plant.

3.4.2 Activity inventory

When all fuel material – in the form of unused fresh powder, fuel plates, samples etc. - has been transferred to DR 3, the only activity left will be in the form of uranium contaminated equipment in the powder room, as well as the connected ventilation system and drain pipes in the whole building.

3.5 Isotope laboratory

3.5.1 General description

The Isotope Laboratory is situated in the western (island) part of the Risø site. It covers an area of approximately 1125 m^2 in one floor, with a basement of 135 m^2 under the northern part, Figure 3.5.1. Inside the building are laboratories, counting rooms, store rooms, offices etc. It is classified as a nuclear facility owing only to the two pneumatic tube systems connecting the laboratory through tubes in the service tunnels with the nuclear reactor, DR 3, Figure 3.5.2. The terminals for the systems are in laboratories 4 and 6 in the southern part of the building. They are used to send samples for neutron irradiation in two facilities in the reactor and receiving them, as well as samples irradiated in other facilities, back to be shipped or processed and shipped from the laboratory. Figure 3.5.2 shows schematically which transport possibilities exist for the two systems.

The work in the laboratory also includes processing of radioactive materials obtained from companies outside Risø, but except for receipt of unmeasured radioactive materials directly from neutron irradiation in a reactor, the work does not differ from the work in other isotope laboratories. As the intention is to continue the work in the building as a general isotope laboratory, the declassification from the status as nuclear facility is only a matter of removing the 300 m of piping from the laboratory to the building seal valves at the DR 3 containment for each of the two systems, the two terminals in lab. 4, the terminal in lab. 6 and the switching station in the basement.

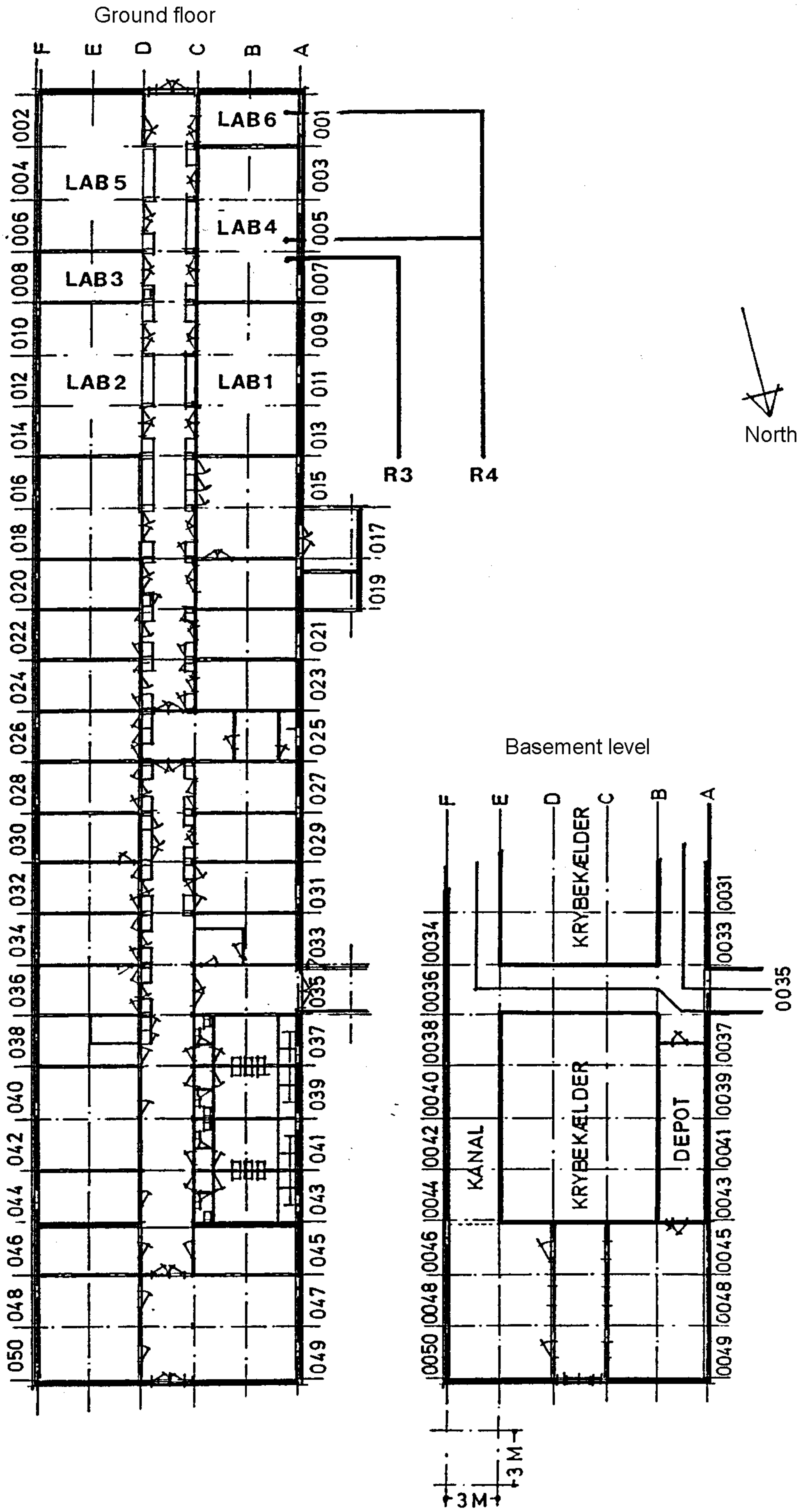

Ground floor
LAB 6
LAB 5
LAB 4
LAB 3
LAB 2
LAB 1
R3
R4
North
Basement level
KRYBEKÆLDER
KANAL
KRYBEKÆLDER
DEPOT
3 M
3 M

Figure 3.5.1 The Isotope Laboratory

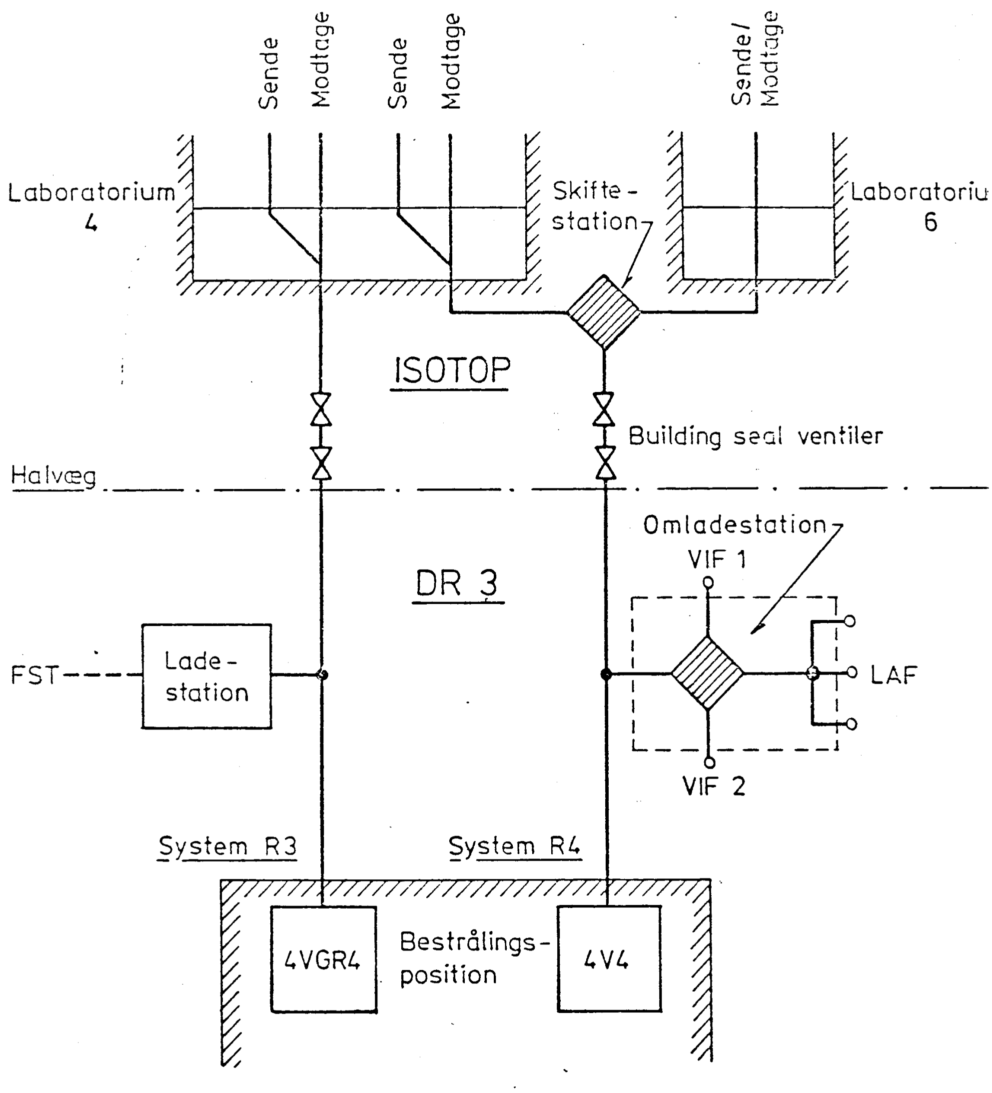


Figure 3.5.2 Piping diagram of pneumatic tube system

3.5.2 Activity inventory

Since the laboratory started its work in 1959 the radionuclide ^{35}S has been and still is produced in laboratory no.2 (module 10 - 14) at the Isotope Laboratory. In the late 60es and 70es the production rate peaked with a maximum of 900 GBq/year, the present rate is some 70 GBq/year. A by-product from this production was the radionuclide ^{36}Cl (half-life 300,000 years) formed from ^{35}Cl in the KCl target material. The amount of ^{36}Cl is around 0.04 % of the ^{35}S produced. The special purified KCl is recovered from the production process and re-used to keep the impurity level low and to build up a higher concentration of ^{36}Cl. The separation of ^{35}S from the KCl is carried out in hydrochloric acid solution. The chloride added amounts to 25-50 % of the chloride from KCl. When recovering the KCl by evaporation of the solution at room temperature HCl now containing ^{36}Cl ($H^{36}Cl$) was continuously released in small amounts into the laboratory. HCl-vapour is highly reactive and was therefore deposited on different surfaces in the laboratory. The deposited activity is fixed at these surfaces and there is no contamination risk involved under normal working operations in the laboratory.

The deposited activity in laboratory no. 2 is unknown but measurements indicate that the total amount adds up to some tens of MBq. Decontamination of laboratory no. 2 is therefore necessary when the Isotope Laboratory is closed down as a nuclear facility.

Lead shielded cells for handling of radionuclides will be slightly contaminated at the inner surfaces and ventilation ducts might be contaminated. The activity inventory will, however, be small.

3.6 Hot Cells

3.6.1 General description

The Hot Cell facility was in active use during the period 1964 - 1989. The six concrete cells have been used for post-irradiation examination of fuel from test reactor fuel pins irradiated in the DR 3 reactor, the Halden reactor in Norway etc. Power reactor fuel pins, including plutonium-enriched pins, from several foreign reactors have been examined. Also HTGR fuel from the English Dragon reactor has been examined. All kinds of non-destructive and destructive physical and chemical examinations have been performed. In addition, various radiotherapy sources - mainly ^{60}Co sources - have been produced from irradiated cobalt pellets in DR 3.

Following a partial decommissioning - mentioned below - a row of six concrete cells now remains from the hot cell facility. Figure 3.6.1 shows a rough sketch of the cells, and Figure 3.6.2 shows their position in the building, which is now being used for other purposes. The dimensions of the cells are given in Table 3.6.1.

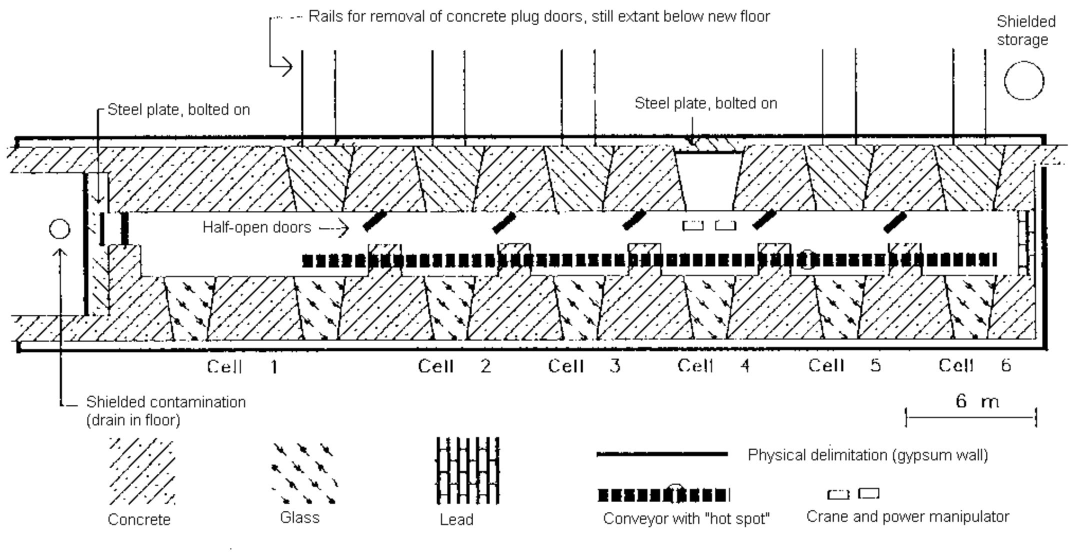


Figure 3.6.1 Cross section of the concrete cells

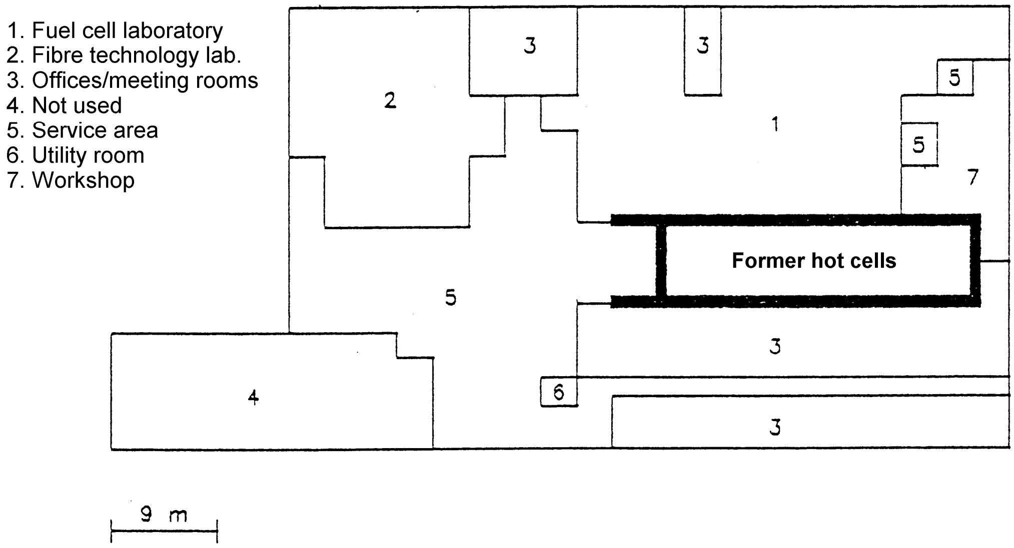


Figure 3.6.2 Location of the cells in the building (ground floor)

Table 3.6.1 Cell dimensions

	Length (m)	Depth (m)	Height (m)	Approximate volume (m^3)
Cell airlock	7.8	5.1	4.1	160
Cell 1	6.5	3.2	4.1	85
Cell 2, 3, 5 and 6	3.3	2.8 - 3.2	4.1	40
Cell 4	3.3	2.8 - 3.2	4.1	44

The interior of the row of cells is lined with a steel box (L=39 m, H=5 m, W=4 m), welded in situ from 8 mm steel plates. Between the interior and the façade of the cell row (also steel plate) there is a steel framework supporting the external façade as well as a number of penetrating pipes, e.g. for master-slave manipulators. The framework also acts as a support for the seven very heavy cell windows (lead-glass). Between the steel box and the external façade concrete is cast. The interior surfaces of the cells are covered by a two-component paint. This surface is intact; but it has had the negative side-effect that much contamination is bound in it. Since the steel box has been welded tight (tightness ~30-50 mm water column) it is expected that the concrete cast around it has not become contaminated, except possibly the cell top where slightly contaminated equipment has been stored. Some contamination may also be found in ventilation- and drain channels below the cell floor.

In each cell a workbench of cast iron is placed. Its surface is porous and holds much contamination.

In the entire extent of the upper part of the cells there is a system of rails where a crane and a power manipulator could run. Both have been left on the rails in cell 4.

The east wall of the row of cells was built from lead bricks with a view to a possible extension. The area outside of the east wall has been piled, possibly with a view to this extension of the facility. This may be utilised in the decommissioning process for the establishment of a house with interlock systems, change- and cleaning facilities etc.

The equipment stored at the cell top comprises parts of the ventilation system and various components with an estimated total volume of 15 m^3 and mass of 3500 kg. In addition six shutters used as shielding between the cells have been retracted into their housings here. Each of these weighs about 2 tonnes and has a volume of about 1 m^3. Furthermore a cable drum is placed (in its housing) on top of cell 4. The housing consists of an external box with a door and an inner casing that contain the cables from power manipulator and cell crane. One of the cables once fell off and onto the workbench and the floor, thereby becoming contaminated. The interior of the cable housing has direct connection to the cells.

Previous decommissioning work

In 1990 a cleanup project was started with the aim to achieve a decommissioning to stage 2 for reactors as defined by the International Atomic Energy Agency (IAEA). The concrete cells were emptied of all fissile material, scientific equipment and general tools and radioactive waste. Thereafter, the cells were cleaned remotely by wiping, hot spot removal by mechanical means and by vacuum cleaning. The internal cell surfaces in two cells were decontaminated by high-pressure water jetting to remove as much loose contamination as possible. A cleanup efficiency of 30 - 40% was achieved from this high-pressure water cleaning. All the master-slave manipulators and part of the contaminated ventilation system for the cells were removed, and the cells were left in a non-ventilated state with connection to the atmosphere through an absolute filter. The cleanup work was completed in 1994.

The cells were sealed and walls were erected around the concrete cell block. The Hot Cell building is now used for offices and laboratories. When cleanup/decommissioning of the cell sarcophagus is to be implemented, parts of or even the whole building should be left empty due to the radiation exposure risk during the cleanup/dismantling process. However, since most of the building is being used by Risø for high-priority research, special actions may be desirable in order to make possible the use of the remaining building during demolition of the hot cells. Under all circumstances planning well ahead (years in advance) is necessary.

As part of the decommissioning work much documentation was collected or produced concerning the facility layout and construction. A large part of this documentation is written in Danish and thus has not been suitable for inclusion in the present description. However, it will serve as a valuable source of information in the planning of the decommissioning work.

3.6.2 Activity inventory

All the concrete cells in the Hot Cell plant have been cleaned remotely (vacuum cleaning and wiping) using the cell master-slave manipulators. In addition, high-pressure water jetting was used to clean cell no. 5 and 6. Based on radiation and contamination measurements the activity content and radionuclide composition in each cell have been determined. 24 TL dosimeters were placed in a matrix in each cell and exposed for an hour. The measured γ-dose rate distribution was used to calculate the activity from the dominating β-/γ-emitting contaminant (^{137}Cs) in the cells assuming a uniform activity distribution over the inner surfaces. Several smear samples over an area of 1 m^2 were taken from tables, walls, and floors and the relative radionuclide distribution determined by α- and γ-spectrometric analyses.

Fission products

Only long-lived fission products are expected in the concrete cells. γ-spectrometric analyses of smear samples taken from the inner surfaces of the concrete cells showed that the fission products ^{137}Cs, ^{134}Cs, ^{154}Eu and ^{152}Eu constitute the major part of the activity. The relative activity of the different radionuclides was measured (1993):

$$^{137}Cs : {}^{134}Cs : {}^{152}Eu : {}^{154}Eu \approx 1 : 0.03 : 0.0006 : 0.02$$

The content of ^{137}Cs in the cells has been determined from the TL-measurements and model calculations. The content of ^{90}Sr in the cells has been estimated from the $^{90}Sr : {}^{137}Cs$-activity ratio of 0.7 in irradiated fuel elements. The value is rather independent of burn-up and also decay, because the two isotopes have similar half-lives. The calculated content of fission product-activity in the cells is shown in Table 3.6.2 (1993).

Table 3.6.2 Activity of ^{137}Cs and ^{90}Sr in the concrete cells based on TL-dosimeter measurements of the γ-dose rate and model calculations.

Radionuclide	Activity in concrete cell 1 - 6 [GBq]						
	Cell 1	Cell 2	Cell 3	Cell 4	Cell 5	Cell 6	Total
^{137}Cs	600	400	700	30	100	20	1,850
^{134}Cs	18	12	21	0.9	3	0.6	56
^{90}Sr	400	300	500	20	70	20	1,310
^{152}Eu	3.6	2.4	4.2	0.2	0.6	0.1	11
^{154}Eu	12	8	14	0.6	2	0.4	37

Actinides

α-spectrometric analyses of smear samples taken from the inner surfaces of the cells showed that the actinides ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{243}Am and ^{244}Cm were present. In irradiated reactor fuel with a burn-up of 33 MWd/kg ^{137}Cs : actinide-activity ratios are approximately:

$$^{137}Cs : {}^{238}Pu : {}^{239}Pu : {}^{240}Pu : {}^{241}Am : {}^{243}Am : {}^{244}Cm$$
$$\approx 1 : 0.03 : 0.003 : 0.005 : 0.001 : 0.0002 : 0.02$$

at the time when the fuel was removed from the reactor. For the fuel investigated in the Hot Cells this occurred 20 to 30 years ago, with the results that a considerable part of the β-emitting ^{241}Pu with a half-life of 14 years now has decayed to ^{241}Am. At the same time much of the ^{244}Cm with a half-life of 18 years has decayed to ^{240}Pu and some decay of ^{238}Pu with a half-life of 90 years has also occurred. Therefore, the following set of ratios is more appropriate at present:

$$^{137}Cs : {}^{238}Pu : {}^{239}Pu : {}^{240}Pu : {}^{241}Am : {}^{243}Am : {}^{244}Cm$$
$$\approx 1 : 0.02 : 0.003 : 0.005 : 0.02 : 0.0002 : 0.005$$

At the same time, some decay of ^{137}Cs with a half-life of 30 years has also taken place, which should result in an increase in the actinide ratios. However, the fact that the mean value for burn-up of fuel investigated in the Hot Cells was 25 rather than 33 MWd/kg mentioned above means a less production of the higher actinide elements ^{240}Pu, ^{241}Pu, Am and Cm as well as ^{238}Pu produced by decay of ^{242}Cm. These two mechanisms will to some degree compensate each other. Improved estimates of the ratios would require detailed computer calculations. A further complication is that the last fuel investigated was of the MOX type and /or had a high burn-up.

The assumed ratios are only valid provided that no separation between the various components has taken place. This is a questionable assumption especially in the cells where partial decontamination has taken place using pressurised water (cells no. 5 and 6). In general, one would expect the caesium to be removed more efficiently than the actinide elements.

The α : ^{137}Cs-activity ratio was determined to be 0.01 - 0.1 (1993) for smear samples taken in the concrete cell no. 5 and 6 which is in reasonable agreement with summation of the above ratios. The (^{234}U + ^{235}U + ^{236}U + ^{238}U) : ^{137}Cs-activity ratio is around $1.5 \cdot 10^{-5}$ and the uranium activity in the cells can therefore be neglected also because the radiotoxicity of the uranium isotopes is much lover than for the other actinides. With the above uncertainties the total α-activity (1993) in the cells can be estimated as shown in Table 3.6.3 (1993).

Table 3.6.3 Activity of actinides in the concrete cells based on the measured ^{137}Cs activity in the cells.

Radionuclide	Activity in concrete cell 1 - 6 [GBq]						
	Cell 1	Cell 2	Cell 3	Cell 4	Cell 5	Cell 6	Total
^{238}Pu	12	8	14	0.6	2.0	0.4	37
^{239}Pu	1.8	1.2	2.1	0.1	0.3	0.07	5.6
^{240}Pu	3.0	2.0	3.5	0.2	0.5	0.1	9.3
^{241}Am	12	8	14	0.6	2.0	0.4	37
^{243}Am	0.1	0.08	0.1	0.01	0.02	0.005	0.3
^{244}Cm	3.0	2.0	3.5	0.2	0.5	0.1	9.3

Activation products

γ-spectrometric analyses of smear samples taken from the inner surfaces of the cells showed that in addition to fission products also ^{60}Co was present. This is due to activated reactor fuel and remains from the production of Co-sources for radiation therapy. The ^{137}Cs : ^{60}Co-activity ratio for irradiated reactor fuel (1993) was measured on smear samples:

$$^{137}\text{Cs} : {}^{60}\text{Co} \approx 1 : 0.03$$

The calculated ^{60}Co-content in the concrete cells is shown in Table 3.6.4 (1993)

Table 3.6.4 Activity of ^{60}Co in the concrete cells based on the measured ^{137}Cs activity in the cells

Radionuclide	Activity in concrete cell 1 - 6 [GBq]						
	Cell 1	Cell 2	Cell 3	Cell 4	Cell 5	Cell 6	Total
^{60}Co	18	12	21	1	3	0.6	55

Small activated Co-pellets (about 1 mm^3) from the production of radiation therapy sources for Danish hospitals are still present in the cells. The specific ^{60}Co-activity at the time of production was about 90 GBq·mm^{-3}. At least one of these pellets is fixed at the conveyor belt system used for transport of small active items between the cells; in 1987 a dose rate of 3 - 4 Sv·h^{-1} was measured close to the pellet when the conveyor was taken out for repair. The activity per Co-pellet is about 3 - 10 GBq today (2000). The dose rate today close to a pellet (5 - 10 cm) will be 0.1 - 1 Sv·h^{-1}.

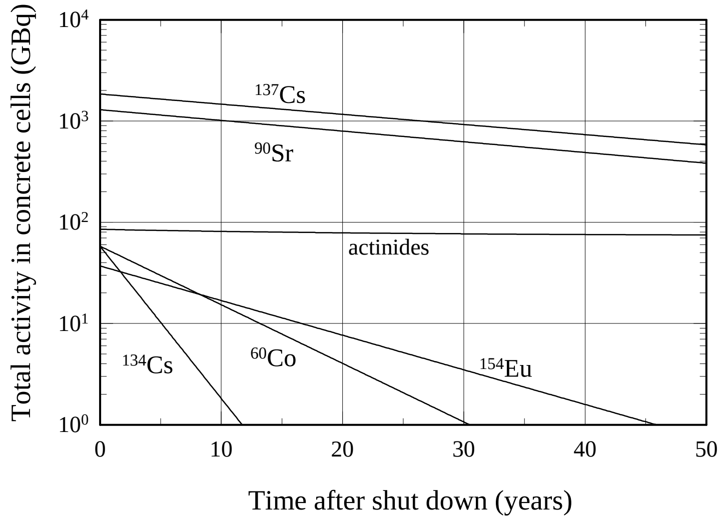


Figure 3.6.3 Total activity in the concrete cell 1 - 6 as a function of time. The cells were shut down in 1993

The activity in the cells will decrease in time due to radioactive decay as shown in Figure 3.6.3 but only the activity of the nuclides ^{60}Co, ^{134}Cs and ^{154}Eu will decrease significantly over the next 20 - 30 years.

The major part of the activity, *i.e.* more than 90% is found in the concrete cell no. 1 - 3. The total activity in the concrete cells (1993) is about 3,000 GBq ^{137}Cs + ^{90}Sr, about 95 GBq actinides, about 60 GBq ^{60}Co (exclusive Co-pellets) and about 100 GBq of ^{134}Cs + ^{154}Eu.

3.7 Waste management plant and storage facilities

3.7.1 General description

The waste management plant is responsible for the collection, conditioning and storage of radioactive waste from the laboratories and the nuclear facilities at Risø and from other Danish users of radioactive materials. The latter is a commercial service provided since 1972 by agreement with the Danish health authority.

No disposal of Danish produced radioactive waste has taken place and the entire collection of waste units produced since 1960 is presently stored in one of the interim storage facilities described below. Decommissioning of the waste storage facilities is primarily a question of providing final disposal for these waste units, while the following decommissioning of the empty facilities should be fairly simple. The process equipment used in treatment of the waste will only be slightly contaminated, but the decommissioning will have to be postponed until there is no more use for the facilities or suitable substitutes have been provided.

After decommissioning of the nuclear facilities at Risø there will still be a need for a treatment system for radioactive waste in Denmark, because radioactive isotopes will continue to be used in medicine, industry and research. Facilities for handling, conditioning and temporary storage of waste from such activities must be available, but could of course be situated elsewhere.

The waste treatment plant is placed on an artificially created pentagon at the northern shore of the Risø peninsula approximately midway between the 'island' with most of the nuclear facilities and the 'land part' of Risø with the low-level and inactive laboratories, etc. see Figure 3.7.1. This position was selected to facilitate treatment of ordinary sewage from the institute, a function, which will have to continue also after the nuclear work is stopped.

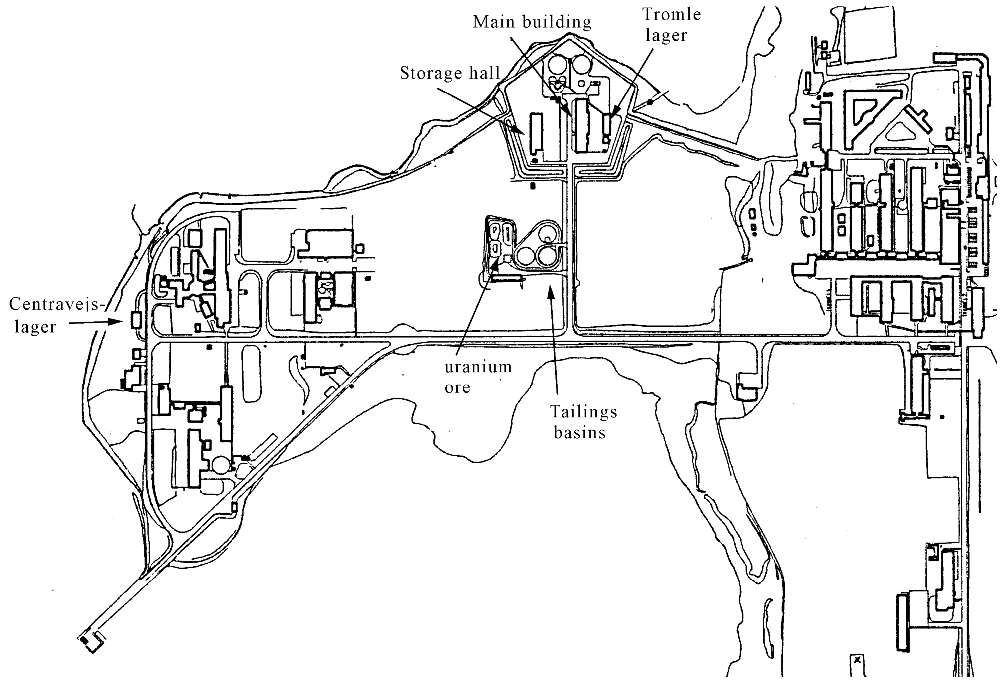


Figure 3.7.1. Facilities belonging to the Waste Management System at Risø

1) <u>The main building</u> 211 (from 1958, about 1200 m^2) contains
- the treatment plant for radioactive waste water,
- a room used for decontamination (mainly of protective clothing)
- laboratories for control analyses and waste characterisation
- offices etc.
- an inactive laundry
- and outside to the north, the treatment plant for inactive sewage

only the two first indents are classified as nuclear facilities (about 300 m^2).

Slightly contaminated wastewater is cleaned at Risø by evaporation using an energy efficient steam recompression <u>evaporator</u> built according to (a long ago expired) Danish patent. The cleaned water is released to Roskilde Fiord while the concentrate with most of the activity is solidified by bituminisation. The layout of the plant is shown in Figure 3.7.2 and the size of the equipment can be estimated from the cross-section of building 211 in Figure 3.7.3. The interior of the equipment is contaminated, but the cell surfaces and floors are mostly clean. The plant is build from stainless steel and is in satisfactory corrosion condition. It will be well suited for processing contaminated water from decommissioning of the other nuclear facilities at Risø.

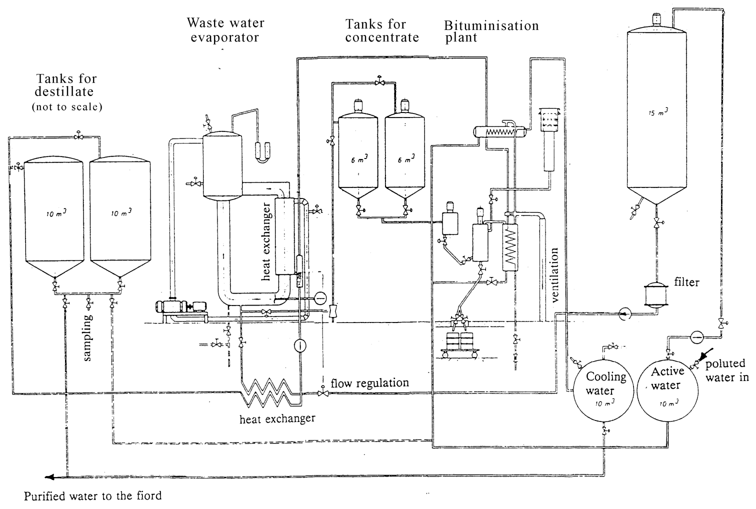


Fig. 3.7.2 Flow sheet for the low-active wastewater purification system showing the evaporator and the bituminisation plant together with the associated tank system.

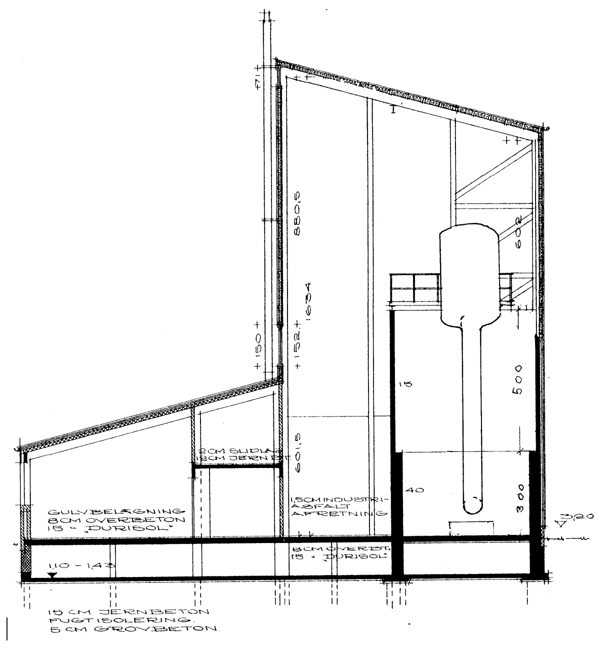


Fig. 3.7.3 Cross-section of the main building of the Waste Management Plant, showing the loop of the wastewater evaporator in its shielding concrete cell.

The <u>decontamination facilities</u> are mainly used for cleaning of contaminated protective clothing, which will also be needed in connection with the decommissioning projects. The existing system is not suitable for large-scale decontamination of components or machinery. The area and the equipment are not or only slightly contaminated.

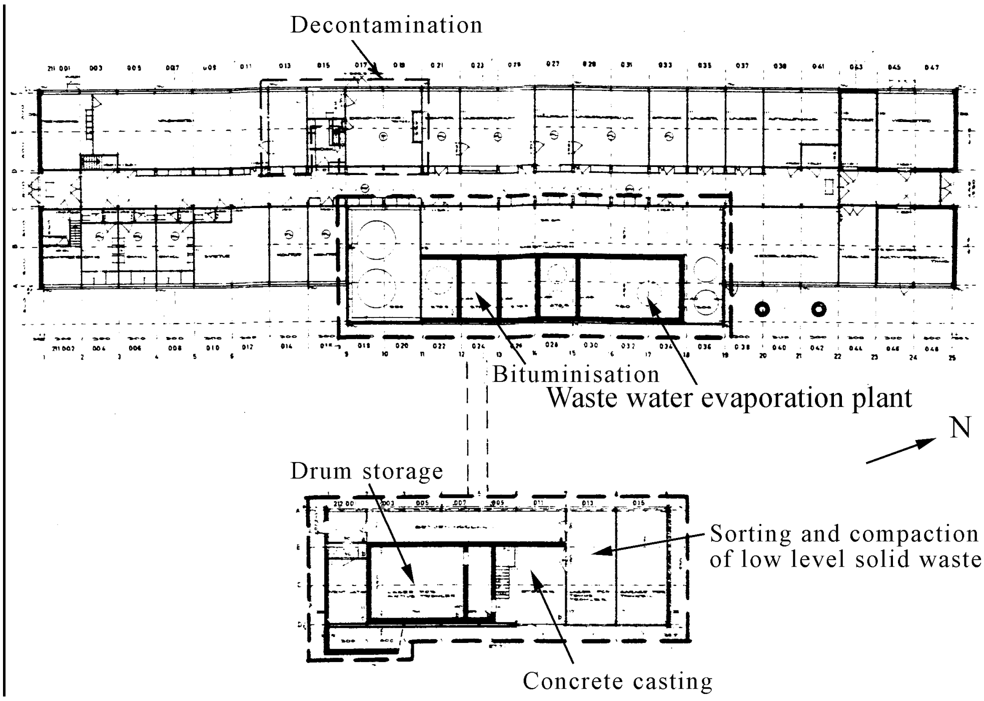


Figure 3.7.4. Plan of the main building of the Waste Management Plant and the associated 'Tromlelager' building connected with the bituminisation cell by an underground passage.

2) Building 212 'Tromlelageret' (from 1958, about 220 m^2) contains
- sorting and compaction equipment for solid waste
- concrete casting facilities
- a shielded storage area for about 80 drums with medium radiation.

The sorting box and low compaction-pressure balling press used for packaging of low-level solid waste is able to handle moderate amounts of low-radiation solids (cloth, paper, wood, glass and metals) which can be accommodated in the Risø standard waste container. It is not suitable for larger items or for a high influx of waste. The box is contaminated on the inside.

The concrete casting facilities are used for preparing the standard containers (a 100 l drum inside a 220 L drum with the annular space filled with cement mortar) and for closing of the containers. Also here the capacity is limited.

The shielded drum storage area was originally intended for waste units with relatively high amounts of short-lived activity. After a suitable period the content could be treated e.g. in the balling press. The store is now mainly filled with drums with long-lived activity. After removal of the drums, the area may be slightly contaminated.

3) Building 244 'Lagerhal for lavaktiv affald' (from 1992, about 780 m^2) is situated west of the main building and contains γ-scanning equipment for measurement of isotope contents in drums. It is mainly used as
- storage facility for up to about 5500 drums with low-level waste.

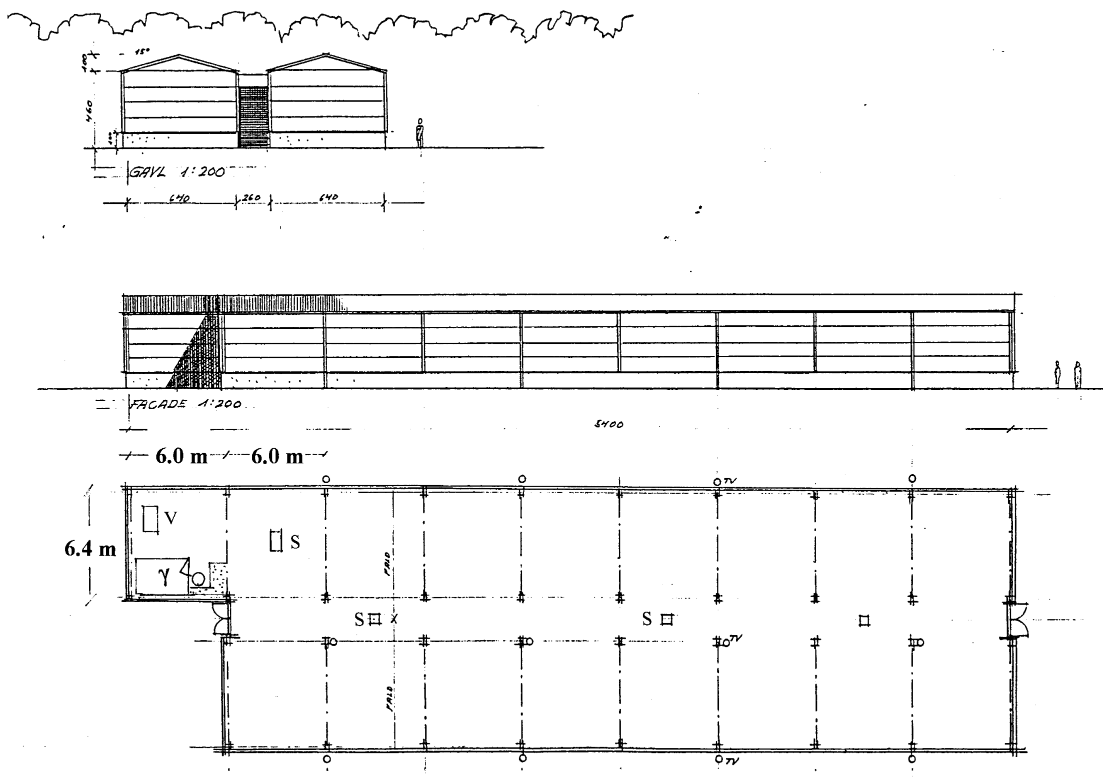


Figure 3.7.5. The storage hall for low-level waste drums (north to the right).

Presently the facility contains about 4700 drums with low-level waste constituting most of the radioactive waste collected at Risø during the past 40 years. A considerable part of the waste units was moved to the storage hall in 1993 to 96 from a previous storage facility "Betonrørslageret" which was decommissioned and released to general use in 1999. Some waste units had to be overpacked on arrival to the new storage hall.

All units were measured using the γ-scanning equipment to establish the contents of ^{137}Cs and ^{60}Co. The scanning facility can be used also for decommissioning waste, but would then have to be moved elsewhere.

The remaining capacity is sufficient for about 10 years continuing production of low-level waste units at the rate typical for later years. The facility is only intended for standard waste units (drums) and cannot provide storage for large irregular components or a large amount of waste units from decommissioning. If needed, the storage hall can be extended by 50% at the north end.

The building is a light thermally unisolated steel construction covered by corrugated aluminium plates. The floor is an about 0.5 m thick concrete slab with a membrane placed about 5 cm below the surface to prevent accidental leakage of contaminated solutions. Some slightly contaminated areas must be expected when the drums are removed.

Elsewhere at Risø some other buildings or facilities also belong to the waste management system:

4) Building 231 ‘Centralvejslageret’ at the end of the central road (from 1968 with later extensions, about 260 m^2, see Figs. 3.7.1 and 3.7.6) contains

- an underground concrete block with holes and cellars for 195 drums,
  90 smaller stainless steel containers with high radiation waste and

15 holes for special containers with e.g. control rods from the DR3.
- 3 larger storage pits mainly for irregular α-contaminated solid waste.

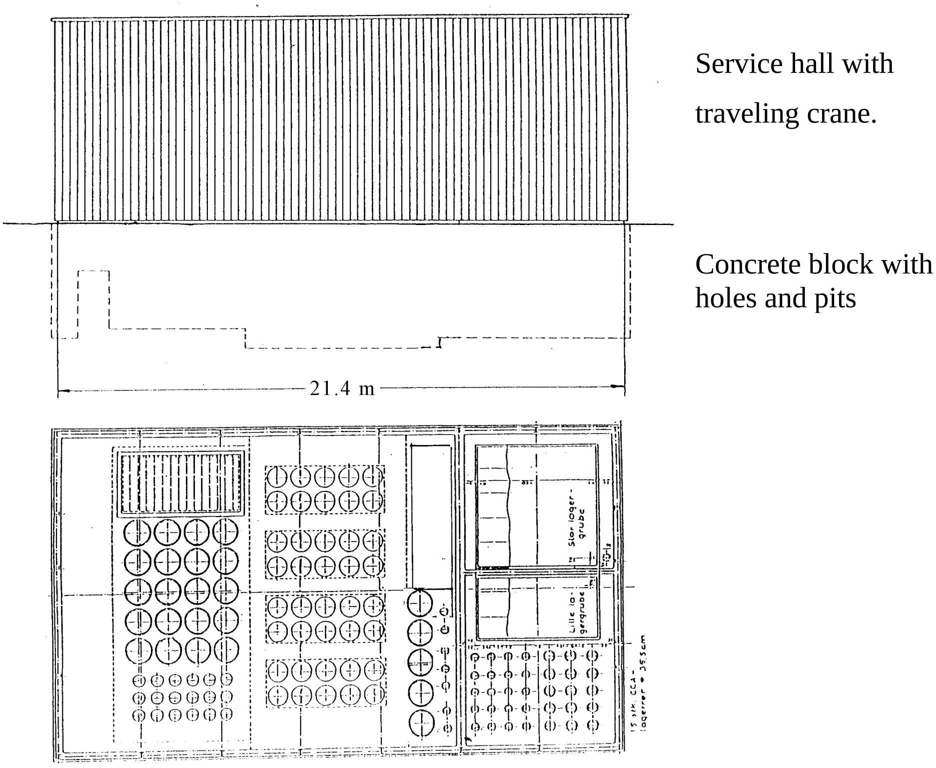


Figure 3.7.6. Facade and ground plan of ‘Centralvejslageret’ for high radiation waste.

The above ground part of the building is a steel construction covered by corrugated aluminium plates. It is a largely empty hall containing an overhead crane used for movement of the heavy lead shielding needed when inserting or removing high-radiation waste units from the storage positions. The floor and upper part of the building is uncontaminated.

The holes and pits in the underground part of the facility are lined with steel plates (~900 m^2) surrounded by concrete. Removable concrete plugs serve as upward shielding. The inside of the holes may be slightly contaminated.

Most of the positions are filled, mainly with waste from hot cell operations and from the partial decommissioning of the hot cells. Some capacity remains in a recently constructed part of the storage facility but the extension cannot accommodate large amounts of waste from decommissioning of the reactors.

Continuing long-term storage of waste in this facility is a management possibility.

5) The <u>‘Tailings basins area’</u> west of the road leading to the treatment plant (from 1982, about 1500 m^2, see Figure 3.7.1 and the aerial photo Fig 3.7.7.) comprises
- two basins with ~1130 t uranium extraction pilot plant tailings (and a third ‘empty’ water filled basin)
- three mounts with the remaining ~3670 t uranium ore from Greenland
- underground tanks for collection of overflowing rainwater from the basins.

Figure 3.7.7. Photo from 1983 of the three tailings basins and the mounts with uranium ores. The waste management plant is seen to the left. The mounts are now somewhat smaller.

The tailings and remaining ore contains only natural radioactivity (uranium and thorium and the associated daughter products) at concentrations about 100 times higher than typical for Danish soil. The tailings are stored under water to avoid emission of radon. Excess rainfall from the basins and the area with ore is collected in underground tanks, controlled for activity and released when the concentration is below specified values.

6) Seven cellars (four in connection with the nuclear facilities and the others in connection with laboratories on the 'land part' of Risø) contain typically:

- two tanks (4 or 8 m^3) for low-level radioactive waste water, which is collected by tank car and transferred to the evaporator described above.
- two tanks (10 or 15 m^3) for so-called cooling water released directly to the sewage system after control for activity.

The tanks are considered part of the Waste Management Plant while the associated pipework under the buildings (in the order of 1000 m totally) is the responsibility of the users.
The three systems at the land part are used only sporadically for activity and are therefore only slightly contaminated. They will probably be taken out of use in 2001, but the actual dismantling may be postponed to avoid interference with the buildings.

The other systems can be used to collect contaminated water during decommissioning of the nuclear facilities and should be kept operable unless other collection facilities are being established. (The DR 2 tanks may not be usable, however, even though they remain in situ, because they have been damaged at an earlier stage).

4 Relevant experience from other countries

Decommissioning of nuclear facilities has already been performed in many countries. For the study reported here it was chosen to achieve knowledge and experience from institutions similar to Risø National Laboratory in other European countries. Visits were paid to three laboratories, as reported below. In addition a literature survey was carried out, in particular with a view to work that had been performed or supported by international organisations.

4.1 Germany

The German Nuclear Act prescribes that operators or owners of nuclear installations reuse or dispose of the nuclear waste produced. Consequently, nuclear facilities have to be decommissioned and dismantled. At the time of writing the present report 65 decommissioning projects were in progress in Germany, covering a variety of types of installation. It has been estimated[3] that the liabilities for all decommissioning projects in Germany add up to 23000 million Euro.

Since the interest of the present study concentrates on research facilities the majority of the information collected for this report is information related to a research centre, the FZK. However, in chapter 15, Literature, references are given to more general information on Germany.

4.1.1 Forschungszentrum Karlsruhe

As part of the project described in the present report a visit was paid to the Forschungszentrum Karlsruhe (FZK - "Research Centre Karlsruhe") in order to hear about the experience gained there. Although larger than Risø, FZK has had similar activities as Risø and has a similar history. Much of the description below is based on information given during this visit, in particular a paper produced by our host, Mr. W. Pfeifer[4].

In the framework of the decommissioning programme at the FZK, five reactors and a reprocessing plant are to be decommissioned. The total budget of this enterprise is about 1900 million Euro. Two of the reactors, Kernkraftwerk Niederaichbach (KKN - "Niederaichbach Nuclear Power Plant") and Heiβdampfreaktor (HDR - "Superheated Steam Reactor"), have already been decommissioned to "green field". These facilities were not located at the FZK premises, but have been part of the programme anyway, because they were experimental or prototype facilities built by the research centre. At the FZK premises the research reactor FR 2 has been brought to a stage of safe storage, with free access to the reactor building, which is now used for exhibitions. Furthermore, the Mehrzweck Forschungsreaktor (MZFR - "Multipurpose Research Reactor"), the Kompakte Natriumgekühlte Kernreaktoranlage (KNK - "Compact Sodium-cooled Nuclear Reactor) and the Wiederaufarbeitungsanlage Karlsruhe (WAK - "Karlsruhe Reprocessing Plant") are being decommissioned now.

The Kernkraftwerk Niederaichbach (KKN) was a pressure tube reactor cooled and moderated by D_2O with an electrical power of 100 MW. It was operated for two years only, from 1972 to 1974. Decommissioning to "green field" was accomplished in 1995.

The Heiβdampfreaktor (HDR) was a boiling water reactor with a thermal power of 100 MW. It operated from October 1969 until April 1971 when it was shut down following a fuel ele-

[3] Schiffer, Klaus, Decommissioning activities in Germany. 7th International Conference & Exhibition on Decommissioning of Nuclear Facilities. London, 30/31 October 2000. IBC UK Conferences Ltd.
[4] Pfeifer, W., Uncertainties in the Costs and Timing of Plant Decommissioning and Waste Disposal. Paper presented at the meeting of the OECD/NEA`s Committee for Technical and Economic Studies on Nuclear Development and the Fuel Cycle (NDC). Paris, 15 June 2000.

ment failure. From 1974 to 1991 the facility was used for reactor safety experiments. Thereafter all components, including the reactor pressure vessel, were dismantled directly and in 1998 the area had been brought to a "green field" state.

The FR 2 was a D_2O moderated and -cooled research reactor, which was taken into operation in 1961 with a thermal power of 12 MW. In 1966 the power was raised to 44 MW. The reactor was shut down in 1981 and has since been brought to a state of safe enclosure of the reactor itself, while all external circuits and facilities have been dismantled. The area inside the reactor building around the biological shield today is used for an exhibition of the FZK's contribution to the development of nuclear technology.

The Mehrzweck Forschungsreaktor (MZFR) was a pressurised heavy water reactor with an electrical power of 60 MW. It was in operation from 1965 to 1984. At present the external circuits have been removed and the dismantling of the reactor internals and moderator tank by remote handling is in progress. The aim is to reach the "green field" state in 2003.

The Kompakte Natriumgekühlte Kernreaktoranlage (KNK) was an experimental fast reactor with an electrical power of 20 MW. It was in operation between 1971 and 1991. At present the fuel and sodium (primary and secondary) has been removed and secondary systems have been dismantled. Remote-controlled dismantling of the reactor tank is planned to start in 2001, and the "green field" state is expected to be reached in 2003.

The Wiederaufarbeitungsanlage Karlsruhe (WAK) was operating as a pilot reprocessing plant from 1971 to 1991. At present the main processing building, the fuel element storage pool, the auxiliary systems and units of the head-end cell have been dismantled. Before decommissioning of the WAK can be completed, however, 80 m^3 of high-level radioactive waste concentrate (HAWC), stored in tanks, has to be solidified. A facility for this is being built and will be taken into operation in 2003. Decommissioning of the WAK is expected to be completed to a "green field" state in 2009. This facility is considered the greatest challenge by the decommissioning organisation.

Due to the fact that a final repository for radioactive waste is not yet available a very large storage facility has been established at the FZK as part of the Central Decontamination Operations Department (HDB)[5].

The decommissioning activities at the FZK are run in a separate division (Geschäftsbereich) in the FZK organisation, parallel to the research area. It has a separate budget, not linked to the research budgets. The funding for decommissioning of the reactors is provided by the Federal Republic of Germany and the State of Baden-Württemberg and is granted annually. For the reprocessing plant a fund of 1000 million Euro has been set up to cover the total process. The latter type of funding is considered preferable because it gives more flexibility to the planning of the work.

The decommissioning division carries out planning of the activities and supervises the work, while most of the actual dismantling work is carried out by external staff.

In the choice between direct and delayed dismantling after a period of safe enclosure the FZK today is in favour of direct dismantling, since suitable remote handling techniques are now available. Safe enclosure is considered relevant only when the costs of conditioning and packaging the disassembled activated components become unacceptably high.

[5] Central Decontamination Operations Department (HDB). Information brochure published by Forschungszentrum Karlsruhe GmbH. 1995.

With respect to the licensing aspects it has become the experience that the dismantling activities should preferably be divided into several steps that are licensed separately. In this way the licensing is less likely to become a delaying factor.

An interesting observation that has been made at the FZK is that the distribution of costs between different categories seems to be very similar for different projects[6]. The cost breakdown for the MZFR is the following:

Remaining operation	50%
Dismantling	~20%
Waste management	~20%
Licensing procedure	~5%
Others	5%

For other projects under execution the proportions vary ±5%. The large share for the activity "Remaining operation" is attributed to the fact that operation staff has to be taken over after shutdown and employed for dismantling activities. This is associated with the disadvantage that the operation organisation cannot be adapted to the dismantling requirements in an optimal way.

4.2 Switzerland

In the preparatory phase of the project reported here a visit was paid to the research centre, Paul Scherrer Institut (PSI), in Villigen, north of Zürich. Although Switzerland has a nuclear power programme, the only information collected so far from the country is the information from PSI.

4.2.1 Paul Scherrer Institut (PSI)

In size the PSI resembles Risø more than the other two research centres visited, FZK and Harwell. The institute has in total around 1200 staff and an annual budget around 250 million Swiss Francs (~165 MEuro ~1250 MDKK), 90% of which comes from the federal government. Two reactors, DIORIT and SAPHIR, are under decommissioning at present. Future dismantling projects comprise a medical irradiation device (PIOTRON), seven large sewage tanks, a pilot incinerator facility, a 72 MeV injector cyclotron and a research reactor (PROTEUS).

The decommissioning organisation is part of the PSI's organisation and comprises a dismantling project for each of the two reactors, a waste management group and a documentation office. It has a staff of about 15, but about 30 people are more or less involved in decommissioning activities and waste management.

The DIORIT reactor was a 30 MW, natural uranium, D_2O moderated research reactor, developed by Swiss industry. It was in operation from 1960 until 1977. A first concept for the dismantling of the reactor was developed and the first phases of dismantling started in 1980. The complete dismantling of the reactor was planned in a two-year study between 1992 and 1994[7]. The last phase of the dismantling, which is now being started, is the dismantling of the reactor block itself. This is planned to be performed from the top down and from the inside out. In doing so, a closed surface is maintained throughout the process, as far as possible. The total process is subdivided into 11 steps. A procedure is chosen that allows modification of the planned work in a step based on experience gained in previous steps.

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[6] Pfeifer, W., Uncertainties in the Costs and Timing of Plant Decommissioning and Waste Disposal.
[7] F. Leibundgut, Decommissioning and dismantling of the DIORIT research reactor. In: Radioactive Waste Management and Environmental Remediation - ASME 1999.

In 1970-72 the reactor tank was exchanged and the old tank put into storage. Therefore, there are two aluminium tanks to dispose of now. In addition there are a number of other aluminium components. In order to reduce the surface area of the resulting waste (to avoid excessive hydrogen production through corrosion) the aluminium parts are re-melted. A reduction of the surface area by a factor of 20 is achieved in this way. A number of remotely operated or automatic tools were developed in order to accomplish the cutting and melting of the aluminium components with a minimum personnel dose.

The bulk of the radioactive waste is steel and cast iron from the thermal shielding of the reactor. The problems with this waste are the high dose rate and the mass, about 24 tons. Furthermore, part of the biological shield (colemanite concrete) is activated and has to be conditioned and disposed of as radioactive waste. The graphite reflector, on the other hand, is not seen as a problem.

The steel, cast iron and concrete material is conditioned for packaging in waste containers by cutting it into suitable pieces by means of diamond-tipped saw blades and diamond wires. In order to fill each waste package most efficiently a combination of steel/iron and concrete is filled into each container.

During the first phases of the decommissioning of DIORIT a total amount of 52 tons of contaminated material was decontaminated, thereby achieving a volume reduction of 50% of the material that had to be disposed of as low level waste.

Leibundgut[7] estimates a total cost of 26 million CHF for decommissioning DIORIT, including the cost of final storage of the waste.

The SAPHIR reactor was a light water moderated and -cooled pool-type reactor operating at a thermal power of 10 MW. Decommissioning is in the first phase and is planned for completion by the end of 2004 at a total cost of 6,5 million CHF, exclusive of the costs of waste disposal.

With respect to the licensing process the two decommissioning projects differ in that permissions for DIORIT are given stepwise by the authorities, whereas for SAPHIR an overall permission is given because a quality management system has been implemented.

With respect to radiation protection it was underlined that this aspect had been part of the planning of the decommissioning projects from the very beginning. The actual doses seem to fit quite well to the values foreseen in the planning. The collective dose to the DIORIT personnel in 1999 was 17 person-mSv, and a total of 150 person-mSv is foreseen for the entire DIORIT decommissioning.

Advice was given to be aware of the fact that behind the operational radiation protection there are analytical services, which also need resources. Furthermore, it had been the experience at PSI that more radiation protection personnel are needed during decommissioning than during operation.

The following procedures for treating waste are applied at the PSI

- sorting
- incineration
- solidification with cement (homogeneously miscible waste)
- imbedding in concrete (irregular solid waste)
- compaction
- gas tight welding (volatile, air polluting (^{3}H, ^{14}C, ^{226}Ra))

Facilities for operational waste are:
- an incinerator - pilot facility (50 t/y)
- a conditioning facility: sorting, compaction by a 100 t press,

Facilities for decommissioning waste from research reactors are:
- an Al-dismantling and -melting facility
- an etching facility
- dismantling tools for cutting of structural materials (thermal and biological shielding)

Furthermore, a stationary facility for clearance measurements and an interim storage facility for 4900 m^3 of waste exist.

With respect to clearance of dismantled components, unconditional clearance can be permitted under consideration of the following parameters:
- bulk activity averaged over 300 kg
- dose rate at 10 cm distance (<100 nSv/h)
- surface contamination averaged over 100 cm^2 (<1 Bq/cm^2)

Similar to Risø's DR 2 the SAPHIR reactor had beryllium reflector elements in the form of BeO- and Be metal elements. These elements contain both ^3H and ^{60}Co activity, which posed a handling problem, in addition to beryllium being poisonous. The reflector elements have been placed into steel cylinders that were welded tight and, for the present, stored in MOSAIK containers. The cost of this particular decommissioning operation was 150-200.000 CHF.

4.3 United Kingdom

Within the UK there are a number of utilities/organisations who own/operate a variety of nuclear installations[8]. These are Magnox Electric (ME), Nuclear Electric (NE), Scottish Nuclear (SNL), British Nuclear Fuels (BNFL), the United Kingdom Atomic Energy Authority (UKAEA) and the Atomic Weapons Establishment (AWE). With the exception of NE, each has one or more installations which have been shut down and which are being decommissioned. ME and SNL are decommissioning large power reactors, BNFL are decommissioning plant generally related to fuel fabrication and reprocessing, the UKAEA are decommissioning mainly prototype and small materials testing reactors and AWE are mainly decommissioning gloveboxes and fume cupboards associated with plutonium and uranium handling.

In the present report attention will be concentrated at the UKAEA, since the facilities owned or operated by the UKAEA are those most similar to Risø's facilities.

The UKAEA has adopted the following policy with regard to decommissioning :
- Redundant facilities to be safe and compliant with legislation
- Decommission for safety, environmental or planning requirements
- Defer decommissioning until justified by safety, environmental or planning requirements.
- Choose options that minimise immediate costs.
- Return site to a safe condition that does not require care or monitoring.
- To minimise the cost net present value of the above.

Decommissioning and **Ra**dioactive **W**aste **M**anagement **Op**eration**s** (DRAWMOPS) is the programme for the decommissioning of redundant radioactive facilities, the transport of radi-

[8] M.W.Davies: A review of the situation of decommissioning of nuclear installations in Europe. European Commission, Directorate-General Environment, Nuclear Safety and Civil Protection. Nuclear science and technology. EUR 17622 EN. 1997.

oactive materials, the reprocessing of fuels and residues, the management through to disposal of all radioactive wastes and any necessary research and development. The programme is managed by UKAEA Government Division (GD) and its current expenditure is £150~200 million/annum. In DRAWMOPS work the relevant UKAEA Government Division components are DRAWMOPS Directorate and Nuclear Site Operations.

The DRAWMOPS Directorate is responsible for:
- Agreeing the DRAWMOPS programme with the Department of Trade & Industry (DTI)
- Long term planning
- Procurement policy
- Decommissioning management at Winfrith, Culham, Harwell, Windscale, Springfields etc.
- R&D management
- Windscale site management

Nuclear Site Operations is responsible for:
- Radioactive waste operations in the UKAEA
- Management of radioactive materials transport in the UKAEA
- Reprocessing of fuels and residues at Dounreay
- Dounreay site management
- Operational radioactive facilities management in UKAEA

Drawmops Directorate, Government Division, UKAEA have the following nuclear liabilities:
- Reactors
- Post Irradiation Examination facilities
- Laboratories,
- Reprocessing plant
- Waste stores, Effluent plant
- Waste disposal sites
- Joint European Torus (JET).

In total they number approximately 150 if all the buildings that comprise an installation are counted, for example the Prototype Fast Reactor (PFR) has 10 buildings and the Dounreay Fast Reactor (DFR) 14 buildings. JET will be a significantly different liability in that it has many new materials, the staff are multi-national and will be dispersed across Europe when the project terminates. In anticipation of this GD are already maintaining records on-site in collaboration with JET personnel.

The following text giving an overview of the decommissioning pattern for UKAEA facilities has been taken from the UKAEA's homepage on the Internet[9].

General Decommissioning Pattern for UKAEA's Facilities

Decommissioning of nuclear facilities can normally be divided into three implementation stages, which may be separated by extended periods of relative dormancy. These are:
- Initial Decommissioning (often referred to as Stage 1). The objective is to remove mobile radioactive items (e.g. fuel, coolant, process materials, rigs) to reduce the risk from the plant and leave it in a safe condition where further decommissioning can be scheduled optimally.
- Surveillance and Maintenance (S&M). During this phase the plant is maintained in a safe condition.
- Dismantling (Stage 2). During this phase the radioactive plant is dismantled and most of the remaining radioactive materials are removed and packaged for disposal.

[9] http://www.ukaea.org.uk/dindex.htm

- Care and Maintenance (C&M). The building structure is maintained under a managed regime until demolition.
- Demolition (Stage 3). The building structure is demolished, normally using conventional techniques.

Safety and economic considerations generally require at least some work to be undertaken immediately after closure. However beyond this the scope of work undertaken at each stage, and the length of time between stages, is determined on a case-by-case basis. The following general patterns emerge for UKAEA's facilities:

Reactors
Reactors are normally defuelled and the coolant removed immediately after closure. This removes most of the radioactivity and reduces substantially the hazard presented by the facility. The remaining activity is normally locked up in the structure in the form of activation products. Delaying the later stages of decommissioning allows the radiation levels to fall as a result of radioactive decay, and as reactor structures are normally robust they suffer little over an extended period of surveillance and maintenance.

Plutonium Plants
In the case of plants which have been used for working with plutonium (e.g. fuel fabrication plants) there are normally benefits to be gained from prompt decommissioning, which also minimises the dose-rate from the ingrowth of americium and deterioration of plant. For these reasons it is normal for process plant to be removed, glove boxes dismantled and vessels washed out during Stage 1. The extent of the period before subsequent stages of decommissioning will normally be dictated by the integrity of the building structure.

Caves and Cells
Caves and cells (e.g. Post-Irradiation Examination (PIE) facilities) are robust concrete structures, which provide a high degree of containment. Mobile activity is normally removed from caves and cells immediately after closure in order to reduce the hazard presented by the facility. The hazard presented by these structures after the removal of mobile activity is normally low, so later stages of decommissioning can be safely deferred until justified on economic grounds.

Other Plants
Other process plants, including waste treatment plants and waste stores, will undergo stage 1 decommissioning immediately following closure. The time scales for the later stages of decommissioning will be determined on a case-by-case basis, taking account of the overall site strategy.

4.3.1 UKAEA Harwell

In the preparatory phase of the project reported here a visit was made to the UKAEA's headquarters at Harwell in order to learn from the experience gained there and to see the facilities. In particular the two reactors DIDO and PLUTO are of interest to Risø because they are of the same type as Risø's largest facility, DR 3.

The decommissioning target for the Harwell site is to bring all facilities to a "green field" state and convert the area to a business centre. There already are tenants at part of the area, but the final removal of the last radioactive facilities lies many years into the future. For instance, the final demolition of PLUTO and DIDO is not foreseen to take place until 2040. At that time the two reactors will have been closed for 50 years, and the ^{60}Co activity will have decreased by almost a factor of 1000. After their closure in 1990 they were brought to phase 2 of decommissioning, all external systems being removed, and since 1995 they have been in the Care and Maintenance phase. The reactor tanks and Internal Storage Blocks are being used as intermediate storage for a number of activated test rigs.

One reactor, LIDO, has been completely decommissioned and the area is now "green field". Two other small reactors, BEPO and GLEEP have been brought to phase 2. The final stage of decommissioning of GLEEP awaits the documentation that the activity in the graphite is below 400 Bq/kg so that the graphite can be cleared. BEPO still contains high levels of ^{60}Co activity, and final decommissioning is foreseen for year 2060.

One of the first decommissioning projects that had been carried through to "green field" was presented by Ed Abel[10]. The facility was the seven storey Chemical Engineering Building, which had housed a large variety of experimental and production-scale equipment for the exploration of all aspects of the nuclear fuel cycle.

Due to the fact that a final repository for intermediate level waste (ILW) is not available - and is not foreseen to be so for many years - an intermediate ILW conditioning- and storage facility has been built at Harwell. The facility is dimensioned for 2000 500 litre drums and has cost GBP 53 million. Low level waste (LLW) is being sent to the repository at Drigg.

The UKAEA plans the individual decommissioning projects and produces bidding specifications for the work, which external companies can bid for. Thus, normally it will not be UKAEA staff performing the demolition; but UKAEA staff will oversee the work and verify that it is performed according to specifications.

The UKAEA has developed a number of computer tools for the planning and supervision of decommissioning projects, some of which could be useful for the Danish decommissioning organisation, as well. One of these, PRICE, has already been evaluated and used in connection with the present project, and it has been decided to acquire it for further use. PRICE is described in some detail in chapter 5.3.

Another tool, the Strategic Planning System (SPS) is being used to facilitate planning and optimisation of the DRAWMOPS programme. This system contains key data for each facility and is used to enable alternative phasing and 'what-if' scenarios to be examined for a whole site's programme. This allows the implications of different decommissioning timings on waste management and funding requirements to be better understood. For example, SPS has been effectively used to provide a rapid assessment of the implications for UKAEA of the delay in the availability in a repository for ILW. The Risø project group has had the opportunity to have SPS demonstrated, and it is found that also this tool can be useful in the planning of the decommissioning of Risø's facilities.

Furthermore, the UKAEA has developed a tool for the undertaking of a Care & Maintenance programme, which seems to be particularly suited in the cases where one or more facilities is placed in safe storage for an extended period of time.

Among the many pieces of good advice received at Harwell and the other two sites visited during the project, the following points, raised by Bob Simpson about Decommissioning experience, project management and project risk, could be mentioned:

- Prepare and plan a project in details (possibly by the use of full-size mock-ups).
- Educate contractors, who are going to carry out the demolition, in the handling of radioactive materials.
- Projects don't always proceed as planned.
- Concerning risk management: expect the unexpected; situations will always arise that have not been anticipated in the planning.

[10] E. Abel and C. Hamblin, Seven storeys of decommissioning – the complete decommissioning of a chemical engineering building. Decom '98, December 1998, London.

4.4 International organisations

In the IAEA as well as in the OECD/NEA and the EC much work has been done in order to collect information about decommissioning work that has been performed and to provide guidance and tools for the planning and performance of future decommissioning. A large number of reports and papers have been issued on the subject, and in chapter 14 at the end of this report a number of the more recent publications are mentioned.

Of particular use to the present study became the collective work of the OECD/NEA, the IAEA and the EC, resulting in a proposal for a standardised list of items to be considered when assessing the costs of decommissioning operations[11]. This list was used directly for estimating the decommissioning work at DR 3, and it was kept in mind when making estimates for the other facilities.

[11] A proposed standardised list of items for costing purposes in the decommissioning of nuclear installations. Interim technical document. OECD/NEA 1999.

5 The approach to cost assessment

The facilities at Risø National Laboratory vary much in size and complexity. Therefore, the approach to cost assessment has been different for different facilities. Although a standardised approach has its advantages it has not been considered sensible to use the same method for e.g. DR 3 and the fuel fabrication laboratory.

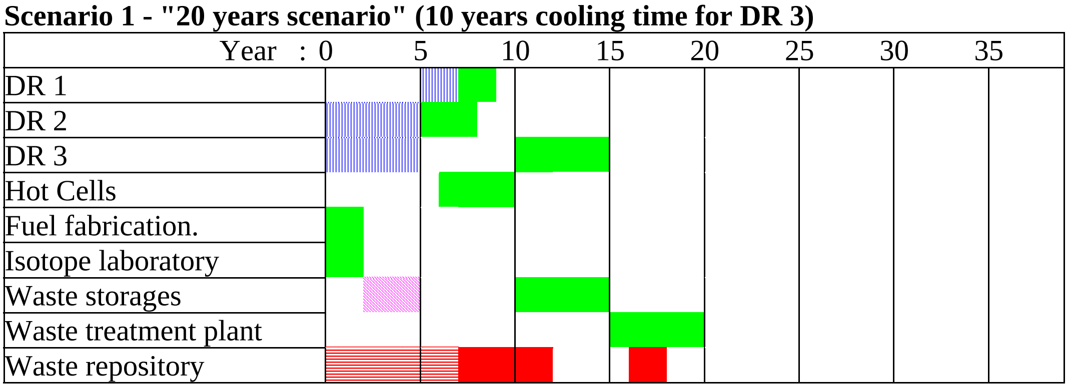
Basically, for each facility the decommissioning tasks and their associated need for manpower and other expenses have been identified and the costs assessed in fixed prices ("Year 2000 prices"). Those tasks where changes in radiation levels result in different needs for protection between the scenarios, described below, have been marked during the task identification, so that differences in cost estimates arising from this reason can be identified. Similarly, differences in costs arising from the different demands to care and maintenance between the scenarios have been identified.

5.1 Scenarios considered

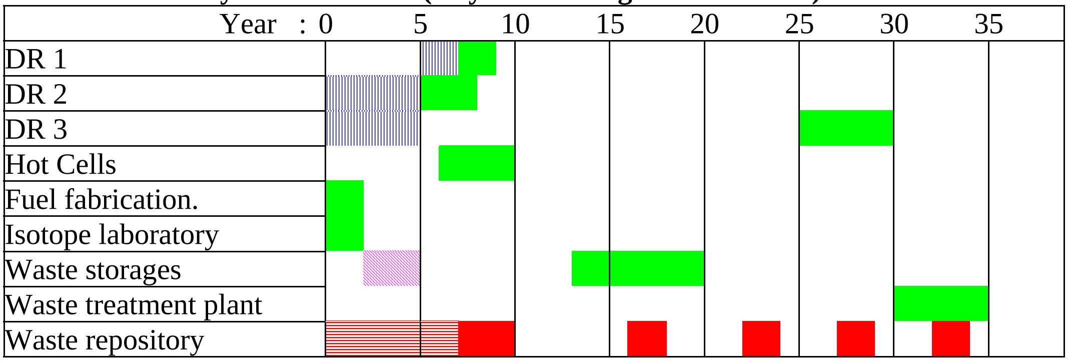
Prior to the study reported here, a preliminary study had been carried out where three different temporal scenarios had been considered. In the present study these scenarios have been adopted with some modifications. The deciding difference between the scenarios is the cooling time foreseen for reactor DR 3 from termination of operation to final dismantling. Cooling times of 10, 25 and 40 years are considered. The total duration of the scenarios is estimated to be 20, 35 and 50 years, respectively. The three scenarios are sketched below.

Construction of the final repository for radioactive waste is foreseen to take place in stages as the demand arises. The stages shown in the figures below should be taken as indicative only and not as an actual plan for the repository - which is outside the scope of the present study.

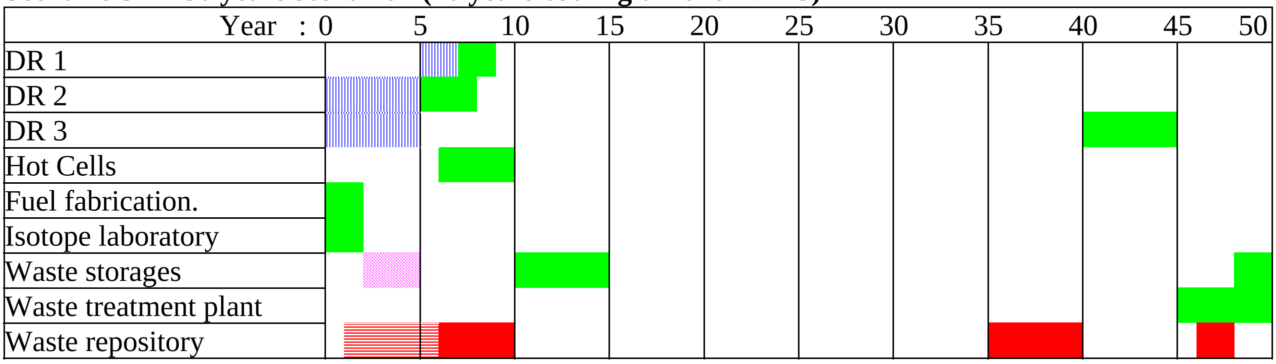
Scenario 1 - "20 years scenario" (10 years cooling time for DR 3)



Scenario 2 - "35 years scenario" (25 years cooling time for DR 3)



Scenario 3 - "50 years scenario" (40 years cooling time for DR 3)



: Dismantling of external circuits etc.
: Final dismantling of reactor block etc.
: Establishment of intermediate storage facility and/or handling facility
: Identification of a disposal site and acceptance and licensing of the construction
: Construction of waste repository (in several stages)

In all three scenarios the two small facilities, the fuel fabrication laboratory and the Isotope laboratory, are decommissioned first. Both are considered as being only relatively lightly contaminated, and the buildings can be used for other purposes. As a matter of fact, the decommissioning of these facilities may have been carried through at the time of issue of the present report. Furthermore, it is assumed that Hot Cells, DR 1 and DR 2 are decommissioned during the first ten years in all scenarios. The transfer of waste from the storages at Risø to the final repository can - more or less - be carried out at any time after the repository has been constructed. Irrespective of the scenario, after the end some activity will still take place either at the waste treatment plant or at the repository, receiving radioactive waste from other activities than Risø's.

<u>In scenario 1</u> both the dismantling of external circuits and the final dismantling of facilities is compressed in time. A more detailed planning may show that the whole process can be carried through in a shorter time than the 20 years foreseen here. In this scenario in particular it may turn out to be necessary to establish an intermediate storage facility for waste arising from the dismantling of active parts. The necessity of this will depend on the time required for selecting a site for the final repository and going through the safety case for this site.

In scenario 2 there is not much activity during the years 11 to 25, apart from a limited activity at the waste treatment plant, the maintenance of the safe store of DR 3 and possible transfer of stored waste to the final repository.

Scenario 3 foresees a very "silent" period from year 15 to year 40 with a limited activity at the waste treatment plant and the maintenance of the safe store of DR 3.

For both scenario 2 and scenario 3 it is foreseen that foreign staff must carry out the final stages of the decommissioning, since people with the necessary knowledge will no longer be available in Denmark. However, it will probably be possible to maintain sufficient knowledge to carry out the inspections of the mothballed facilities that have to be made during the dormancy period.

5.2 General assumptions

As mentioned previously, the establishment of the final repository is outside the scope of the study reported here. Furthermore, as a consequence of not knowing the location of this repository, no attempt has been made to assess the costs of transporting the waste from Risø to the repository.

5.3 Method used for cost assessment

As the facilities are different with respect to complexity, the assessment of labour and cost of decommissioning has been approached differently. For some facilities, such as the Isotope Laboratory, the necessary work could easily be identified, whereas for others a systematic approach was necessary. In particular for DR 3 a standard list of costing items[12] was used as a template for specifying the costs of decommissioning operations. This list has been established in collaboration between the OECD/NEA, the IAEA and the EC. It is aimed at nuclear power plants, but most of the items listed are valid for a research reactor, as well. Also for other facilities than DR 3 the list has been used as a checklist.

For each of the items addressed the required labour effort was estimated - either by Risø staff, where we felt we had sufficient insight, or with the help of consultants or the PRICE programme, described below. A standard rate of 231 DKK/hour was used to calculate the labour cost. This cost was obtained by calculating a suitable average of the costs of the staff categories foreseen for the decommissioning organisation.

For DR 3 the costs were entered into an Excel sheet, based on the costing items in the above-mentioned standard list. For DR 1, DR 2 and the Hot Cell facility decommissioning operations were identified by Risø staff and PRICE was used to calculate the cost.

One point where we have deviated from the list is in the assessment of the health physics assistance needed. Here the list prescribes the specification of health physics effort for each task. However, we have found that the necessary health physics staff and the required equipment can be assessed on an overall basis, taking into consideration more broadly the tasks that are to be performed.

PRICE
The PRICE programme has been developed by the UKAEA and is being used by a number of institutions in other countries, as well. During the project Risø was given the opportunity to have PRICE for evaluation and the programme was found very suitable for our purpose, so that Risø decided to buy the programme.

PRICE incorporates:

[12] A proposed standardised list of items for costing purposes in the decommissioning of nuclear installations. Interim technical document. OECD/NEA 1999.

- a standard Work Breakdown Structure (WBS)
- a methodology for mensuration of component quantities
- a classification system which relates to the physical complexity of the task ("Complexity" classification)
- a classification system which relates to the radiological condition and the level of radiological protection required ("Task" classification)

In PRICE a facility is broken down into simple building blocks or "Components". For each component data is stored on the resources (man-hours) required to remove unit quantity of that component. This is termed the "Norm", which varies depending on the "Complexity" and "Task" classification attributed to the component. Components can have up to five "Complexity" classifications and three "Task" classifications and thus any one component can have up to 15 "Norm" values.

Each of the standard components is sub-divided into a range of five complexity ratings ranging from "Complexity 1" for relatively simple to "Complexity 5" for the most complex.

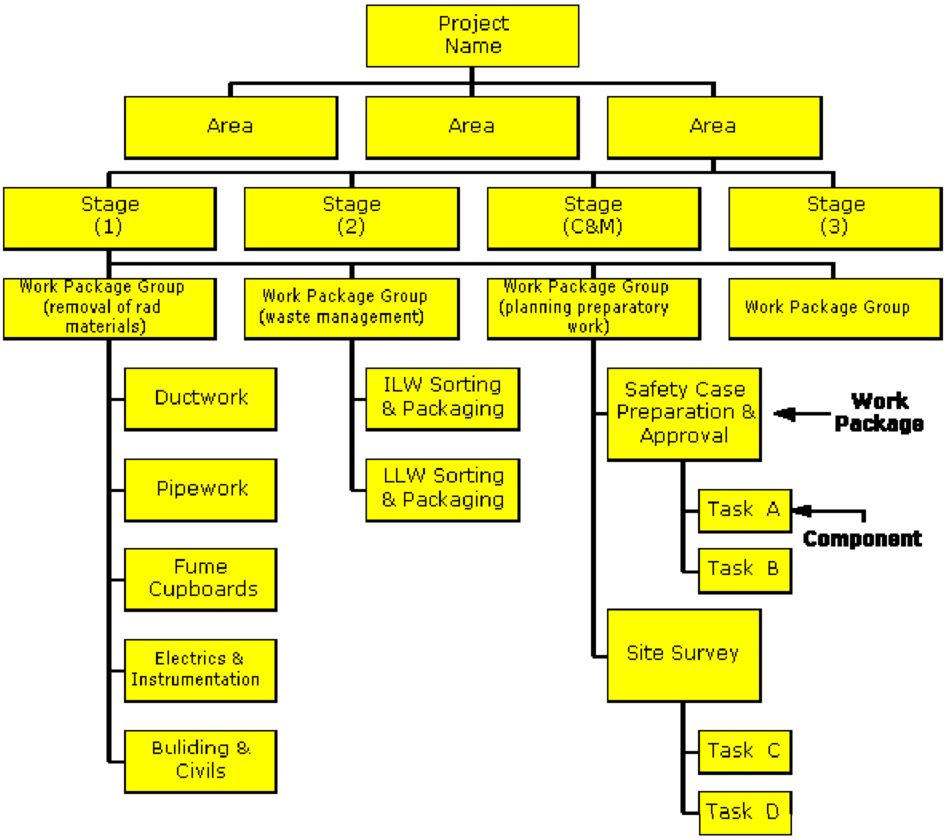
The Task classification provides a means of taking into account the degree of radiological protection required when dealing with the standard components. There are three available Task classifications as follows:

- Task R - "Remote" Defined as operations where operatives at the work face use manipulators, robotics, hot cells etc.
- Task C - "Complex protection" Defined as operations where operatives at the work face must wear pressurised suits.
- Task M - "Minimum protection" Defined as operations where the protection of operatives at the work face necessitates, at the most, the wearing of ori-nasal masks.

A single aggregated man-hour rate or "Unit Rate" for a typical mixed grade team, together with tools and plant, is applied to all components. The system does however allow the user to add a unique "user defined cost" to a task.

The overall cost estimate is produced by summating the individual component costs plus additional sums for items which cannot be treated in this way i.e. capital cost items such as RH equipment, change room facilities, waste packaging facilities etc.

PRICE offers a hierarchical approach that can be used to identify costs in key areas and also those associated with identified "stages" throughout a project lifetime. The hierarchical structure or Work Breakdown Structure used by PRICE is shown below.



PRICE Work Breakdown Structure

5.4 Limitations of the study

It should be underlined that the study reported here is the first attempt to go into detail in the assessment of costs of the operations to be performed when decommissioning Risø's nuclear facilities. Therefore, there are many tasks for which no prior experience exists concerning the manpower needed. As far as possible, experience from other countries has been taken as a guideline; but it must be anticipated that the cost estimates given here will change as experience grows and the study can go into more detail.

The study has focused on estimating the total labour effort to be put into performing the various tasks without going into detail concerning the size of the staff needed at a given time or during a given period to perform the work. This question, of course, will be an important part of the planning to be carried out by the decommissioning organisation.

6 Detailed description of the decommissioning work to be done at each facility

In this chapter the work to be performed at each facility is described in the detail possible at the present stage, and an estimate of the costs is produced. In section 6.9 the costs of some general activities, such as health physics are estimated. These are activities, which we have found can be better assessed on an overall basis, taking into consideration more broadly the tasks that are to be performed at all facilities. Likewise, some general procurements have been included in that section.

6.1 Decommissioning DR 1

6.1.1 Operations to be performed

The first decommissioning task to be performed at DR 1 is to drain the fuel solution into criticality safe bottles. After that a preliminary decontamination can be made by flushing water through the primary system. This water has to be collected under controlled conditions.

The following construction parts can then be removed:

1. Recombiner and belonging pipe circuit
2. Control rods
3. Core tank
4. Graphite blocks from reflector
5. Reflector tank
6. Remaining pipe circuit
7. Lead shield
8. Biological shield

It might be easier to remove the core tank and reflector graphite in one single operation by lifting the whole reflector tank out of the biological shield.

When all the active items have been removed the building can be demolished and the area can obtain status as ‘green field’.

6.1.2 Necessary manpower

The assessment of the costs of decommissioning the active parts of DR 1 has been made by means of the PRICE programme. The database behind that programme holds information about the number of man-hours needed for the operations considered, but these figures are not output directly from the programme, only the resulting labour cost is given (cf. Appendix 1).

The manpower needs are not influenced by any dose constraints because the activity levels in the different systems in DR 1 are small. The different decommissioning processes like cutting free the reactor tank and removing the activated part of the piping circuit will alone determine the manpower resources needed as well as the necessary time period for the decommissioning tasks.

In the decommissioning phase at least one health physics assistant is needed full time for surveillance of the working environments and measurements of all demolished parts, rubbles and scrap that are to be cleared as non-active waste. A part time health physicist is needed for radiation advice and for preparation of the documentation of measurement results and evaluation of the health physics consequences of any release of low level radioactive waste.

6.1.3 Special equipment needed

No special equipment for dismantling the components is foreseen. If the reflector tank with core tank and graphite blocks is to be lifted in one operation the crane in the reactor hall can be used. Special γ-spectrometric equipment is needed for the comprehensive measurement programmes necessary to document the residual activity levels in all items.

6.1.4 Radiation doses to be expected

When the fuel solution, the control rods, the recombiner and the core tank have been removed it is assumed that the remaining components can be handled in free air without causing significant doses to the personnel involved.

6.1.5 Other safety related aspects

The top of the reactor block is only 3 m above the floor level so the consequences of personnel falling down is considered to be limited. However, the risks depend on the method selected for the dismantling, so it is difficult to make general statements about the risks.

A possible stress factor caused by high radiation levels is not foreseen.

The reactor building is a tall steel structure with an inside crane and its removal will involve some risks, in particular of fall-down accidents.

6.1.6 Amounts of radioactive waste produced and its treatment

The amount of waste after decommissioning DR 1 is dominated by the concrete, which constitutes about 60 m^3. If, contrary to expectation, the concrete cannot comply with the clearance level recommended by EU, the determining activity will be from ^{60}Co with a half-life of 5 years.

The graphite amount comprises about 2 m^3 and the remaining components mainly steel about 600 kg.

6.1.7 Documentation to be produced

All activity and radiation measurements, including γ-spectrometric analyses on single items, scrap and rubble etc., must be documented rather detailed for the purpose of releasing as much as possible of the items with only minor content of activity. In addition to that, it is necessary to make detailed scenario calculations of the health physics consequences of any conditional or unconditional releases of low-level radioactive waste as non-radioactive waste.

6.1.8 Assessment of the costs

The assessment of the costs for decommissioning DR 1 is based on the PRICE calculation, shown in Appendix 1, except for demolition of the reactor building, which is based on an estimate from the Danish consultant company DEMEX, also shown in Appendix 1. The total cost is about 7 mill. DDK.

This cost estimate does not take into account the expenses for possible reprocessing and final storage of the 16 litres of high-radioactive fuel solution.

6.1.9 Uncertainties in the assessments

The main expense in the decommission costs of DR 1 is the demolition of the biological concrete shield. If the concrete can be considered as non-radioactive waste the total cost can be reduced substantially.

However, the cost of final disposal of the irradiated fuel solution can easily double the total DR 1 decommissioning costs.

6.2 Decommissioning DR 2

6.2.1 Operations to be performed

After decommissioning of a nuclear facility the condition 'green field' is established when all major active items have been removed. Only minor activity is left on-site that might cause radiation doses less or equal to the dose constraint set for public exposure (critical group) during operation of the facility (e.g. 0.1 - 0.3 $mSv \cdot a^{-1}$). The following items and construction parts will contain more or less activity and most of it would have to be removed as radioactive waste:

(1) beam plugs, guide tubes and S-tubes
(2) beryllium reflector elements
(3) grid plate at the bottom of the reactor tank
(4) the aluminium reactor tank
(5) the lead shield around the reactor tank
(6) the graphite in the thermal column, the igloo, the concrete inner door and the front door
(7) some tens of centimetres of the inner surface of the concrete shield, mainly from the heavy concrete
(8) the primary cooling system (heat exchangers, hold-up tank, tubes etc.)
(9) the underground tanks placed outside the reactor building

The 'green field' status is not necessarily a complete removal of all items and buildings. If all active parts have been removed from the reactor building and cellars 'green field' can include the reactor hall.

6.2.2 Necessary manpower

The assessment of the costs of decommissioning DR 2 has been made by means of the PRICE programme. The database behind that programme holds information about the number of man-hours needed for the operations considered, but these figures are not output directly from the programme, only the resulting labour cost is given (cf. Appendix 2).

The manpower needs are not influenced by any dose constraints because the activity levels in the different systems in DR 2 are small. The different decommissioning processes like cutting free the reactor tank and removing the activated part of the concrete block will alone determine the manpower resources needed as well as the necessary time period for the decommissioning tasks.

In the decommissioning phase at least one health physics assistants is needed full time for surveillance of the working environments and measurements of all demolished parts, rubbles and scrap that are to be cleared as non-active waste. A part time health physicist is needed for radiation advice and for preparation of the documentation of measurement results and evaluation of the health physics consequences of any release of low level radioactive waste.

6.2.3 Special equipment needed for various operations

As mentioned in section 3.2.1, a project is running at present to prepare the decommissioning of DR 2. One task of this project will be to find out which equipment will be needed for the work. But it can be foreseen that special equipment may be needed, e.g. for cutting free the reactor tank, removing the inner part of the (heavy) concrete shield, cutting free the tubes through the concrete block etc. Furthermore, special γ-spectrometric equipment will be needed for the comprehensive measurement programmes necessary to document the residual activity levels in all items, scrap, rubble etc. that are to be cleared as non-radioactive waste.

6.2.4 Radiation doses to be expected

The estimated total activity of around 80 GBq distributed on several items within the reactor tank will not create any health physics problems. Any sub-activity of 80 GBq can be handled in free air without causing significant doses to the involved personnel. It is expected that the decommissioning of DR 2 will result in individual doses less than 5 $mSv \cdot y^{-1}$ per person.

6.2.5 Other safety related aspects

The top of the reactor block is almost 8 m above the floor level of the reactor hall and the working space at the top is limited. This means that the dismantling of the upper parts of the reactor block may involve some risks. However, the risks depend on the method selected for the dismantling, so it is difficult to make general statements about the risks, but one of them is that equipment, debris and personnel may fall down.

At the core region the reactor tank is provided with a lead layer situated between the tank wall and the concrete shield. The cutting of this layer may cause some difficulties and some risks due to its weight and activity.

It is likely that the removal of the graphite of the thermal column will give rise to difficulties. Due to the growth of the graphite during operation most of the graphite will presumably have to be broken up before it can be can be taken out. Since the graphite is radioactive, e.g. ^{14}C, care has to be taken to avoid contamination and inhalation.

At the end of the thermal column there is a layer of lead, which has been situated next to the core, and which may be rather radioactive. Due to its weight it may also be difficult to handle.

The primary circuit, i.e. hold-up tank, pumps, heat exchangers, valves and piping are very slightly radioactive. Removing this circuit should not pose any major difficulties or risks.

The reactor building is a high steel tank with an inside crane and its removal will involve some risks, in particular of fall-down accidents.

6.2.6 Amounts of radioactive waste produced and its treatment

It has been estimated that the total amount of waste after decommissioning DR 2 will be about 100 m3. This amount will include the reactor tank, the lead shield, the inner part of the concrete shield, parts of the primary cooling system, thermal column graphite, doors and lead shield, guide tubes, grid plate, and beryllium elements. If the activity content in the primary cooling system, contrary to expectation, cannot comply with the recommended clearance levels from EU, decontamination of the system component might be worthwhile to reduce the volume of radioactive waste. Sand blasting of the large tanks used in the former process plant in the DR 2 hall showed high decontamination efficiency and all tanks were completely cleaned for activity and deposited outside Risø.

6.2.7 Documentation to be produced

All activity and radiation measurements, including γ-spectrometric analyses on single items, scrap and rubble etc., must be documented rather detailed for the purpose of releasing as much as possible of the items with only minor content of activity.

6.2.8 Assessment of the costs

As mentioned in section 6.2.2, the costs have been assessed by means of the PRICE programme, cf. Appendix 2. The total cost is estimated to be about 28 million DKK.

6.2.9 Uncertainties in the assessments

Since a rather thorough examination has already been carried out of the radiological conditions at the plant, no major surprises are expected here. The demolition tasks identified for the PRICE calculation are very much standard operations, where it is expected that the PRICE database gives relatively certain values. A source of uncertainty may be the amount of concrete that can be cleared as non-radioactive.

6.3 Decommissioning DR 3

6.3.1 Operations to be performed

The description of the operations to be performed (see Appendix 3) is based on the NEA/IAEA/EC list of costing items[13]. The description contains the expected main groups of operations, the judged amount of manpower (man-weeks) and the expected procurements, all arranged in the expected chronological order from shut-down to final decommissioning.

6.3.2 Necessary manpower

In Appendix 3 an assessment is given of the manpower needed to perform the decommissioning operations. This estimate is based on the list of costing items proposed by the IAEA/NEA/EC. It takes into consideration only the amounts of man-hours needed and does not consider the restrictions that may be set by the magnitude of radiation fields. This aspect is addressed below, based on experience from UKAEA's Harwell site.

The British reactors PLUTO and DIDO were finally closed down in 1990. In the period 1990 - 1996 all external components at the PLUTO/DIDO reactors, which could conceivably constitute a hazard in the future, have been dismantled. These decommissioning tasks are shown in Table 6.3.1. The total collective (effective) dose from these operations has been measured to 20 - 30 man mSv, corresponding to an annual average dose of about 4 - 5 man·mSv. The dose distribution on the different tasks is shown in Figure 6.3.3.

The components removed can be characterised as "low-active". If the collective doses from dismantling the DR 3 components would be similar and, at the same time, the maximum annual effective dose should be less than a few mSv/person from these operations alone, the necessary manpower would be of the order of a few persons/year. If rigs/thimbles and major structural components would be dismantled during the first decade after a final shut down of DR 3 the collective dose would be much higher and remote handling might be necessary. If, for instance, the collective dose were 100

[13] A proposed standardised list of items for costing purposes in the decommissioning of nuclear installations. Interim technical document. OECD/NEA 1999.

times higher than from the dismantling operations shown in Table 6.3.1, i.e. around 3,000 man·mSv, the necessary manpower would be around 600 - 1,000 man·years for a maximum annual individual dose of 3 -5 mSv/y. Therefore, remote handling during dismantling of the structural components is needed in order to reduce significantly the individual and collective exposures.

In the decommissioning phase at least four full time health physics assistants is needed for surveillance during dismantling and measurements of all demolished parts, rubbles and scrap that are to be cleared as non-active waste. Three full time health physicists are needed for health physics evaluation and advice during the different dismantling operations, preparation of the documentation of measurement results and evaluation of the health physics consequences of any release of low level radioactive waste.

6.3.3 Radiation doses to be expected

The dose rate at a distance *r* from the each of the major components (considered as a point source) in DR 3 (top shield, reactor aluminium tank, reactor steel tank, biological shield, annular shield and graphite reflector) has been calculated for the γ-emitting nuclides with activity Q_i as:

$$\dot{H}_\gamma \approx \sum_i \Gamma_\mathrm{i} \cdot \frac{Q_i}{r^2}$$

Γ_i is here the exposure rate constant for the single radionuclides, *i*. The dose rate from each of the major components at a distance of one metre is shown in Figure 6.3.1. During a decommissioning operation where the massive shielding components are being demolished a significant self-shielding will reduce the dose rate by a factor of 3 - 5. For the top shield most of the activity will be situated in the bottom disc and the self-shielding effect will not be significant for the top shield bottom disc, reactor aluminium tank, steel tank and graphite elements.

If it is assumed that the average activity content in rigs and thimbles is dominated by ^{60}Co as indicated in Figure 3.3.5 by the reduction rate in activity, the dose rate after 10 years of cooling will be some hundred mSv/h at a distance of one metre. For the reactor aluminium tank and the bottom disc of the top shield the dose rate after 10 years of cooling will be a few Sv/h at one metres distance. For the remaining components shown in Figure 6.3.1 the dose rate after 10 years will be less than a hundred mSv/h at a distance of one metre when the self-shielding effect is included. These components are rather large and they will have to be cut into smaller pieces for final disposal. It is therefore not possible to be exposed to a full component at a distance as close as one metre, only to smaller fractions of these components. It is rather difficult to give a reliable dose estimate from any decommissioning tasks of the major structural components.

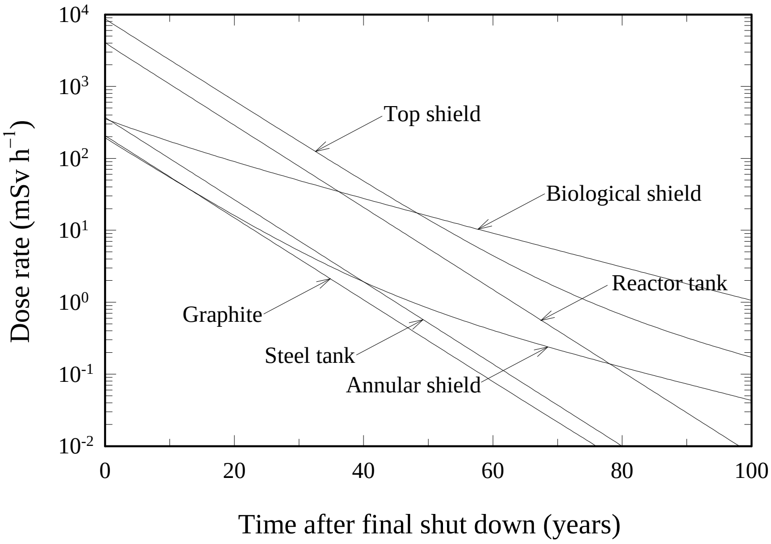


Figure 6.3.1 Dose rate one metre from structural materials in DR 3 as a function of time after shut down. The self-shielding within the massive shielding components will reduce the dose rate by at least a factor of three.

From Figure 3.3.5 and Figure 6.3.1 it appears that the dose rate from rigs and thimbles as well as from any sub-part of all the major structural components - except the biological shield - will decrease with a 5-year half-life, *i.e.* to 0.1% of the dose rate today after 50 years. For the biological shield (and the annular shield after 50 years) the dose rate is dominated by the radionuclides ^{152}Eu and ^{154}Eu with a half-life of 13.5 years and 8.6 years, respectively.

The dose rate inside the empty reactor tank (position C4) has been measured with TL dose meters placed in different positions below the bottom of the top shield. The results are shown in Figure 6.3.2 together with calculated dose rates in the same positions from a ^{60}Co content in the reactor tank and top shield as given in Table 3.3.1.

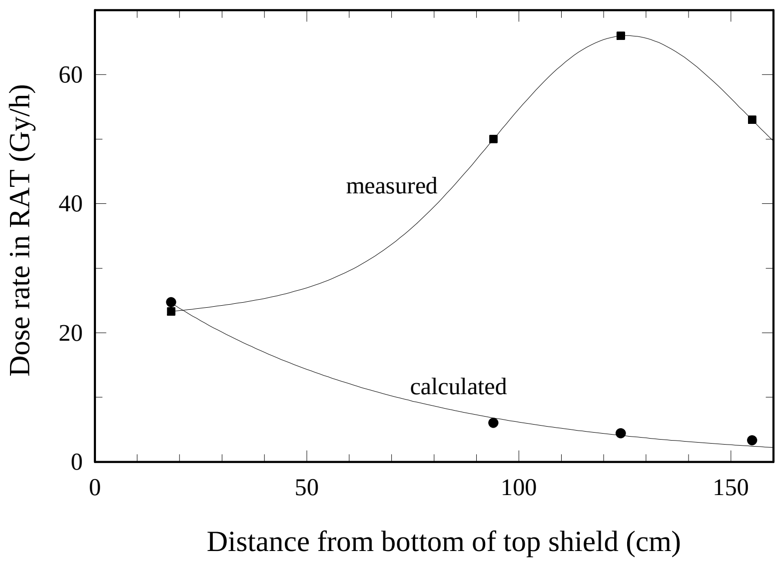


Figure 6.3.2 Measured dose rates in position C4 in the reactor tank on 6 January 2000 without heavy water, fuel elements and CCAs in the tank. The dose rates are calculated from a ^{60}Co content in the aluminium tank and bottom of top shield of 10^{13} Bq and $2{\times}10^{13}$ Bq, respectively (taken from the activity inventory Table 3.3.1 decay corrected for zero cooling time)

There is a significant difference between the measured and calculated dose rates at different distances from the bottom of the top shield. Several reasons for this are likely, *e.g.* the contribution from the surrounding 7V-thimbles. A steel plate has been inserted in the bottom of the thimbles for correction of the thermal neutron fluence rate, and the radiation from these plates might also have contributed to the measured dose rate around the core centre plane. Other radionuclides like ^{59}Fe and ^{65}Zn may also have contributed significantly at the time of measurement.

Several decommissioning tasks of "low-active" components were performed at the PLUTO and DIDO reactor as indicated in Table 6.3.1. The actual measured doses from these decommissioning tasks are shown in Figure 6.3.3 with a total of 20 - 30 man·mSv for each reactor. Before implementing the decommissioning tasks the collective dose was estimated to be 82 man·mSv for the DIDO reactor and 43 man·mSv for the PLUTO reactor.

Table 6.3.1 Decommissioning tasks at the DIDO and PLUTO reactors

Reactor system/Decommissioning task	
1	Secondary cooling
2	Shield cooling
3	Experimental rig cooling
4	Reactor control system
5	Emergency core cooling
6	Electrical systems
7	Experimental rig disposal
8	CO_2 and He supply systems
9	Experimental rig ancillary equipment
10	Flask disposal
11	Internal Storage Block cooling
12	Internal view cell
13	Waste processing

The experimental rig disposal, dismantling of experimental rig ancillary equipment, dismantling of the internal viewing box and waste processing are the major contributing tasks to the collective dose as indicated in Figure 6.3.3.

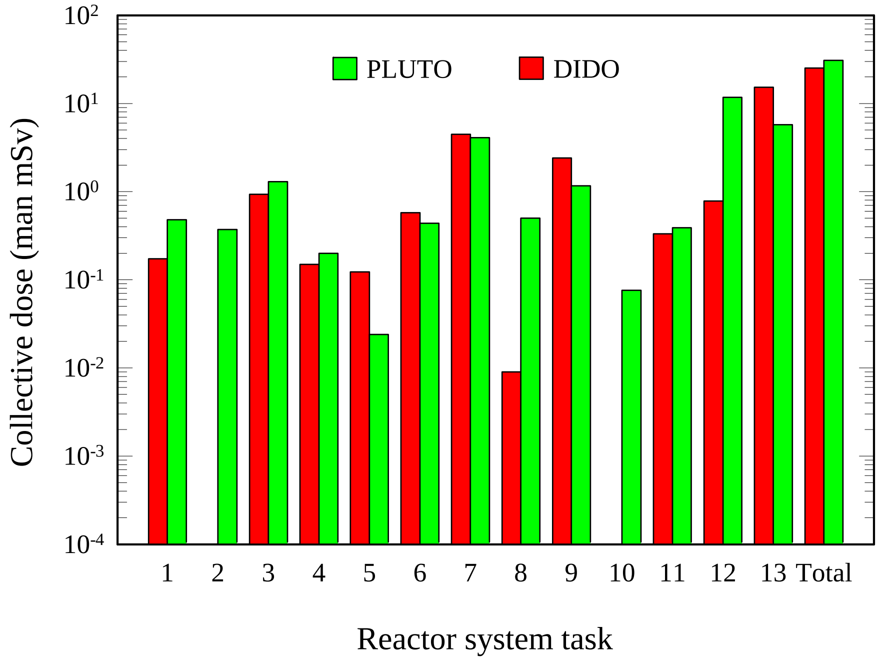


Figure 6.3.3 Actual collective doses from decommissioning tasks 1 - 13 at the reactors DIDO and PLUTO. The single tasks 1 - 13 are described in Table 6.3.1

The expected radiation doses to staff members from the dismantling of major structural components, *e.g.* top-shield and reactor tank, are difficult to assess without a detailed description of the physical dismantling process and a detailed knowledge of the radionuclide specific activity content in these components. However, a dose constraint on the individual doses can be used to assess the maximum collective doses from dismantling the more active components. Assuming that the manpower necessary to dismantle the major components is of the order of 30 - 40 man·years/a, and a dose constraint on individual doses is set at 5 mSv/a, the maximum annual collective dose will be *less than a few hundred man·mSv/a*. The dose constraint applied will implicitly set an upper limit for the speed of decommissioning of the more active components. An increased decommissioning speed can then only be achieved by increasing the number of persons involved in the operations. The collective dose would increase accordingly in this case.

6.3.4 Other safety related aspects

Apart from the radiological hazards, the safety related aspects mainly are concentrated in the fact that there will be many operations with hoists of heavy loads. Otherwise the risk pattern will be much alike that of conventional mechanical and electrical industry. However, the risk of accidents as a consequence of high working speed due to high radiation levels should be borne in mind when planning the operations.

6.3.5 Amounts of radioactive waste produced and its treatment

A description of the activated parts of the reactor is given in section 3.3.2. By using data for the DIDO reactor at Harwell an assessment has been made of the activity contents. The results are shown in Table 3.3.1 and, in simplified form, in Table 3.3.2.

Management of waste from decommissioning of DR3 will be complicated by the high contents of γ-emitters and the resulting requirement for shielding during handling and temporary storage. When it comes to waste disposal it is more the contents of long-lived radioisotopes that are important: ^{14}C (200

GBq), ^{63}Ni (920 GBq) and some additional even more long-lived ones, such as ^{59}Ni (estimated 6 GBq), ^{41}Ca (estimated 10 GBq) and a few others. The high content of tritium in the biological shield is also of concern, due to the high mobility of this isotope. The large amount of chemically toxic heavy metals must also be taken into account.

The treatment of the waste has to be evaluated and planned as a part of the decommissioning project. Packaging of the waste in waste units suitable for transport, storage and final disposal should to a high degree be carried out in close connection with cutting and other demolition work on the active systems. Thick-walled containers must be used for waste with high contents of γ-emitters.

A detailed plan for waste processing, storage and disposal including an overall safety analysis should be set up.

6.3.6 Documentation to be produced

Decommissioning operations should be planned in details and approved within a safety management system. A decommissioning operational safety document (DOSD) should be prepared containing operating rules, operating instructions, emergency instructions and project/safety management structures. Decommissioning operations might be classified in safety categories after their exposure risk and the DOSD should contain instructions on how the requirements in each safety category are to be applied. The following classification scheme in terms of potential individual doses is suggested:

- *Category A*: doses to off-site population > 0.1 mSv
- *Category B*: doses to decommissioning personnel > 10 mSv or to on-site personnel > 10 μSv
- *Category C*: doses to decommissioning personnel 1 - 10 mSv
- *Category D*: doses to decommissioning personnel < 1 mSv

For each safety category an approval procedure should be defined involving a safety committee, peer reviews, local management etc.

All activity and radiation measurements, including γ-spectrometric analyses, on single items, scrap and rubble etc., must be documented rather detailed for the purpose of releasing as much as possible of the items with only minor content of activity. In addition to that, it is necessary to make detailed scenario calculations of the health physics consequences of any conditional or unconditional releases of low-level radioactive waste as non-radioactive waste.

6.3.7 Assessment of the costs

The assessment of the costs has been carried out in accordance with the IAEA/NEA/EC list. The filled-in list and a spreadsheet with assessments of manpower and cost for the single activities are given in Appendix 3. As can be seen from the list and spreadsheet, there are tasks where we have felt able to estimate the man-weeks needed directly, while the cost of other tasks has been assessed as a bulk figure, appearing under the heading "Procurement" in the spreadsheet. Some of these bulk estimates have been derived from UK experience, using the PRICE programme, while others are based on engineering judgement carried out by the DR 3 staff. The estimates will include various amount of work required for specification, procurement control, testing and training. Some of these services will be done using internal resources. In the list all the original headings from the IAEA/NEA/EC's version have been preserved, although a number of the headings are not relevant for DR 3 or are addressed elsewhere, e.g. in section 6.9.

The total cost for the decommissioning of DR 3 has been found to be about 420 million DKK, depending on the scenario considered. This amount does include only labour cost and procurements for the operations identified. Administration costs and health physics costs have been estimated separately for the whole decommissioning organisation.

6.3.8 Uncertainties in the assessments

The assessments carried out in this first study of the decommissioning project are based on common engineering judgement of necessary manpower to projects and processes that have to undergo a much more detailed and elaborate planning process before they can be considered as data for budgeting. However they give an estimate of the order of magnitude of the costs. For the decommissioning of some main systems the cost evaluations have been taken from the preliminary cost estimate performed by the UKAEA on similar systems in the DMTR reactor in Dounreay, Scotland.

6.4 Decommissioning the fuel fabrication facility

6.4.1 Operations to be performed

It is expected that most of the contaminated equipment rather easily can be totally decontaminated. Floor and walls can also easily be cleaned, whereas decontamination of the big shake out table and the connected ventilation as well as the total drain system may be more time consuming.

6.4.2 Necessary manpower

The manpower needed is probably only a few man months in total. A health physics assistant should be present during most of the work.

6.4.3 Radiation doses to be expected

The work to be performed is not expected to give radiation doses different from those achieved in connection with the normal handling of uranium during the fuel element production. Most of the activity will be in the form of dust, so during the clean up work an appropriate mask has to be worn.

6.4.4 Other safety related aspects

Most of the work to be done is decontamination of surfaces contaminated with dust from handling unirradiated uranium. If this is planned and performed in a correct manner there should be no risk for spreading of radiation to the environment. Furthermore, the actual level of radiation is found to be so low, that the cleaning work is not foreseen to cause any stress or forced working speed giving an elevated risk for accidents.

6.4.5 Amounts of radioactive waste produced and its treatment

The total volume of radioactive waste will be very small – less than 1 m^3.

6.4.6 Documentation to be produced

When all the nuclear material accounted for has been removed - transferred to DR3 - and the active filters sent to the Waste Treatment Plant there will only be minor activities left from surface contamination inside the channels in the ventilation system and inside the pipes in the drain system. If also these installations are removed there will be no need at all for a Safety Documentation with operation conditions issued by the Authorities. If the installations are not removed, it will still be necessary to have some kind of Safety Documentation for the nuclear facility during dismantling.

6.4.7 Assessment of the costs

In total all the work to be done is expected to cost in the order of 100,000 DKK, and this is all labour costs.

6.4.8 Uncertainties in the assessments

The amount of manpower and thus the costs needed for the necessary decommissioning work could be higher, but not the double, and it is more likely that it would be lower than expected.

6.5 Decommissioning the Isotope laboratory

6.5.1 Operations to be performed

The ^{36}Cl-activity in laboratory no. 2 is fixed on walls, laboratory tables, metal supports etc. A feasible cleanup method is sandblasting of walls/ceiling and maybe metal supports after removal of the laboratory tables. The floor is clean. The activity is not expected to penetrate deeper into the walls/ceiling as the small amounts of HCl will react with the paint, concrete or metal at the surface. The cleanup of inner cell surfaces is not a difficult task; normal vacuum cleaning and washing/wiping will probably remove most of the contamination.

6.5.2 Necessary manpower

The manpower resources needed are probably measured in (a few) man·weeks. Dismantling of lead-shielded cells and ventilation systems is a more time-consuming operation, probably of the order of some man·months. A health physics assistant should be present during the operation.

6.5.3 Radiation doses to be expected

^{36}Cl can be classified as a low-toxic radionuclide. It is a pure β-emitter ($E_{\beta,\,max}$ = 709 keV) with no external radiation risk. The inhalation dose per unit activity inhaled is 0.5 mSv·MBq^{-1}. If the deposited activity in the laboratory (some tens of MBq) is released during the cleanup operations decontamination workers can inhale only a very small fraction of this activity. Even if a large fraction (≈10%) of the deposited activity is inhaled by one person the resulting dose will be of the order of a mSv (zero dose if respiration protection is used). The cleanup of laboratory no. 2 is therefore a low-risk operation and no special safety precautions are needed. The cleanup and dismantling of cells and ventilation ducts are not expected to cause any significant doses.

6.5.4 Other safety related aspects

No particular hazards are foreseen for the decommissioning of the Isotope Laboratory.

6.5.5 Amounts of radioactive waste produced and its treatment

Decontamination of laboratory no. 2 will produce a small amount of radioactive waste, *i.e.* less than a few m^3. The rubble created by the cleanup might be released as non-radioactive waste if the concentration of ^{36}Cl is below recommended clearance levels. The EU-recommended clearance level for ^{36}Cl-activity in building rubble[14] is 1.1 kBq·kg^{-1}. For steel scrap[15] the clearance level is 13 kBq·kg^{-1}.

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[14] *Definition of Clearance Levels for the Release of Radioactively Contaminated Buildings and Building Rubble.* Final Report. Contract C1/ETU/970040 carried out on behalf of the European Commission, Directorate General XI (1999).

The lead-shielded cells and ventilation channels are more bulky and the waste volume might be several m^3, but most likely the lead bricks can be reused or recycled and the major part of the ventilation ducts cleared as non-radioactive. The piping for the wastewater is according to existing experience not contaminated but the piping for the two rabbit systems might be contaminated at a low level. The total volume for the two systems $\approx 1\ m^3$ if compressed.

6.5.6 Documentation to be produced

A description of the facility and the decommissioning work carried out should be kept for the record.

6.5.7 Assessment of the costs

Assuming that the necessary manpower, as indicated in section 6.5.2, is of the order of 10 man weeks, and using the labour rate of 8550 DKK/man week, the labour cost can be estimated to the order of 100,000 DKK. Material- and equipment costs may add another 100,000 DKK, bringing the total cost to 200,000 DKK.

6.5.8 Uncertainties in the assessments

The uncertainty of the cost given in section 6.4.7 is estimated to be +50% - 100%.

6.6 Decommissioning Hot Cells

6.6.1 Operations to be performed

The first task is to set up a survey programme based on the previous survey, but this time with decommissioning in mind. Special attention should be paid to the conduits underneath the cells, which might be contaminated.

As the cells are encapsulated and the operational external facilities have been removed, some changes in the building must be foreseen, including the construction of an entrance area.

On the top of the cells various contaminated mechanical and venting equipment are stored. This equipment and the shutter installations must be removed before the cell top can be cleaned.

Ventilation in the cells must be restored and the crane checked for operation.

Before decommissioning of the Hot Cells is started, most of the activity content within the concrete cells has to be removed by various decontamination techniques. This is a necessary step to avoid an unacceptable exposure of the workers when the heavy components and concrete walls are to be demolished. Several decontamination techniques are applicable, e.g. sandblasting, shot blasting and different high-pressure jetting techniques. Removal of the wall strip coating will remove most of the contamination at the walls.

Important tasks are the removal of the conveyor belt system and the iron tables within each of the concrete cells. It is expected that several Co-pellets might be trapped in cracks and slits between the cell front wall and the tables. Hot spots were detected on the floor after the remote cleaning and they

[15] *Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations.* Radiation protection 89. European Commission (1998).

were covered by lead bricks. These hot spots should also be removed before the general surface cleanup is implemented.

Outside the cells a rotating storage facility is situated under the floor. This facility must be decontaminated and removed.

After decontamination of the cells the crane and the venting system can be decontaminated and removed.

The concrete walls of the cells and the steel reinforcement can be removed using conventional tools and protection aids.

Finally the building must be restored.

6.6.2 Necessary manpower

In Appendix 4 an assessment is given of the manpower needed to perform the decommissioning operations. This estimate is produced with the help of the PRICE programme. The operations to be performed have been identified by Risø staff with the assistance from retired staff, knowledgeable with respect to the Hot Cell facility's design and construction. The estimate takes into consideration only the amounts of man-hours needed and does not consider the restrictions that may be set by the magnitude of radiation fields. This aspect is addressed below, based on knowledge of the activity inventory, measured during the earlier decommissioning project.

Effective doses to workers decontaminating the inner surfaces of the concrete cells should respect existing dose limits. Effective doses should be less than 5 - 10 mSv/person and skin doses less than 50 - 100 mSv/person. The collective dose from a future decontamination can be roughly estimated from the doses caused by the cleanup of the cell no. 5 and 6. The external collective γ-dose per removed amount of activity has been estimated to 0.2 person·mSv·GBq^{-1} (see Section 6.4.5). Assuming a similar cleanup efficiency in a future decontamination and an additionally exposure from 'hot spots' the collective dose might be 400 - 800 person·mSv or even higher. It is here assumed that the internal dose component as calculated in Section 6.4.5 can be neglected due to application of breathing protection. A maximum effective dose is 5 - 10 mSv/person render a staff of decontamination workers of 40 - 160 for a complete manual decontamination. More efficient methods than high-pressure water jet would reduce the collective dose per unit activity removed and thus a reduced manpower need. Also the use of remote controlled decontamination equipment should be given a high priority in order to avoid a substantial part the predicted collective dose.

During the period of cleanup at least two health physics assistants are needed full time for surveillance, air and smear sampling etc. and a part time health physicist for advice and evaluation of the overall cleanup operation. In the decommissioning phase at least two health physics assistants is needed full time for surveillance and measurements of all demolished parts, rubbles and scrap that are to be cleared as non-active waste. A full time health physicist is needed for radiation advice and for preparation of the documentation of measurement results and evaluation of the health physics consequences of any release of low level radioactive waste.

The necessary manpower for the work is estimated to be 10 persons for three years, bearing in mind that health physics survey and project management are general costs (approx. 20%). Of course, the degree of robotisation chosen will affect the number of people needed - both from a work point of view and from a radiation protection point of view.

6.6.3 Radiation doses to be expected

External γ-doses to the staff members involved in the cleanup of the concrete cell 5 and 6 was measured using personal dosimeters. The inhalation doses were zero because breathing protection was applied (frogman suits). The external radiation field in the cells was measured with a matrix of TL-dosimeters before and after the high-pressure water cleaning. The radiation field measurements were used as input to an activity-dose model to calculate the activity reduction by cleanup, ΔQ. The collective dose to the involved persons, S, was calculated as the sum of the individual doses. The "cost" of the cleanup, $S/\Delta Q$, of concrete cell 5 and 6 was calculated to 0.2 person·mSv·GBq^{-1}. The remaining activity in the concrete cell 1 - 6 is about 3,300 GBq (1993). A total cleanup would "cost" about 600 person·mSv if the conditions for cleanup were similar to those during cleanup of cell 5 and 6 (cleanup efficiency etc.).

The potential doses from any future cleanup can also be estimated from the measured external γ-dose rate and surface contamination density of actinides and fission products in each cell. The surface contamination density can be converted to inhalation dose rate (more correctly inhalation dose per unit residence time) for a given value of the resuspension factor (ratio of activity concentration in air to surface contamination density). The inhalation doses can normally be neglected because breathing protection is obligatory for cleanup operations of this nature. The radiation environment in the cell 1 - 6 is presented below.

External dose rate in the concrete cells

The average γ-dose rate in cell no. 1 - 3 is about 0.1 - 0.2 mSv·min^{-1}. Close to the 'hot spots' in the cells the γ-dose rate can be as high as 1 - 2 mSv·min^{-1} and the corresponding β-dose rate 10 - 100 mGy·min^{-1} (reference time 1993). The β-dose to skin can be reduced by proper protective clothing.

The dose rate from the single radionuclides in the concrete cells has been calculated from the measured total dose rate and the activity distribution on radionuclides. Assuming that this is similar to the distribution found on the smear samples the dose rate from the single nuclides can be calculated from:

$$\dot{E}_i \cong \dot{E}_{\text{meas}} \cdot \frac{\Gamma_i \cdot Q_i}{\sum_i \Gamma_i \cdot Q_i}$$

Γ_i is here the dose rate constant and Q_i the activity of radionuclide i. The calculated dose rate for each of the nuclides is given in Table 6.6.1 (1993).

Table 6.6.1 Average dose rate in the concrete cells based upon dose rate measurements with TL-dosimeters

Radionuclide	Dose rate in concrete cell 1 - 6 [mSv·h^{-1}]					
	Cell 1	Cell 2	Cell 3	Cell 4	Cell 5	Cell 6
^{137}Cs	5.2	5.2	9.6	0.5	1.6	0.4
^{134}Cs	0.4	0.4	0.8	0.04	0.1	0.04
^{152}Eu	0.01	0.01	0.01	0.001	0.002	0.001
^{154}Eu	0.2	0.2	0.4	0.02	0.06	0.02
^{60}Co	0.6	0.6	1.2	0.06	0.2	0.04
Total	6.5	6.5	12	0.6	2.0	0.5
Range	3.0 - 11	4.5 - 12	9.0 - 15	0.4 - 0.9	1.6 - 2.7	0.4 - 0.9

The average γ-dose rate in each cell is shown in Figure 6.6.1 as a function of time. As ^{137}Cs is the dominating radionuclide the dose rate will be reduced by only a factor of two over the next 30 years.

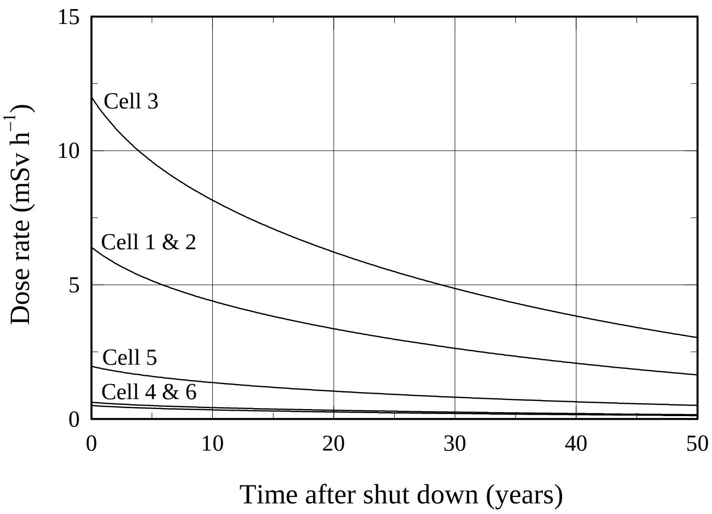

Figure 6.6.1 Dose rate in the concrete cell 1 - 6 as a function of time. The cells were shut down in 1993

Internal dose rate in the concrete cells

Potential doses from inhalation of air contaminated by resuspension can be calculated from the activity in the cells expressed by its radiotoxicity. The total activity in cell no. 1 - 6 is shown in Table 6.6.2 (1993) for each of the radionuclides together with the number of ALIs (**A**nnual **L**imit on **I**ntake) for inhalation and oral intake of these nuclides (calculated as the activity content divided by the ALI). Inhalation of one ALI will cause an internal (committed effective) dose of 20 mSv.

Table 6.6.2 Total activity content in the concrete cell 1 - 6 and its potential radiotoxicity for inhalation and oral intake

Radionuclide	Activity [GBq]	ALIs for inhalation	ALIs for oral intake
^{60}Co	55[16]	$1{\cdot}10^5$	$2{\cdot}10^4$
^{90}Sr	1,310	$2{\cdot}10^7$	$3{\cdot}10^5$
^{137}Cs	1,850	$9{\cdot}10^5$	$2{\cdot}10^6$
^{238}Pu	56	$2{\cdot}10^8$	$2{\cdot}10^5$
^{239}Pu	5.6	$2{\cdot}10^7$	$1{\cdot}10^5$
^{240}Pu	9.3	$3{\cdot}10^7$	$2{\cdot}10^5$
^{241}Am	1.9	$6{\cdot}10^6$	$6{\cdot}10^4$
^{243}Am	0.3	$1{\cdot}10^6$	$1{\cdot}10^4$
^{244}Cm	12	$2{\cdot}10^7$	$2{\cdot}10^5$
Total	-	$3{\cdot}10^8$	$3{\cdot}10^6$

The potential doses from inhalation of resuspended material can be calculated from the total number of ALIs in the cells. Assuming a homogeneous activity distribution over the inner cell (floor and wall) area of 460 m^2 and a resuspension factor of 10^{-5} m^{-1} (which can be much higher during a decontami-

[16] This activity is from activated reactor fuel alone. An unknown number of Co-pellets are present from the production of radiotherapy sources.

nation operation) the inhalation dose per unit residence time without any protection can be calculated as:

$$\dot{E}_{\text{inh}} = 0.02\ \text{m}^3 \cdot \text{min}^{-1} \cdot 20\ \text{mSv} \cdot \text{ALI}^{-1} \cdot 10^{-5}\ \text{m}^{-1} \cdot \frac{3 \cdot 10^8\ \text{ALI}}{460\ \text{m}^2} \cong \underline{3\ \text{mSv} \cdot \text{min}^{-1}}$$

More than 90% of this dose is caused by inhalation of actinides. The inhalation dose from cleanup of the most contaminated cell no. 1 - 3 might be twice as high as the average inhalation dose. The inhalation doses can be significantly reduced when breathing protection is applied.

6.6.4 Other safety related aspects

Apart from the radiological hazards, the safety aspects will mainly be conventional worker safety, e.g. related to operations with hoists of heavy loads. As much work inside the cells is foreseen to be carried out with full frogman suit protection, the risk of such "conventional" accidents may be slightly enhanced.

If it is decided to perform the demolition of the facility without evacuating the building, the risk of accidents affecting other people in the building must be assessed.

6.6.5 Amounts of radioactive waste produced and its treatment

Slightly contaminated items are placed on top of the concrete cells, e.g. parts from the cell ventilation, tubes and other items, which should be removed and stored as low-active waste before a decommissioning of the cells can start. The volume of these items is approximately 10 - 20 m^3. Heavy equipment within the concrete cells and the cell airlock should also be removed after the cell decontamination before the cells can be demolished. This heavy equipment includes shutter doors, steel covered lead doors between cells, conveyor belt system, steel tables, doors, cell crane, 30 cm lead end wall, and parts of the ventilation system. After removal from the cells the equipment should as far as possible be decontaminated. Otherwise they have to be stored as low active waste. The volume of this 'heavy machinery' is approximately 40 - 50 m^3. In addition, some secondary waste (~10 m^3) will be produced during decontamination.

The thickness of the cell ceiling is 100 cm concrete, the outer wall thickness is 170 cm concrete and the thickness of the walls between cells is 100 cm magnetite concrete. Large amounts of concrete scrap will be produced from the decommissioning. The volume of produced concrete scrap has roughly been estimated to several hundreds m^3 per cell. The total volume of concrete scrap is therefore several thousands m^3. Unless a complete decontamination of the inner cell surfaces (or removal of the inner contaminated part) is undertaken this concrete scrap has to be stored as low active waste.

After a complete decontamination of the inner cell surfaces and the 'heavy machinery' both concrete scrap and 'heavy machinery' might be deposited as non-active waste outside Risø if the activity concentration can comply with internationally recommended 'clearance levels'. However, the authorities should give a final approval in each case.

6.6.6 Documentation to be produced

Cleanup and decommissioning operations should be planned in details and approved within a safety management system. A decommissioning operational safety document (DOSD) should be prepared containing operating rules, operating instructions, emergency instructions and project/safety management structures. Decommissioning operations might be classified in safety categories after their exposure risk and the DOSD contains instructions on how the requirements of the safety categories are to be applied.

All activity and radiation measurements, including γ-spectrometric analyses, on single items, scrap and rubble etc., must be documented rather detailed for the purpose of releasing as much as possible of the items with only minor content of activity. In addition to that, it is necessary to make detailed scenario calculations of the health physics consequences of any conditional or unconditional releases of low-level radioactive waste as non-radioactive waste.

6.6.7 Assessment of the costs

The costs of final decommissioning of the Hot Cells were estimated by means of the PRICE programme, as shown in Appendix 4. The resulting cost is about 25 million DKK.

6.6.8 Uncertainties in the assessments

One source of uncertainty in the assessment of the work is the question of whether the ducts below the cell floor are contaminated. Furthermore, the estimates of the costs for decontamination and removal of the cells are considered uncertain.

6.7 Decommissioning the waste storage facilities

6.7.1 Operations to be performed

The principal problem in connection with the waste storage facilities (items 2 to 6 in Section 3.7) is to provide disposal for the stored waste units. After the waste has been removed the dismantling of the empty storage facilities should be straightforward involving only surface contamination at rather low levels.

Part of the storage system can be dismantled as soon as a disposal facility is available, but some buffer storage will continue to be needed for waste coming from other decommissioning work or from outside suppliers.

Approximate amounts waste stored in the various storage facilities are given in Table 6.7.1. Possibilities for extending and improving the credibility especially the activity inventories are being pursued. The information in Table 6.7.1 (with additional details) together with estimates for the amounts of decommissioning waste from the nuclear facilities presented elsewhere in this report is needed as input to the planning of the Danish low-level near-surface repository. Details concerning the existing waste are not a topic of this report.

Table 6.7.1 Approximate amounts of waste in the storage facilities at Risø (at 1/1-2000). At the end of 2001 there will be about 4700 waste units in the storage hall.

		Tromlelager	Storage hall	Centralvejs-lager	Tailings basins and ore storage
Stainless steel containers (30 l etc)	m^3			(8)	
	pcs			77	
Drums (210-280 l)	m^3	(25)	(1500)	(55)	
	pcs	75	4600	165	
Other waste packages	m^3			150	10
Uranium pilot plant waste	m^3				3400
	t				4800

Comment: In the repository a 210-280 L drum is assumed to occupy 330 L and a 30 L stainless steel container about 100 L leaving space for some back-fill material. Disregarding that a minor amount of the waste may not be permitted in a near surface disposal facility the total volume required for the existing waste units in the disposal facility is about 1740 m^3 plus the 3400 m^3 uranium pilot plant waste. Actually the volume requirement per waste unit might be somewhat higher.

Cost and manpower requirements for transfer of the waste units from the storage to the final disposal facility will depend on the siting of the disposal facility. Experience from the transfer carried out 1993 to 96 of 3288 low-activity drums from 'Betonrørslageret' to the now used Storage Hall indicates that 1 to 2 man year (distributed on at least 3 operators) may be required if the transfer takes place inside the Risø area. Due to further decay and easier handling the radiation dose should be less than the 70 man-mSv received by the operators during the previous transfer. Actually, the doses will depend very much on the practice adopted for handling and transfer of the relatively few waste units with significant external radiation. No attempt has been made to estimate manpower requirements for handling the waste from the uranium extraction pilot plant.

The operations to be carried out after removal of the waste units in Tromlelageret, the low-level Storage Hall and Centralvejslageret comprise decontamination by chemical means or by sand blasting of internal steel surfaces in Centralvejslageret and U-irons used as support for the drums in the Storage hall. Removal of the upper few cm of slightly contaminated concrete from floors in the Storage hall and maybe the shielded area in Tromlelageret may be necessary. The amount of radioactive waste will be slight, maybe ~30 m^3 of concrete rubble and sand with very low activity. After the decontamination dismantling of the buildings should be possible using conventional demolition technique resulting in ordinary, inactive building waste.

As part of the discussion about siting of the final disposal facilities, it may be decided that the tailings and remaining uranium ore have to be moved elsewhere and in that case the concrete structure of the two used storage basins has to be demolished. Even if the tailings remain in the facility the tank system used for overflowing rainwater will probably not be needed. Clean up of the area where the ore is presently stored must also take place. The demolition waste will be contaminated with uranium and to lesser degree with thorium and isotopes from the decay chains, e.g. radium. In general the level is expected to be low, lower than the content in the ore and approaching the level in soil or ordinary building materials. Declassification for other use may well be possible, but the 350 m^3 concrete (or 600 m3 as rubble) and slightly contaminated soil (maybe 400 m^3) have been included as low level waste in Table 6.7.2.

Table 6.7.2 Estimated amounts of waste and radiation dose for complete decommissioning of four storage facilities belonging to the waste management system.

		Tromlelager	Storage hall	Centralvejs-lager	Tailings basins and ore storage
Low level waste:	m^3	5	20	5	1000
Radiation dose:		~0	~0	~0	~0
Inactive demolition waste					
Concrete etc	t	600[1)]	1000	1000	-
Aluminium:	t	1	5	2	-
Steel:	t	1	15[2)]	60[3)]	7
Lead:	t	-	-	15[4)]	-

1) Concrete and bricks from demolishing the whole building, the shielding walls is about 120 t.
2) 2 t from decontaminated U-steel used or drum support, the rest is construction steel in the building.
3) 40 t from decontaminated steel cladding from pits and tubes, the rest is construction steel + crane.
4) Shielding in lead flask and other handling equipment

6.7.2 Other safety related aspects

Apart from the radiological hazards - which are more or less negligible in this case, the safety aspects will mainly be conventional worker safety, e.g. related to operations with heavy loads.

6.7.3 Documentation to be produced

Logs should be kept of the amounts and the activity contents of waste transferred to an intermediate storage or the final repository. Documentation of clearance measurements should be produced and archived.

6.7.4 Assessment of the costs

As mentioned in section 6.7.1 (page 86) it is estimated that emptying the storage facilities will take 1-2 man years; in the table below we have assumed it to be 2 man years (where 1 year equals 45 working weeks, and a labour rate of 8550 DKK is being used).

Furthermore, in Table 6.7.3 rough estimates have been added for the costs of demolition of the storage facilities. The estimates for the Centralvejslager, Tromlelager and Low level storage hall have been produced by applying a cost per m^2 that has been established on the basis of the costs per m^2 found for other buildings by DEMEX. It has been assumed that the decontamination of the buildings will not be excessive.

Table 6.7.3 Cost estimates for decommissioning waste storage facilities

Facility	Area, m^2	Cost/m^2, DKK	Man-weeks	Cost kDKK
Emptying storage facilities			90	770
Demolition and removal of:				
Centralvejslager	260	800		208
Tromlelager	220	600		132
Low level storage hall	780	400		312
Tailings and ore			50	427
Sum				**1,849**

6.7.5 Uncertainties in the assessments

The cost estimates given in section 6.7.4 are rough estimates. But they can be considered as being within a factor of two of the "correct" value. Anyway, it should be borne in mind that the way of disposing of some of the material, e.g. the uranium ore, has not yet been decided.

6.8 Decommissioning the waste management plant

6.8.1 Operations to be performed

The waste treatment facilities (evaporation and bituminisation plants, solid waste compaction and decontamination of protective clothing, items 1 and 2, described in Section 3.7) will be needed during demolition of the other nuclear facilities. Decommissioning of the waste treatment facilities must therefore occur late in the process.

Especially the waste sorting and compaction facility will have to be supplemented with cutting and compaction facilities at the decommissioning sites. Large-scale decontamination must also be carried out at the sites. Waste from demolition of such additional facilities are not included here. Compatibility of decontamination waste with the wastewater treatment plant must be evaluated.

The interior of the wastewater treatment facilities is contaminated with β,γ and at a lower level also with α emitters. Taking present concentrations in the concentrate in the evaporator (e.g. 50 Bq β/ml and 1 Bq α/ml) as typical contamination level the internal surfaces in the evaporator and bituminisation plant (in total about 400 m^2) might be contaminated with the order of $400\times10^4\times0.1\times50\times10^{-6}$ = 20 MBq β after emptying. However, nooks and corners in the complex system may contain more activity. At the end the plant should be run with clean water for some time taking care of cleaning e.g. the steam dome for solid deposits. After dismantling complete decontamination of the stainless steel surfaces should be possible. Recovery of the removed activity must take place in another manner since the evaporator no longer will be available. Also the bituminisation plant can be decontaminated although disposal without attempting to remove remaining bituminised material may be preferable for the evaporator pot. In total some 10-15 t of stainless steel may be recoverable for recycling.

The cells where the plant is situated (150 m^2) are not contaminated except cell no. 2 where the floor should be removed and treated as low-level waste (15 m^2, ~2 m^3). The concrete shielding of the cells can be demolished as inactive, as is also the case for the steel frame/ aluminium plate structure of the evaporator hall. Some 350 m^2 wall covering with asbestos- containing material must be treated appropriately and will complicate the demolition work.

The various tank systems (item 6 in Section 3.7.) for collecting active wastewater should be cleaned for sludge and the inner protective epoxy layer removed by sand blasting in situ or in a central facility after the tanks have been cut into pieces. Complete decontamination of the slightly contaminated tanks should normally be possible, but the bottom part of some may be better treated as low-level waste (in total resulting in estimated ~2 m^3 very low-activity sludge, sand and steel). The cooling water tanks are not contaminated and can be dismantled directly. In total about 20 t ordinary steel may be released.

The drainage tubes conducting active wastewater from the laboratories etc. are mostly of some plastic material. They will be difficult to clean completely and disposal may be preferable (~1000 m, ~20 m^3, some might be burned but older parts would be PVC).

Slightly contaminated spots on the concrete floors in the tank cellars or (especially) in the engineering channels where the tubes have passed through may occur but should only constitute a very minor part of the underground channel system, (e.g. 5 m^3 rubble).

The decontamination facility for working clothes is classified as blue contamination area, but the equipment itself is only slightly contaminated and can probably be released after simple cleaning. The deeper layers in the floor covering are somewhat contaminated and should be removed and treated as low-level waste (30 m^2 , ~1 m^3).

The sorting box and compaction press for solid waste are also internally contaminated to a low but unknown level. The components can probably be decontaminated but the whole facility may also be cut and treated as low-level waste (~5 m^3 steel and perspex).

Table 6.8.1 Estimated amounts of waste, manpower requirements and cost for complete decommissioning of the treatment facilities belonging to the Waste Management Plant.

		Evaporator + bituminisation	Tank systems	Decontamination room	Sorting box and balling press
Activity inventory	GBq β	0,1?	low	low	1 ?
Low level waste:	m^3	3	2	1	5
Radiation dose:		~0	~0	~0	~0
Inactive demolition waste					
Concrete etc	t	800[1)]	~100	0[2)]	0[3)]
Asbestos material	t	50	-		
Aluminium:	t	2	-		
Steel:	t	-	20[4)]		
Stainless steel:	t	15	-		
Manpower (excl. demolition of buildings):	man-weeks	50	100	10	10
Costs (excl. demolition of buildings):	kDKK	427	855	86	86

1) 440 t from shielding walls, the rest is from floors in the evaporator part of building 211.
2) The decontamination room is cleaned but like the rest of building 211 not demolished.
3) Demolition waste for this part of building 212 is included in the total for Tromlelageret in Table 6.7.2.
4) Excluding tanks at DR2 and DR3

6.8.2 Other safety related aspects

Apart from the radiological hazards - which are more or less negligible in this case, the safety aspects will mainly be conventional worker safety, e.g. related to operations with heavy loads.

However, when planning the demolition of contaminated parts of the facility consideration should also be given to the risks of accidental release of radioactive material to the environment.

6.8.3 Documentation to be produced

Documentation of the facilities should be preserved. Furthermore, logs should be kept of the amounts and the activity contents of waste transferred to an intermediate storage or the final repository. Documentation of clearance measurements should be produced and archived.

6.8.4 Assessment of the costs

In Table 6.8.2 a summary is given of the costs of decommissioning the waste treatment plant, including the wastewater tank systems. Similar to the procedure in Table 6.7.3 the rough estimates of the costs of demolition of the buildings have been produced by applying a cost per m^2 that has been established on the basis of the costs per m^2 found for other buildings by DEMEX.

Table 6.8.2 Summary of costs for waste treatment plant

Facility	Area m^2	Cost/m^2 DKK	Costs from Table 6.8.1	Total cost kDKK
Decommissioning of systems				
Evaporator + bituminisation			427	427
Decontamination room			86	86
Sorting box and balling press			86	86
Decommissioning of buildings			855	855
Tank systems	~7×100	600		420
Building 211	1200	600		720
Sum				**2594**

6.8.5 Uncertainties in the assessments

The cost estimates shown in Table 6.8.2 are rough estimates. But they can be considered as being within a factor of two of the "correct" value.

6.9 General activities

6.9.1 Health physics

The issues of organisation and qualification requirements, radiation protection tasks during decommissioning of nuclear facilities (protection of the workforce, population and environment as well as emergency response preparation) and research activities are all addressed below.

Organisation and qualification requirements
The EU Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising

radiation (L 159) includes fundamental requirements to the radiation protection organisation. Necessary arrangements shall be made to recognise, as appropriate, the capacity of:

- the approved medical practitioners,
- the approved occupational health services,
- the approved dosimetric services,
- the qualified experts[17].

To this end, it shall be ensured that the training of such specialists is arranged.

It is required that the means necessary for proper radiation protection are placed at the disposal of the units responsible. A specialised radiation protection unit, distinct from production and operation units in the case of an internal unit, authorised to perform radiation protection tasks and provide specific advice shall be required for the installations, which the competent authorities consider necessary. This unit may be shared by several installations.

In the IAEA Safety Guide Decommissioning of Nuclear Power Plants and Research Reactors, No. WS-G-2.1, it is stated that those charged with the day to day responsibility for radiation protection should have the resources, access to decommissioning management and independence necessary to effect an adequate radiation protection programme.

Radiation protection tasks

The radiation protection tasks necessary for conducting the practice *decommissioning of nuclear installations* include protection of the workforce, protection of population and environment and preparation of intervention measures, including operation of a health physics emergency response organisation.

Protection of the workforce and support for decommissioning

Radiation protection tasks include the process of ensuring that the approach and methodologies in use are within established guidelines and accepted health and safety practices. Major radiation protection activities are:

- controlling field operations
- radiation protection specialist tasks, including *e.g.* shielding design
- setting work area/access requirements
- maintaining documentation of the facility's changing radiological conditions
- area monitoring, including:

- continuous air monitoring
- tritium monitoring
- special case monitoring

- personal dosimetry, including:

- advice on the use of personal dosimeters
- advice on the use of special dosimeters
- assessment of internal doses from urine sample analyses
- use of digital dosimeters

- advice on personnel contamination check, including:

- hand and clothing monitoring
- whole body monitoring
- handheld monitoring

- operation activities, including:

- surveying of radiation and contamination levels during operations

[17] Persons having the knowledge and training needed to carry out physical, technical or radiochemical tests enabling doses to be assessed, and to give advice in order to ensure effective protection of individuals and to correct operation of protective equipment, whose capacity to act as a qualified expert is recognised by the competent authorities. A qualified expert may be assigned the technical responsibility for the tasks of radiation protection of workers and members of the public.

- work area monitoring, including smear sample analyses
- advice on personal protective equipment
- advice on personnel decontamination
- radiation protection training

Other important tasks include the support for decommissioning engineering and planning as well as supervising the decontamination and dismantling processes:

- sampling for radiological inventory characterisation in the installations
- determination of the inventory of radioactive materials inside the nuclear installations and in their immediate surroundings
- facilities for decontamination of systems for dose reduction
- radiological inventory characterisation for decommissioning and decontamination
- removal of waste from decontamination operations
- decontamination of areas and equipment in buildings to facilitate dismantling and release
- final radioactivity survey
- characterisation of radioactive materials

Protection of the population and the environment
In accordance with the principles of health protection of the population in the area of radiation protection the following tasks shall be carried out:

- achieving and maintaining an optimal level of protection of the environment and the population with regards to operational and potential releases of radioactive materials;
- acceptance into service, from the point of view of surveillance of radiation protection, of equipment and procedures for measuring radioactive contamination of the environment.

Qualified experts and, as appropriate, the specialised radiation protection unit referred to above shall be concerned in the execution of these duties.

Emergency response preparedness
Account shall be taken of the fact that radiological emergencies may occur in connection with practices on or outside its territory and affect it. Appropriate intervention plans shall be drawn up, taking account of the general principles of radiation protection for intervention, in order to deal with various types of radiological emergency and that such plans are tested to an appropriate extent at regular intervals. Where appropriate, provisions shall be made for the creation and appropriate training of special teams for technical, medical and health intervention. A health physics emergency response organisation shall be available to respond to any emergency situation that involves exposure of workforce and population and contamination of the environment.

Compliance with EU-requirements on radiation protection at Risø
The radiation protection work at Risø has since 1957 been organised and authorised by the competent authorities to perform radiation protection tasks and provide specific radiation protection advice, *distinct from the operation of the nuclear facilities* as required in the EU Basic Directive on Radiation Protection. The requirement for independence of the radiation protection unit is due to the risk of mixing safety and operational issues, and that radiation protection otherwise might be short of resources. At present, the radiation protection tasks at the nuclear facilities are performed by the Section of Applied Health Physics as a part of the Nuclear Safety Research Department, which has radiation protection as one of its major activities.

6.9.2 Necessary major pieces of equipment and services

The table below comprises equipment and services for general use during the decommissioning of all the nuclear facilities.

Equipment/service	Cost MDKK
Equipment for handling and conditioning of decommissioning waste	
Equipment and a large-scale facility for sandblasting	2.0
Frogman area for cutting and packaging of contaminated waste	2.0
Lead cell for trans-shipment of "A-bins", sorting of the contents of selected drums and possibly compression of Hot Cell waste into	6.0
Access to a concrete crushing facility	1.0
Measurement	
γ-scan of units at the "Drum storage", "Centralvejslager" and selected units from the LLW storage. Identification of radium contents	0.5
Interior structure of e.g. "A-bins"	1.0
Equipment for sample extraction (e.g. bore-cores)	0.3
Measurement methods for β-emitters ^{14}C, ^{41}Ca, ^{63}Ni etc.	1.0
Measurement facility for declassification purposes	10.0
Dose meter system	1,5
Transport and storage	
Development and production of concrete container (to hold four waste drums or larger pieces of decommissioning waste). About 1000 pieces	3.0
Truck with powerful crane for handling of waste units	0.4
Intermediate storage facility with gantry crane	3.0
Barn-like construction for declassified items destined for recycling	1.0
Total cost	**32.7**

6.9.3 Research and development of decontamination, radiation measurement and dismantling processes, tools and equipment

To decommission nuclear facilities, radiation protection expertise at a high professional level is needed. Health physics research is therefore necessary to maintain and improve competence, especially within the areas necessary for decommissioning. The research disciplines would include:

- radiation protection philosophy - principles for protection of workers and population
- radiation protection related to decommissioning of nuclear facilities
- radiation protection related to disposal of radioactive waste
- radiation interaction with and transport through matter
- determination of radiation fields and radiation doses from radioactive sources of complex geometries
- assessment of internal radiation doses from uptake of radioactive materials by inhalation or ingestion
- emergency health physics, including protective actions for workers and population after nuclear or radiological accidents
- optimisation of cleanup of areas contaminated with radioactive and non-radioactive materials
- biological effects of ionising radiation, including the assessment of the probability of causation for radiation being the cause of detected cancer diseases

Additional research with special emphasis on decommissioning operations include:

- considerations on actual and future dismantling and decontamination strategies and techniques

- status review to determine the actual positions for:
- free release of slightly contaminated materials
- decontamination with respect to cost savings (optimisation of decontamination)
- comparative work in the world
- development of adapted measurement devices and calculation techniques
- computer codes for radioactive inventory estimations

Research work is deemed necessary also for future recruitment of staff members.

6.9.4 Documentation

A general task of high importance is the retrieval and maintenance of documentation for the facilities and for the work performed. In particular if a facility is decommissioned in stages with a long "silent" period in between, it is important to maintain records of design information and what has been done in the first phase.

In order to secure the knowledge existing about the construction and early days of the facilities, senior staff - and even retired staff - should be consulted and their knowledge recorded. Needless to say, this task should be initiated as soon as possible.

7 Environmental aspects of the decommissioning

7.1 Release of radioactive materials

Information concerning the radiological characteristics of releases is necessary for assessment of the consequences of any release of radioactive materials to the environment. The most important exposure pathways are:

- external radiation from airborne radioactive materials
- inhalation of airborne materials (volatiles, aerosols, particulates)
- external radiation from radioactive materials deposited on ground
- ingestion of contaminated foodstuffs and water
- inhalation of resuspended airborne material

External γ-dose from the plume
In a continuous release of airborne materials to the atmosphere, the activity disperses downwind as a plume. The concentration at ground level at specific distances from the release point will depend on the quantity released, the height of the release point, wind speed, atmospheric stability, heat contained in the release, precipitation on the terrain, physical and chemical form of the released material, and other factors.

Once the radioactive material has reached ground level and is dispersed uniformly throughout a hemisphere several hundred meters in radius with respect to the location of a receptor, the receptor is assumed to be immersed in the cloud. This assumption is usually referred to as the *semi-infinite cloud* approximation. With this assumption, the external γ-dose can be calculated as the product of standard (nuclide specific) dose conversion factors and time-integrated air concentration (exposure integral) calculated by an atmospheric dispersion and transport model.

Inhalation doses from the plume
A person immersed in the plume would inhale an amount of radioactive material proportional to the time of passage of the plume, the person's respiration rate, and the concentration of radioactive material at the person's location. The total radiation dose from inhalation would thus be received over a period of time that could vary from a few weeks to many years, depending again on the chemical and physical nature of the radioactive material and the half-life of the radionuclides.

The dose calculation for inhalation exposure is more straightforward than it is for external exposure, since exposure is directly related to the air concentration at the receptor location as a function of time. The time-integrated air concentration is directly related to the amount of material available for inhalation. Therefore, the total inhaled activity by an individual is obtained by multiplying the breathing rate by the time-integrated air concentration. Given the dose conversion factors for the particular radionuclides of interest, the inhalation dose can be calculated for the total inhaled activity.

Contamination density on the ground
Radioactive material can be deposited on the ground by dry or wet deposition and by gravitational settling of particulates. Dry deposition is caused by material in the near surface layer of the plume being deposited on ground level surfaces by impinging on vegetation, buildings and soil. This process continues to deplete the plume as it travels away from the source, and is parameterised in dispersion models by a dry deposition velocity.

Wet deposition is caused by radioactive material coming into contact with a precipitation system (*e.g.* rain or snow), whereby the radioactive gases and particulates (not the noble gases) are scavenged by the precipitation process and deposited on the ground. This is an efficient scavenging process, which can lead to high ground levels of radioactive materials. Wet deposition is usually parameterised in dispersion models by exponentially depleting the plume and depositing the depleted material on the ground as a function of distance and precipitation rate.

After the material on the ground has been accounted for by each of the processes mentioned above, the dose at a receptor point is calculated by summing the exposure of an individual to each radionuclide deposited on the ground surface.

7.2 Release of non-radioactive materials

The atmospheric dispersion of radioactive and non-radioactive materials does not differ. The exposure of the population from an atmospheric release of non-radioactive materials is by inhalation and by deposited material on ground surfaces. The latter includes internal exposure from contaminated foodstuffs and from inhalation of resuspended materials from the ground.

7.3 Consequences of releases to the environment

Release of radioactive materials
Radioactive materials produced during the operation of nuclear reactors include fission products, transuranics generated within the fuel material itself, and activation products generated by neutron exposure of the structural and other materials within and immediately around the reactor core. The fission products consist of a very large number of different kinds of radionuclides. The amount of these fission products and their potential for escape from their normal places of confinement represent the dominant potential for consequences to the public. Virtually all activation products exist as non-volatile solids. The characteristics of these materials show quite clearly that the potential for releases to the environment decreases dramatically in the order: (1) gaseous materials, (2) volatile solids, and (3) non-volatile solids.

Potential doses and potential surface contamination densities from an atmospheric release of activity during decommissioning has been estimated for the radionuclides present as activation and fission products in Hot Cell, DR 3 and DR 1 as shown in Table 7.3.1. Neutral atmospheric stability (occurring 60% of all time) has been used for these calculations and the exposure and surface contamination relative to these calculated values is shown in Figure 7.3.1.

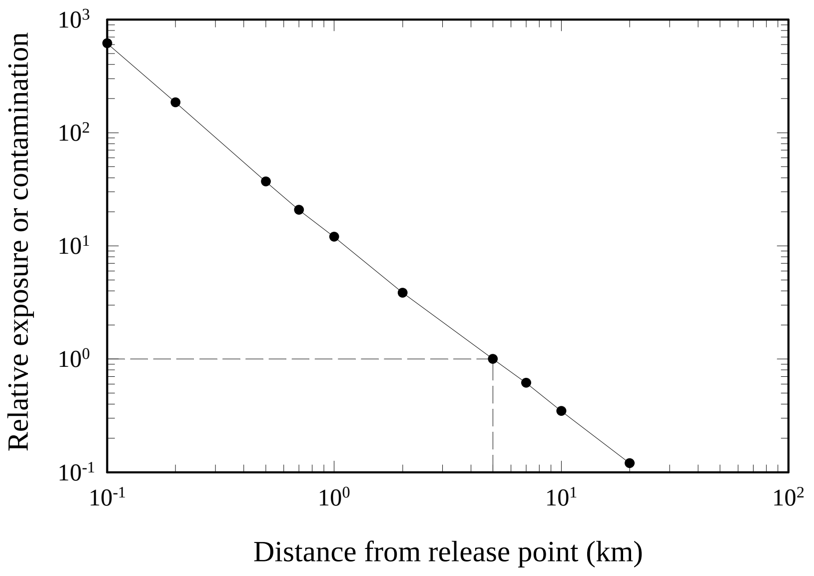

Figure 7.3.1 Relative exposure or surface contamination density as a function of distance from the release point under neutral atmospheric conditions (Pasquill D). The relative exposure and surface contamination density at the distance of 5 km is equal to 1.

As can be seen from Table 7.3.1 an inhalation dose of 10 μSv at a distance of 5 km will require a release of about 0.1 - 10 TBq of the γ-emitting radionuclides and about 1 GBq of the actinides.

Table 7.3.1 Dose and contamination levels at a downwind distance of 5 km from an atmospheric release of 1 Bq of activation products during the most probable meteorological conditions.

<table>
<tr><th rowspan="2">Radionuclide</th><th colspan="3">Dose and contamination levels at a distance of 5 km from the release point</th></tr>
<tr><th>External γ-dose
[Sv/Bq released]</th><th>Inhalation dose
[Sv/Bq released]</th><th>Surface contamination
[$Bq \cdot m^{-2}$/Bq released]</th></tr>
<tr><td>^{3}H</td><td>-</td><td>$3 \cdot 10^{-21}$</td><td>-</td></tr>
<tr><td>^{60}Co</td><td>$4 \cdot 10^{-19}$</td><td>$4 \cdot 10^{-18}$</td><td rowspan="12">$2.4 \cdot 10^{-9}$</td></tr>
<tr><td>^{65}Zn</td><td>$9 \cdot 10^{-20}$</td><td>$9 \cdot 10^{-19}$</td></tr>
<tr><td>^{90}Sr</td><td>-</td><td>$7 \cdot 10^{-17}$</td></tr>
<tr><td>^{137}Cs</td><td>$9 \cdot 10^{-20}$</td><td>$4 \cdot 10^{-18}$</td></tr>
<tr><td>^{238}Pu</td><td>-</td><td>$7 \cdot 10^{-15}$</td></tr>
<tr><td>^{239}Pu</td><td>-</td><td>$7 \cdot 10^{-15}$</td></tr>
<tr><td>^{240}Pu</td><td>-</td><td>$7 \cdot 10^{-15}$</td></tr>
<tr><td>^{241}Am</td><td>$3 \cdot 10^{-21}$</td><td>$7 \cdot 10^{-15}$</td></tr>
<tr><td>^{243}Am</td><td>$6 \cdot 10^{-21}$</td><td>$7 \cdot 10^{-15}$</td></tr>
<tr><td>^{244}Cm</td><td>-</td><td>$6 \cdot 10^{-15}$</td></tr>
<tr><td>^{152}Eu</td><td>$2 \cdot 10^{-19}$</td><td>$2 \cdot 10^{-17}$</td></tr>
<tr><td>^{154}Eu</td><td>$2 \cdot 10^{-19}$</td><td>$2 \cdot 10^{-17}$</td></tr>
</table>

Release of non-radioactive materials

Calculated values of the exposure integral and the surface contamination density from a unit release of non-radioactive materials as particulates are shown in Table 7.3.2. As can be seen from the table inhalation of 1 μg of particulates (breathing rate $2 \cdot 10^{-4}$ m^3/s) at a distance of 5 km will be caused by a release of about 2 kg material as particulates. A release of about 400 g material as particulates will cause a surface contamination density of 1 μg/m^2. Values for other downwind distances can be found from Figure 7.3.1.

Table 7.3.2 Exposure integral and contamination level at a downwind distance of 5 km from an atmospheric release of 1 gram of particulates during the most probable meteorological conditions.

Materials	Exposure integral [g·s·m^{-3}/g released]	Surface contamination [g·m^{-2}/g released]
Particulates	$2.4 \cdot 10^{-6}$	$2.4 \cdot 10^{-9}$

8 Licensing aspects

Apart from the closure and partial dismantling of DR 2 and Hot Cells, decommissioning of nuclear facilities is a new undertaking in Denmark - as in many other countries. Therefore, it must be foreseen that the "conditions for operation" of the various facilities will have to be rewritten more or less from scratch, and that new procedures will have to be considered for the interaction with the authorities. The authorities, as well, will have to reconsider their procedures in supervising the safety at the facilities; but this question is not an issue for the present report.

One subject, which will be of importance when large quantities of material has to be disposed of, is the question of how much of this material need to be treated as radioactive waste and how much can be considered non-radioactive. Therefore, an overview of the international situation regarding this issue has been included in section 8.1.

8.1 Exclusion, Exemption and Clearance

8.1.1 Introduction

The Basic Safety Standards (BSS) from six international organisations (WHO, OECD/NEA, FAO, IAEA, PAHO and ILO) set down requirements for protection against the risks associated with exposure to ionising radiation[18]. These requirements are based, amongst other things, on the recommendations of the International Commission on Radiological Protection (ICRP)[19]. The Standards address **practices** which are human activities that add radiation exposure to that which people normally incur due to background radiation, and **interventions** which are human activities that seek to reduce radiation exposure that is not part of a controlled practice.

As the resources for radiation protection are limited it is necessary to seek a limitation of the System of Radiation Protection. Is it necessary, say, to control all types of practices with the same degree of detail, independently of the risk of the practice? And similarly, how many resources should be spent to reduce de facto exposures of populations, if the risk of those exposures is low? These fundamental questions have been discussed within the international radiation protection organisations ICRP/IAEA for many years and new concepts have been defined, *e.g.*:

(a) exemption
(b) clearance
(c) exclusion
(d) intervention exemption levels

Items (a) and (b) are used for practices or for sources within a practice. Item (c) is used mainly for intervention situations and item (d) only for intervention situations, *e.g.* for foodstuffs moving in international trade. The concepts of exemption, clearance and exclusion are described in the BSS as follows:

Exemption
Schedule I to the BSS provides the following description of exemption "Practices and sources within practices may be exempted from the requirements of the Standards, including those for notification,

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[18] International Atomic Energy Agency. International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. Safety Series No 115, IAEA, Vienna (1996).

[19] International Commission on Radiological Protection. Recommendations of the ICRP, Publication 60, Ann ICRP 21. Pergamon Press, Oxford (1990).

registration or licensing. …..... Exemption should not be granted to permit practices that would otherwise not be justified". Generally, in using the term 'exemption' it is important to state from what the practice, *etc*, is being exempted. The term exemption itself is not defined in the glossary to the BSS.

Clearance
This is defined in the glossary to the BSS as "Removal of radioactive materials or radioactive objects within authorised practices from any further control by the Regulatory Authority". Furthermore, the BSS state that clearance is subject to clearance levels which are "Values, established by the Regulatory Authority and expressed in terms of activity concentrations and/or total activity, at or below which sources of radiation may be released from regulatory control".

Exclusion
Exclusion is described in the BSS as "Any exposure whose magnitude or likelihood is essentially unamenable to control through the requirements of the BSS is deemed to be excluded from the BSS". A typical example is the exposure caused by radioactive elements - like potassium - that are constituents of our body and essential for our normal living. An example of an exposure that is essentially unamenable to control is that due to cosmic rays at ground level.

The exclusion, exemption and clearance concepts are illustrated graphically for waste at Figure 8.1.1.

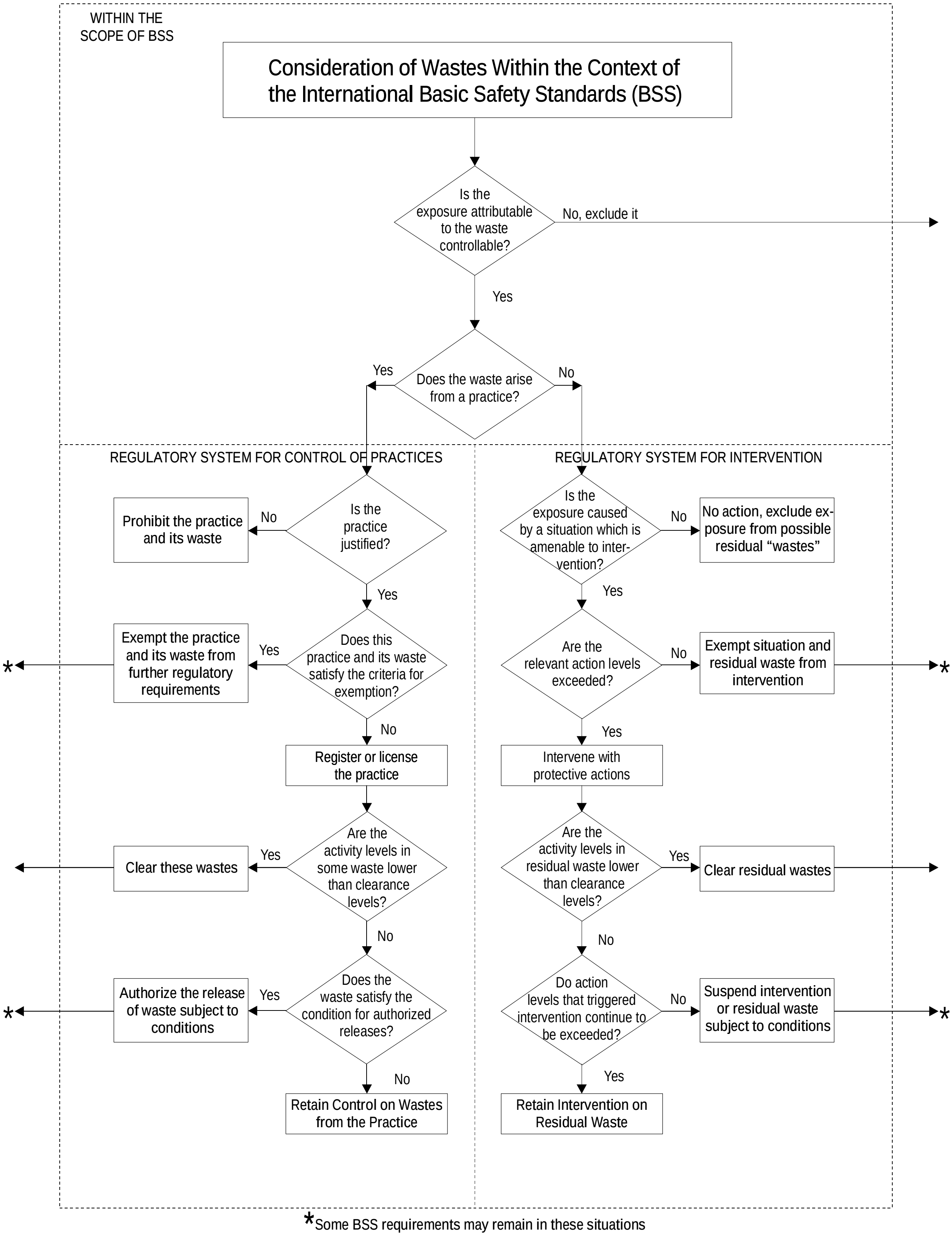

Figure 8.1.1 The concepts of exclusion from regulation, of exemption from regulatory requirements and of clearance. Exemption is used as part of a process to determine a priori the nature and extent of application of the system of registration or licensing of a practice. Clearance is intended to mean exemption a posteriori, i.e., exemption, from within the system, of sources which for one reason or another are under regulatory control and should not continue to be so.

8.1.2 Exemption of a practice

Historically, the area where most work has been done is exemption. It was established early on in the development of the BSS that some practices do not warrant full imposition of the regulatory system. Over ten years ago, the IAEA, jointly with the Nuclear Energy Agency (NEA) of the OECD, set out the following general principles for exemption[20]:

- individual risks must be sufficiently low as not to warrant regulatory concern;
- radiation protection, including the cost of regulatory control, must be optimised; and
- the practice should be inherently safe.

These principles were further developed by IAEA/NEA. The first principle was interpreted as meaning that situations involving trivial risks would not warrant regulatory control (the other conditions being satisfied of course). Comparison with society's response to, and perception of risks from, other activities led to the conclusion that annual risks of death of the order of 10^{-6} to 10^{-7} are generally not of concern to individuals. Using the then current risk factor for fatal cancer, this could be converted to an annual individual dose of around 10 μSv to 100 μSv.

Considerations of doses from natural background radiation and of their natural variations (in Denmark the variation is 2 - 20 mSv per year, including exposure from radon in dwellings) supported the idea that doses in this range could be regarded as trivial. From this range, a value of about 10 μSv per year was proposed on the basis that an individual could be significantly exposed to more than one exempt source. Knowledge of radiation risks has advanced since this IAEA/NEA study in 1988 and it is now believed that the risk to health is greater. However, as it is now considered unlikely that an individual would be significantly exposed to more than one exempt source at a time, the conclusion stands that a level of individual dose, regardless of origin, may be regarded as trivial if it is of the order of 10 μSv per year or less.

Turning to the optimisation principle, IAEA/NEA made the point that a practice could be considered as a candidate for granting exemption if the result of the assessment of optimisation showed that exemption is the optimum radiological protection option. Furthermore, the resources required for regulation were a factor that needed to be considered in the optimisation of protection. IAEA/NEA suggested on cost-benefit grounds, that if the collective dose[21] committed by one year of the unregulated practice was less than around 1 manSv, the total detriment would be low enough to permit exemption without more detailed consideration of other options. This does not mean that a practice giving rise to a larger collective dose could not be exempted; rather it would have to be shown in such cases that exemption is the optimum solution in radiological protection terms. However, the 1 manSv collective dose criterion has, in general, not been a determining factor in the exemption of practices[22].

The dose criteria, together with the requirement for inherent safety, have been accepted internationally as a basis for the exemption of practices from regulatory control. Schedule I of the BSS allows a practice or a source within a practice to be exempted from the requirements

[20] International Atomic Energy Agency. Principles for the Exemption of Radiation Sources and Practices from Regulatory Control. Safety Series No 89, IAEA, Vienna (1988).

[21] The collective dose is the sum of all individual doses for a group of people; the collective dose to the population in Denmark from natural background radiation is about 15000 manSv.

[22] Harvey M et al. Principles and Methods for Establishing Concentrations and Quantities (Exemption Values) Below which Reporting is not Required in the European Directive. Radiation Protection No 65. Doc XI-028/93, European Commission, Luxembourg (1993).

of the BSS, except justification, without further consideration provided that the following criteria are met in all feasible situations:

(a) the effective dose expected to be incurred by any member of the public due to the exempted practice or source is of the order of 10 μSv or less in a year, and

(b) either the collective effective dose committed by one year of performance of the practice is no more than about 1 manSv or an assessment for the optimisation of protection shows that exemption is the optimum option.

These dose criteria, however, may not have immediate practical value; their application would entail an assessment of each candidate practice. The criteria have, however, been turned into radionuclide-specific levels, which can be applied directly[22]. In doing so, the concept of exemption was further refined as follows:

- a practice is taken to be a use of radionuclides for a specific purpose [not considered were industries where large quantities of naturally radioactive ores or materials were being processed but not for their radioactive properties]
- candidate practices involve small-scale usage of radionuclides, *e.g.*, medical research, *etc.* [practices involving large quantities of radionuclides, *e.g.*, nuclear installations, may not be 'inherently safe']
- the dose criteria apply to individuals working in the practice as well as to members of the public exposed incidentally to discharges (this is implied in the IAEA/NEA[20] document)

On the basis of these assumptions, a set of exposure scenarios was constructed and used to derive radionuclide-specific concentrations and total quantities that corresponded to the dose criteria. These derived radionuclide-specific levels are included in Schedule I of the BSS (the same values are also given in Annex I of the Euratom Basic Safety Standards[23]). Their use allows automatic exemption from the requirements of the Standards except that the practice should be justified; exemption should not be invoked to allow frivolous or unwarranted usage of radionuclides. Thus, a practice that is so exempted is not outside the system of radiological protection nor is it outside the scope of a regulatory system. Rather the exemption is from the bureaucratic aspects of a regulatory system. Furthermore, such regulatory involvement should not be required at any stage including disposal of wastes. However, the exposure scenarios used in calculating the radionuclide-specific levels all assumed small scale usage of radionuclides; situations involving large volumes of materials with very low activity concentrations, such as can arise during decommissioning of nuclear installations, were not explicitly considered. If the radionuclide-specific exemption levels are used in these types of situations, doses in excess of trivial levels could theoretically be received (although probably not in excess of the dose limit for members of the public). This fact has provided support for the establishment of the concept of 'clearance' as a separate entity with its own derived radionuclide-specific levels.

8.1.3 Clearance of radioactive materials from a practice

Clearance applies in the case of practices that have not been exempted as described above. Initially, at least, the term was considered to apply to solid materials but recently it has also been used in the context of liquid and gaseous effluents[24]. Essentially, clearance is the release

[23] European Commission. Council Directive 96/29 Euratom, Laying down Basic Safety Standards for the Protection of the Health of Workers and of the General Public against the Dangers arising from Ionising Radiation. Official Journal of the European Communities, L159 (1996).

[24] International Atomic Energy Agency. Clearance of materials resulting from the use of radionuclides in medicine, industry and research. IAEA-TECDOC-1000, IAEA, Vienna 91998).

of materials (wastes, *etc.*) from a regulated practice, with the minimum of regulatory involvement[25].

Practices involving radioactive materials may generate wastes ranging from those that have no additional radioactive content (besides natural radioisotopes) to those that have activity levels so high that special precautions are required for protection. Some of the wastes may be candidates for release to the environment, while others will require isolation in an appropriate facility. Generally, controlled releases of radioactive materials from authorised practices are governed by an authorisation. Such authorisations may have conditions attached to them including, for example in the case of effluent discharges, requirements for environmental monitoring, retrospective assessment of critical group doses, *etc*. The greater the assessed dose to members of the public, the more stringent can be the requirements[26]. It makes sense to define some point on this spectrum where there are no such requirements. Thus, such a point defines clearance. It is the release of materials whose activity level is sufficiently low that any form of post-release regulatory involvement is not required in order to verify that the public is being sufficiently protected. This regulatory involvement could be a requirement for monitoring of the environment or, in the case of solid material, specification of the destination for the discharged material or of the use to which it should be put. Thus, clearance is analogous to the exemption of practices with the difference that clearance only applies to the materials being released by practices. Thus, the dose criteria developed for exemption could equally be applied to clearance.

An alternative interpretation of clearance is that it represents the lower boundary in the definition of radioactive waste. Materials for which no future use is foreseen with activity levels above clearance levels, would be regarded as radioactive waste, whereas materials with activity levels at or below, would not be regarded as radioactive for regulatory purposes.

Clearance levels have now been developed for a number of materials. Within the European Union, the Article 31 Group made recommendations on clearance levels for a number of important radionuclides in metals from the dismantling of nuclear installations[27]. IAEA has developed clearance levels for release of materials from medicine, industry and research[28] and is also developing clearance levels for general application to any solid material for which specific values had not been developed[29]. All of these studies applied the quantitative dose criteria developed for exemption, in particular, the 10 µSv individual dose criterion.

Taking all of these studies into account, for any particular radionuclide, a range of derived values for radionuclide concentrations in materials is often obtained. When compared with the values derived for exemption, there is perhaps a tendency for the clearance values to be the lower. One reason is that much larger quantities of materials are generally taken into account in calculating clearance levels than in deriving exemption levels. There have been some discussions as to whether one set of radionuclide-specific values should be used to allow both exemption of practices and clearance of materials from regulated practices. Such an approach

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[25] Cooper J R. Clearance: Historical Perspectives and Possible Future Developments. IN: Proceedings 2nd Int Conference on Release of Material from Regulatory Control. Hamburg (1999) (ISBN 3-933402-04-2).

[26] International Atomic Energy Agency. Regulatory Control of Radioactive Discharges into the Environment. Draft Safety Guide NS-25.

[27] European Commission. Recommended Radiological Protection Criteria for the Recycling of Metals from Dismantling of Nuclear Installations. Radiation Protection Report No 89. EC, Luxembourg (1998).

[28] International Atomic Energy Agency. Clearance of Materials resulting from the use of Radionuclides in Medicine, Industry and Research. AEA-TECDOC-1000, IAEA, Vienna (1998).

[29] International Atomic Energy Agency. Clearance Levels for Radionuclides in Solid Materials: Application of Exemption Principles. Interim Report for Comment. IAEA-TECDOC-855, IAEA, Vienna (1996).

has the advantage of simplicity; one set of values would be easy to apply and could be interpreted as a definition of a radioactive material for regulatory purposes. There are, however, counter arguments. The values for exemption were derived on the basis of different assumptions and for a different purpose from those derived for clearance. A consequence of choosing one set of values is likely to be selection of the lowest of those available. This may in turn limit their utility for exemption of practices with limited radiological risks. Nevertheless, there may be a case for choosing one set of values for clearance levels: a plethora of levels each specific to a material or industry will lead to confusion. One possibility is to use a specified fraction of the exemption levels.

One area in which agreement is developing, is that for any radionuclide, clearance levels should not exceed exemption levels for, if they do, there is the possibility that cleared materials could re-enter the regulatory system at a later stage.

At the IAEA International Conference on the Safety of Radioactive Waste Management convened in Cordoba, Spain in March 2000, the Chairman of the ICRP stated that "…. if, at the outset, we had realised what a complex system we were going to end up with and had thought of the various possible scenarios, we would probably not have had to make a distinction between exemption and clearance ….. exclusion and exemption are reasonably straightforward; we have criteria for them. However, there are problems with clearance, and perhaps a better term would be 'authorised release' … [and] … the maximum dose does not have to be 10 μSv per year. One authorises discharges of radionuclides in such a manner that, in accordance with ICRP's latest relevant recommendations, the dose to the most exposed members of the public does not exceed 300 μSv per year." He emphasised, however, that we should not be "looking for a single 'magic number' [as] there is a whole spectrum of authorised release, and it is the <u>situations</u> which regulators approve."

8.1.4 Documentation of compliance with clearance criteria

The principles for applying exemption in the case of practices where radionuclides are being used for their fissile, fertile or radioactive properties are well established: the practice should be justified, inherently safe and doses in plausible exposure situations should be trivial. If the practice is exempt, discharges of materials are also exempted from regulatory requirements. In the case of practices that are *not* exempted, for whatever reason, it is possible, provided a specific criterion is met in advance, to discharge material without any subsequent regulatory requirements being enforced. This is clearance. The specific criterion is that the radiological risks from the discharged material are trivial in all plausible circumstances. Thus, the same dose criteria are used for establishing whether practices can be exempted and whether materials can be cleared. Importantly, clearance is part of the general process of authorisation of discharges.

The resources to document that candidate materials for clearance will comply with clearance levels should not be underestimated. Rather expensive measurement equipment to perform γ-spectrometric analyses on bulky materials produced by decommissioning nuclear installations is needed. The necessary manpower to perform analyses of the activity concentration as well as calibrations for many different source geometries would be substantial. The measurements have to be made in a specially equipped laboratory in which the γ-radiation level is close to the natural background level and to which bulky material easily can be transported wrapped in plastic sheets *etc.* to avoid contamination of the equipment and the laboratory.

Even more expensive is to obtain documentation for 'difficult to measure' radioisotopes, the most important being the α-emitting actinides, but also some long-lived β-emitters such as ^{14}C, ^{90}Sr and others. Here sampling of the often heterogeneously distributed activity in materials will be necessary with the inherent uncertainty on representatively of the analysed samples. In some cases radionuclide contents can be estimated from contents of more easily measured isotopes, or - for induced activity - from the material composition and a known exposure to neutron irradiation. From a practical point of view an operational system for clearance of waste materials (and also for declassification of buildings and areas) is important for minimising the volume of material, which have to be disposed of as radioactive. It should be noted that documentation for radioisotope contents is needed also for waste materials placed in a disposal facility, but the accepted levels is higher and often more easily measured.

9 Project management, engineering and site support

The NEA/IAEA/EC list's section 8, " Project management, engineering and site support" has the following sub-headings that have to be addressed: "Mobilisation and preparatory work", "Project management and engineering services", "Public relations", "Support services", "Health and safety" and "Demobilisation". In the present study it has been chosen not to treat these items in detail for assessing the costs but to base the cost assessment on the size of the "Danish Decommissioning" organisation that has been planned for.

9.1 Assessment of general costs

The cost is assumed to be proportional to the number of employees and is based on an extrapolation of the budget of the year 2001. In Table 9.1.1 an estimate is given of the size of the permanent staff for the three scenarios considered in the study.

Table 9.1.1 Number of employees during 5-year periods for the three scenarios

Period	1	2	3	4	5	6	7	8	9	10
20 years	70	70	70	35						
35 years	70	70	20	20	20	20	50	20		
50 years	70	70	20	10	10	10	10	20	50	25

In Table 9.1.2 the estimated annual costs have been summarised. The general cost are made up by Site Security, Inspection and Maintenance of buildings and other services from Risø, all of which are to be negotiated with Risø on a yearly basis. The operational costs cover computer licence and maintenance, stationary, travel costs, etc. Salary for administration includes the Board of Directors, the Secretariat and the Department of Administration. The salaries for the Department of Planning and the Department of Operations, shown in brackets, are not included in the sum, as these salaries are part of the costs for each facility.

Table 9.1.2 Annual general costs and salaries

	General costs Mill. DKK	**Salary costs Mill. DKK**	**Number of staff**
Site Security	0,8		
Inspection and Maintenance	5		
Services from Risø	3		
Operational Costs	3		
Salary Administration		4,0	10
Salary Quality Management		1,0	3
Salary Applied Health Physics		3,2	9
Salary Waste Plant		5,6	16
Salary Planning*		2,8	8
Salary Operations*		8,4	24
Total	**11.8**	**25.0**	**70**

* salary costs included in Operational Tasks

10 Total costs of the decommissioning of all nuclear facilities

During the work on the present project it has been realised that there will probably not be big differences between the three scenarios studied with respect to the protective measures needed for the personnel carrying out demolition work. Even the long scenario with a cooling time of 40 years for DR 3 does not result in a reduction of the ^{60}Co activity in the reactor internals large enough to permit manual manipulation. However, it must be expected that personnel doses from the demolition work will be lower.

10.1 Expenses

Thus, the costs for the three scenarios will be equal in fixed prices, apart from the differences due to expenses for keeping the organisation running for different periods of time and for keeping some facilities in safe storage for the longer scenarios.

The tables and figures shown below give the cost distributions over the three scenarios in five-year periods. For convenience the scenarios have been copied from chapter 5.1. As can be seen clearly from the histograms, the dominant contributor to the total costs in all three scenarios is DR 3. A further refinement of the present cost assessment, therefore, should look at this facility first. During the review seminar it was, for instance, suggested to examine the possible savings by leasing expensive special equipment instead of buying it. Expenses to the small facilities, the Isotope Laboratory and the Fuel Fabrication facility, disappear in the histograms.

Total costs for the three scenarios range from about 1080 to about 1180 million DKK, i.e. on the average about 55 million DKK per year in the years where substantial work is taking place. A previous, shorter, study came to a very rough estimate of 300-600 million DKK for the decommissioning, based on extrapolation of costs experienced by the decommissioning of foreign facilities similar to Risø's. However, in the present project a much more detailed study of the work to be done at the individual facilities has been possible. Furthermore, the costs of running the decommissioning organisation and maintaining the facilities, including the Waste Treatment Plant, throughout the decommissioning period have been included. It is doubtful whether these costs were included in the costs reported from the foreign projects. In addition, the DR 3 costs include 68 million DKK for the transfer of spent fuel to the USA. Thus, there may not be as large a contradiction between the two cost estimates as the figures suggest. Nevertheless, as also stated by the reviewers, a large uncertainty must be expected in the assessment of decommissioning costs, in particular at an early stage.

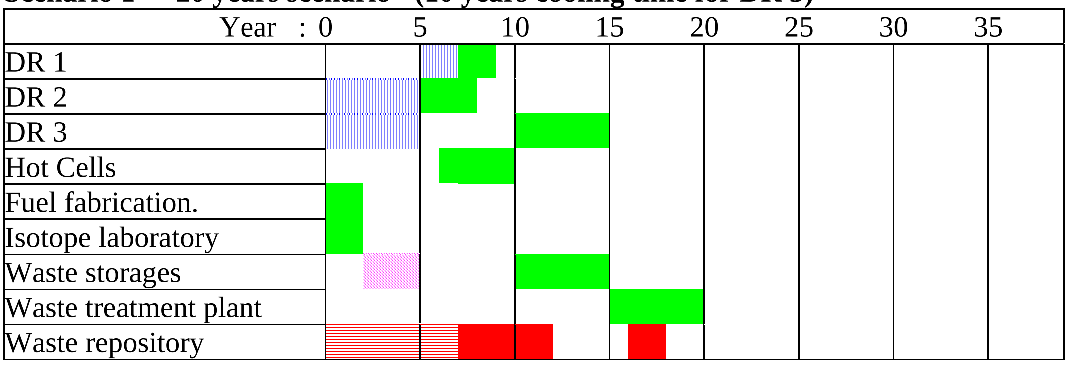
It will be noted that even for the short scenario there is an expense at the Waste Management Plant continuing throughout all ten five-year periods. This is the continuing cost for processing and storing radioactive waste from other places in Denmark, and as such it is not actually related to the decommissioning costs. It has been included here to indicate that this expense continues even after Risø's facilities are gone.

10.1.1 Scenario 1 (20 years scenario)

Table 10.1.1 Costs at 20 years scenario, millions DKK

Period	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**Total**
Year	1-5	6-10	11-15	16-20	21-25	26-30	31-35	36-40	41-45	46-50	
Staff pr year	**70**	**70**	**70**	**35**							
General Costs	59	59	59	29,5	0,0	0,0	0,0	0,0	0,0	0,0	206,5
Administration	20,0	20,0	20,0	10,0	0,0	0,0	0,0	0,0	0,0	0,0	70,0
AHP+QM	21,0	21,0	21,0	10,5	0,0	0,0	0,0	0,0	0,0	0,0	73,5
DR 1	0,0	5,6	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	5,6
DR 2	0,0	27,9	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	27,9
DR 3	181,3	0,0	237,7	0,0	0,0	0,0	0,0	0,0	0,0	0,0	419,0
Hot Cells	0,0	24,7	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	24,7
Fuel Fab.	0,1	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,1
Isotope Lab.	0,2	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,2
Waste Plant	43,7	43,0	44,2	31,6	15,0	15,0	15,0	15,0	15,0	15,0	252,5
Total	**325,3**	**201,2**	**381,9**	**81,6**	**15,0**	**15,0**	**15,0**	**15,0**	**15,0**	**15,0**	**1080,0**
Salary share	125,0	125,0	125,0	62,5	0,0	0,0	0,0	0,0	0,0	0,0	437,5

Scenario 1 - "20 years scenario" (10 years cooling time for DR 3)



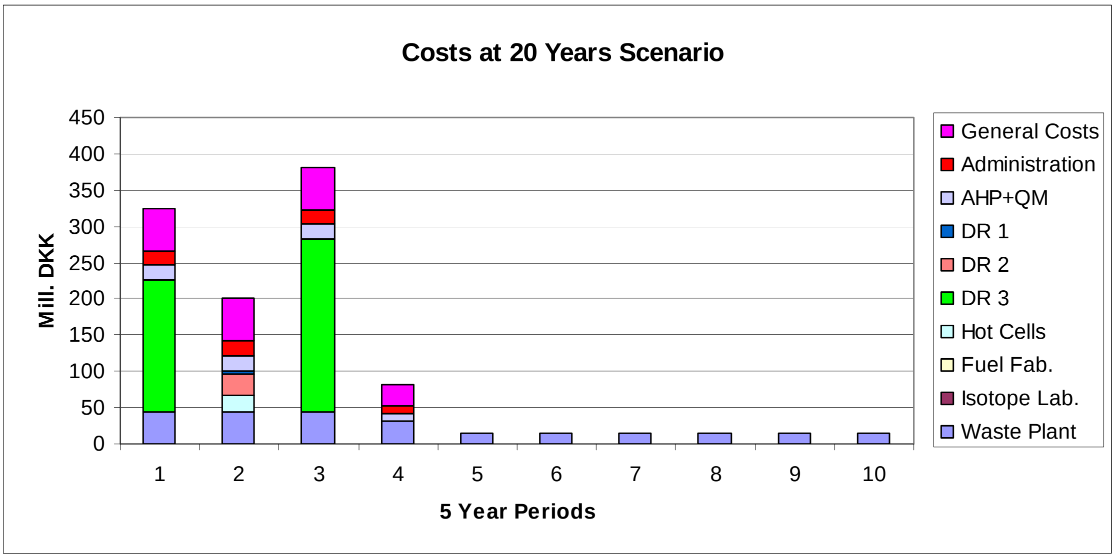


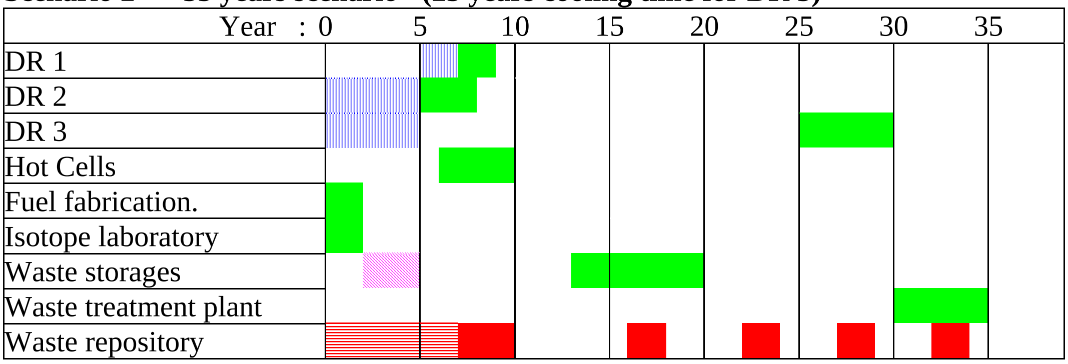
Figure 10.1.1 Costs at 20 years scenario

10.1.2 Scenario 2 (35 years scenario)

Table 10.1.2 Costs at 35 years scenario

Period	1	2	3	4	5	6	7	8	9	10	Total
Year	1-5	6-10	11-15	16-20	21-25	26-30	31-35	36-40	41-45	46-50	
Staff pr year	**70**	**70**	**20**	**20**	**20**	**50**	**20**				
General Costs	59,0	59,0	16,9	16,9	16,9	42,1	16,9	0,0	0,0	0,0	227,6
Administration	20,0	20,0	5,7	5,7	5,7	14,3	5,7	0,0	0,0	0,0	77,1
AHP+QM.	21,0	21,0	6,0	6,0	6,0	15,0	6,0	0,0	0,0	0,0	81,0
DR 1	0,0	5,6	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	5,6
DR 2	0,0	27,9	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	27,9
DR 3	188,2	0,0	0,0	0,0	0,0	234,7	0,0	0,0	0,0	0,0	422,9
Hot Cells	0,0	24,7	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	24,7
Fuel Fab.	0,1	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,1
Isotope Lab.	0,2	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,2
Waste Plant	43,6	43,0	23,4	23,9	23,0	35,0	25,6	15,0	15,0	15,0	262,5
Total	**332,1**	**201,2**	**52,0**	**52,5**	**51,6**	**341,1**	**54,2**	**15,0**	**15,0**	**15,0**	**1129,6**
Salary share	125,0	125,0	35,7	35,7	35,7	89,3	35,7	0,0	0,0	0,0	482,1

Scenario 2 - "35 years scenario" (25 years cooling time for DR 3)



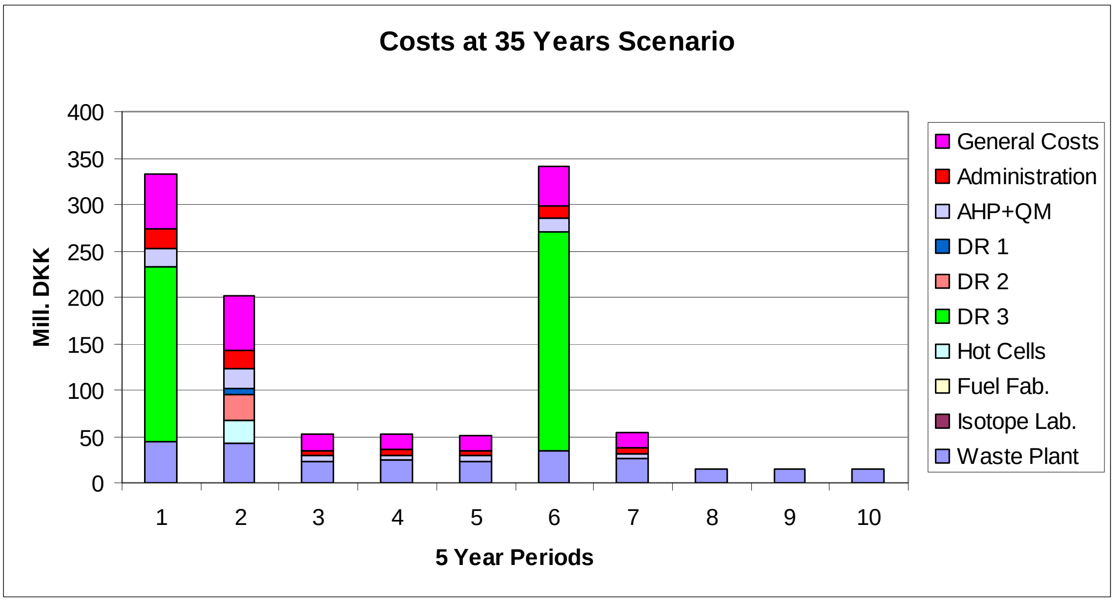


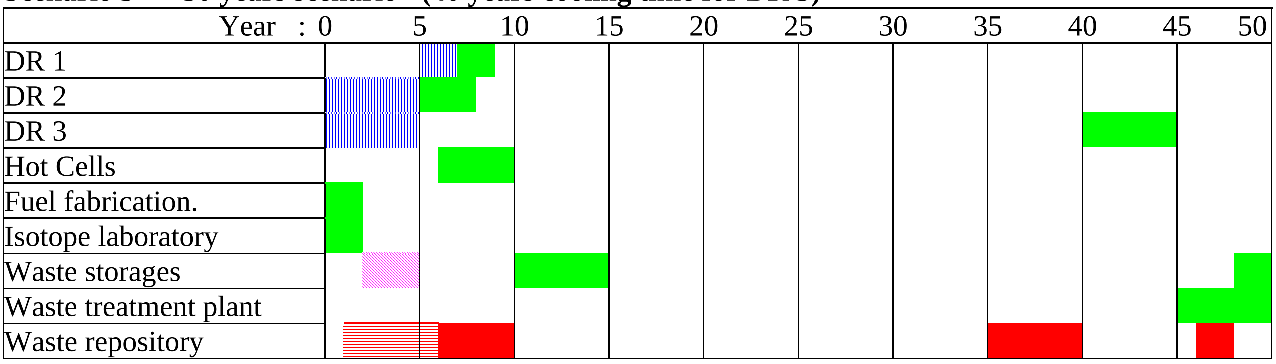
Figure 10.1.2 Costs at 35 years scenario

10.1.3 Scenario 3 (50 years scenario)

Table 10.1.3 Costs at 50 years scenario

Period	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**Total**
Year	1-5	6-10	11-15	16-20	21-25	26-30	31-35	36-40	41-45	46-50	
Staff pr year	**70**	**70**	**20**	**10**	**10**	**10**	**10**	**20**	**50**	**25**	
General Costs	59,0	59,0	16,9	8,4	8,4	8,4	8,4	16,9	42,1	21,1	248,6
Administration	20,0	20,0	5,7	2,9	2,9	2,9	2,9	5,7	14,3	7,1	84,3
AHP+QM	21,0	21,0	6,0	3,0	3,0	3,0	3,0	6,0	15,0	7,5	88,5
DR 1	0,0	5,6	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	5,6
DR 2	0,0	27,9	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	27,9
DR 3	188,2	0,0	0,0	0,0	0,0	0,0	0,0	0,0	234,7	0,0	422,9
Hot Cells	0,0	24,7	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	24,7
Fuel Fab.	0,1	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,1
Isotope Lab.	0,2	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,0	0,2
Waste Plant	43,6	43,0	23,0	19,9	19,0	19,0	19,0	23,0	35,0	28,0	272,5
Total	**332,1**	**201,2**	**51,6**	**34,2**	**33,3**	**33,3**	**33,3**	**51,6**	**341,1**	**63,7**	**1175,4**
Salary share	125,0	125,0	35,7	17,9	17,9	17,9	17,9	35,7	89,3	44,6	526,8

Scenario 3 - "50 years scenario" (40 years cooling time for DR 3)



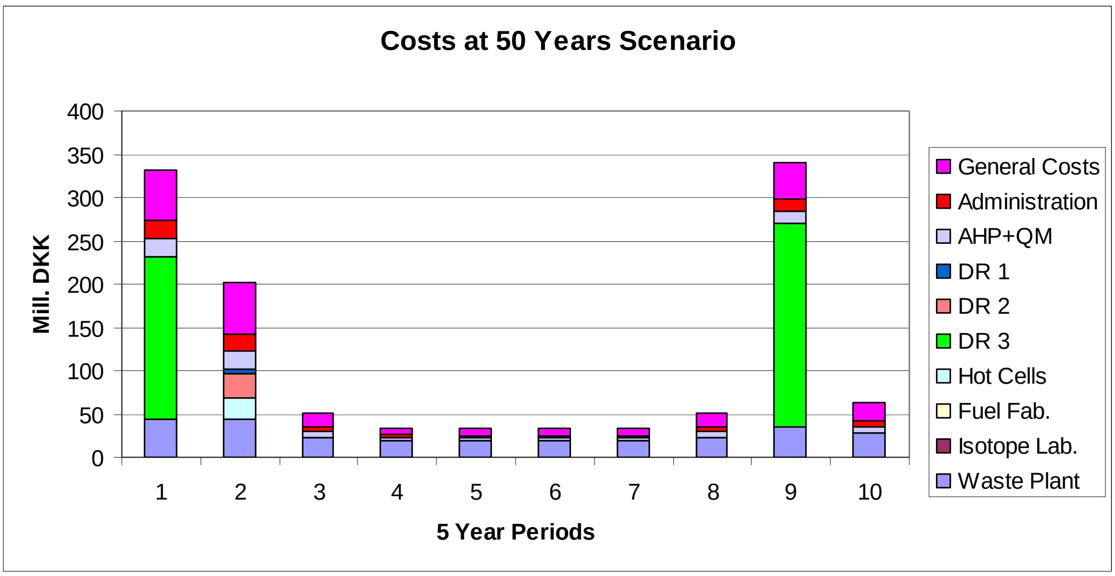


Figure 10.1.3 Costs at 50 years scenario

It can be noted that in scenario 2 and 3 there will be a substantial increase in staff in period 6 and 9, respectively, when the final demolition is going to start. Some of this staff probably will have to be recruited in other countries due to lack of Danish expertise.

10.2 Radiation doses

In the descriptions of the decommissioning activities for the individual facilities rough estimates have been given of the radiation doses to be expected. These expected collective doses for scenario 1 are summarised in Table 10.2.1. It should be stressed that the values are very uncertain at this stage, and that a more precise estimate of the doses requires more precise assessment of the activity contents and the work operations to be performed. In particular for DR 3 the dose estimates are believed to be conservative. Due to radioactive decay the total collective doses for scenarios 2 and 3 would probably be somewhat less than for scenario 1. However, 'hot operations' will in all three scenarios be performed by remote handling, and the effect of radioactive decay on individual doses will, therefore, be only marginal for such operations. For operations not requiring remote handling the effect of radioactive decay would be more pronounced. On the other hand, if operations in scenarios 2 and 3, expected to be performed remote, would be changed from remote to non-remote because this would be considered to be safe due to the reduced activity content, the total collective dose might be higher for scenarios 2 and 3 compared to scenario 1.

Table 10.2.1 Radiation doses from decommissioning of Risø's nuclear facilities. Scenario 1 (20 years). For comparison the collective doses registered at Risø during the last years have been ~150-200 person-mSv per year.

Facility	Estimated total personnel dose person-mSv
DR 1	25
DR 2	100
DR 3	2000
Fuel fabrication	0
Isotope Laboratory	1
Hot Cells	300
Waste storage facilities	70
Waste management plant	0
Sum	**2496**

11 Capacity requirements to a repository for radioactive waste

The waste to be brought to a repository from decommissioning of Risø National Laboratory’s nuclear facilities will be of the low level and intermediate level type. The main activity comes from relatively short-lived radioisotopes (up to 30 years half-life) but some waste will contain long-lived actinides and β-emitters.

11.1 Waste inventory

The volume of the waste to be stored and later placed in a repository depends on shielding requirements as determined by the contents of γ-emitters, and on how efficient it is possible to pack the waste into containers. Here it is assumed for decommissioning waste that the effective mean density is one t/m^3.

Table 11.1.1 Volume estimates and activity inventories for existing waste in storage and waste from decommissioning of the nuclear facilities at Risø National Laboratory

	Volume of conditioned active waste	βγ-activity short-lived (+ tritium)	β-activity long-lived >30 years	α-activity long-lived actinides	Mass of nearly inactive and in-active waste	
	m^3	GBq			tons	
DR1	2	1	low	low	200	+ 1000
DR2	120	60	low	~0	300	+ 600
DR3-complex	1000	13600* (+20000*)	1100*	~0	1800	+ 11000
Fuel fabricat.	1	-	-	low	-	-
Isotope lab.	5	-	low	-	-	10
Hot cells	50	3300	low	100	2500	-
Waste plants	50	1	low	low	100	+ 3600
Waste drums	1800	25000	1000	1000		
Total, about:	3000	42000 (+20000)	2100	1100	5000	16000
UP ore + waste	3400	daughters	-	100	1000	-

*Might well be a factor 2 too high, due to the differences in operating the DIDO reactor and DR3.

Another important area of uncertainty is how much of the nearly inactive part of the demolition waste, which in practice can be declassified and got rid of for reuse or disposal as inactive material. This is estimated by the first figure in the right hand column in Table 11.1.1, while the second figure is an estimate of the amount of ordinary metals and rubble waste from demolition of the buildings themselves.

It is seen that a major amount of active waste is coming from decommissioning of the DR3. Also much of the γ-emitting activity is of this origin. The content of long-lived β-emitters is relatively high while actinides probably are absent. The content of tritium must be taken into consideration.

Waste from the Hot Cells decommissioning contains actinides and the same is the case for part of the waste drums presently kept in storage. The origin of the α-content is also in this case the post-irradiation investigation of spent fuel in the hot cells. Most of the βγ as well as the α-activity is concentrated in a minor number of drums and other waste units. The activity estimates given here correspond to the presence of about 5 kg of spent fuel in these units.

Some of the waste units contain spent sources of various types, e.g. old radium sources from hospitals. Although some may be relatively large they are not expected to contribute significantly to the above indicated activity inventories.

Two types of fissile active waste are special: The spent core solution from DR1, and about 230 kg of spent fuel remaining as sectioned pins etc. after hot cell post-irradiation investigations.

Table 11.1.2 Irradiated fissile materials.

	Weight of fissile material:	βγ-activity short-lived <30 years	β-activity long-lived	α-activity long-lived actinides
	kg		GBq	
DR1 core solution	5	300	1	3
Spent fuel from hot cell investigations	230	1300 000	8000	47 000

Obviously the activity of the core solution from DR1 is not significantly different from the contents in the waste drums mentioned above, and the core might be disposed of together with these after suitable conditioning. However, it might also be sent for processing outside Denmark. Disposal of waste that has been returned from such an operation is not problematic.

The spent fuel from the hot cell investigations is in another class. However, the amount is such that it can be stored without heat generation problems and it is therefore handled and stored as medium level waste. Long-term possibilities for getting rid of this material are discussed elsewhere.

The about 15 m^3 of used heavy water from DR3 contains a considerable amount of tritium, about 3,000,000 GBq. It is not suitable for disposal but may possibly be saleable to other users of heavy water.

The last row in Table 11.1.1 mentions UP-waste, i.e. tailings and unprocessed ore from pilot experiments with uranium extraction from ore from Greenland. The character of these materials is different from the other waste, and they should possibly be disposed of in a different manner.

11.2 Capacity requirements

Only some preliminary concepts for a Danish final repository for low and intermediate waste are available. However, it follows from the volumes given in Table 11.1.1 that it can be rather small. A capacity of the order of 3000 to maximum 10 000 m^3 will be needed, depending on possibilities for declassification etc.

Acceptance criteria for the disposal facility must also be established, and it would be reasonable to take not only activity concentrations, but also the rather small size of the facility into account.

Beside the radioactivity in the waste the disposal facility must also be able to accommodate a certain amount of chemically toxic materials, ranging from some not-irradiated uranium to other heavy metals such as beryllium, cadmium and a quite considerable amount of lead.

Proposals for design, siting, construction and operation of a disposal facility is not a topic for the present report, and no assessment of the cost of disposal is, therefore, attempted. However, the quantitative estimates given in the report for amounts and types of waste produced during decommissioning of the facilities at Risø are very valuable inputs to the coming planning for such a facility.

It should be noted that in addition to holding the waste arising from the decommissioning of the facilities at Risø National Laboratory, the repository must accommodate radioactive waste that keeps coming from other sources, e.g. medical or industrial applications of radioisotopes.

12 Conclusions

In the present study all nuclear facilities to be decommissioned at Risø National Laboratory have been described and assessments have been made of the activity remaining in the facilities. Furthermore, the work operations to be performed have been identified to the detail possible at this stage, and assessments have been made of the costs of performing these tasks. Also the costs of necessary equipment and general costs such as the expenses to health physics and management have been estimated. The total costs for the three scenarios considered lie in the interval 1080-1180 million DKK, corresponding to an average of about 55 million DKK per year in the years where substantial work is taking place.

During the work on the present project it has been realised that there will probably not be big differences between the three scenarios studied with respect to the protective measures needed for the personnel carrying out demolition work. Even the long scenario with a cooling time of 40 years for DR 3 does not result in a reduction of the ^{60}Co activity in the reactor internals large enough to permit manual manipulation. Thus, the costs for the three scenarios will be equal in fixed prices, apart from the differences due to expenses for keeping the organisation running for different periods of time and for keeping some facilities in safe storage for the longer scenarios.

As can be seen clearly from the graphs in chapter 10, the dominant contributor to the total costs in all three scenarios is DR 3. A further refinement of the present cost assessment, therefore, should look at this facility first. In particular concerning a number of procurements in the DR 3 figures, only a rough estimate has been possible at this stage. It is believed, however, that the cost assessments in these cases have been conservative. This assumption is supported by comments from the reviewers who found DR 3 estimates rather high. Nevertheless, as also suggested by the reviewers, a large uncertainty must be expected in the assessment of decommissioning costs, in particular at an early stage.

In addition to the cost estimates, assessments have been made of the amounts of radioactive and non-radioactive wastes arising from the decommissioning work and of the personnel doses incurred by the work. The total (net) volume of waste was estimated to be of the order of 3000 m^3 low- and intermediate level waste (plus 3400 m^3 of uranium ore). In addition, an estimated amount of 20,000 tons of inactive or nearly inactive waste will be produced from building demolition etc. In chapter 11 an overview has also been given over waste types, which require special consideration, and which cannot directly be brought into a repository for low- and intermediate level waste of the type thought of as the most suitable.

The personnel doses have been estimated to about 2.5 person Sv, with 80% coming from the decommissioning of DR 3.

As a general finding it is recommended, for the further planning of the decommissioning tasks at the individual facilities, to involve as soon as possible staff who have a recollection of the construction and operation of the facilities - possibly retired staff. Already during the present study the assistance of retired staff members has proven valuable. Furthermore, the importance of securing the documentation of the facilities should be stressed.

13 Acknowledgements

As reported in chapter 4 visits were paid to the Forschungszentrum Karlsruhe, the Paul Scherrer Institut, and the UKAEA at Harwell. The hospitality and willingness to share their experience with us was great at all three institutes. Sincere thanks are expressed to Mr. Wolfgang Pfeifer and Dr. Peter Klumpp and their colleagues at the FZI, Dr. Roger Andres and his colleagues at the PSI and Dr. John Williams and colleagues at Harwell. In addition, Wolfgang Pfeifer, Roger Andres and John Williams undertook a very qualified review of the draft of this report and provided us with valuable comments. Furthermore, the UKAEA and in particular Mr. Roy Manning and Dr. Ian Warneford are thanked for letting us have their programmes PRICE and SPS for evaluation and for instructing us so well in the use of the programmes.

Last, but not least, the editor wishes to thank all his good colleagues in the project- and steering group for their valuable contributions to writing large parts of the text and reviewing the work as it progressed. Also the information provided by other colleagues at Risø is gratefully acknowledged.

14 Literature

Very much has been written about decommissioning of nuclear facilities, and it will be almost impossible to include all relevant documents in a literature list. Therefore, the list given below should be taken as a list of documents that the authors of the present report have found - and have found useful. A substantial part of the list has been produced during the DR 2 decommissioning study.

In addition to usual articles and books etc. references to a number of sites at the Internet have been included. These sites are very valuable sources of information and can serve as points of departure for further literature search.

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Appendix 1

Cost estimate for DR 1

Risø National Laboratory

PRICE ESTIMATE

Project Management - No Costs Allocated
Implementation Cost KR 5.603k
Waste Costs - No Costs Allocated

21-12-2000

Overall Cost KR5.603k

Local Labour Rate KR 230

"R" Task Factor 25,0

DR1 ESTIMATE REPORT ver. 1

Site: Risø

Definitive List : 110010000 **Estimate Version 001**

Facility Name : DR1 reactor

Stag	Area	WPG	WP	Description	Comp. Code	Complexity	Task Typ	No.	Item Quantity	Total Quantity	Cost (KR)
10											
	0			**Reactor Hall**							
		D1	020	Remove core structure	020	3	R	1	10,00 kg	10,00 kg	2,1k
		D1	020	Remove recombiner	040	3	C	1	25 kg	25 kg	7,8k
		D1	030	Remove graphite blocks	030	1	C	1	2.500 kg	2.500 kg	46,0k
		D1	030	Demolition of irradiated biological concrete	030	2	C	1	200.000 kg	200.000 kg	5.520,0k
		D1	040	Dismantle small bore SS pipework	040	2	M	1	50 kg	50 kg	12,4k
		D1	040	Dismantle small carbon steel pipes	040	3	M	1	100 kg	100 kg	8,3k
		D1	050	Dismantle argon system	050	5	M	1	400 kg	400 kg	6,6k
										Area Total =	**5.603,2k**
										STAGE TOTAL =	**5.603,2k**

Cost estimate, produced by DEMEX Consulting Engineers A/S, for demolition of the DR 1 building

Introduction

This paper gives a budget estimate for the demolition of the building containing the reactor DR1 at "Risø Forskningscenter". The estimate is based on the assumption that all installations and structures connected to the reactor or other nuclear installations are removed before the demolition of the building starts.
The ground floor of the building contains a reactor room, control room, dark room, lavatory and changing rooms, plus a transformer room. The basement contains the counter laboratory and two service rooms with installations for heating, ventilation and air compression.
The floorage is estimated to be 600 m^2. The building was erected in the late 1950s and has a bearing construction made from steel. Bricks and aluminium-plates serve as a climatic screen.
This estimate is based on an inside, as well as an outside, visual inspection, supplemented by a plan of the building.
Figures for waste quantities have been calculated on the basis of the plan, supplemented by DEMEX empirical figures for building waste.
Prices for waste disposal have been calculated on the basis of information from receivers of building waste approved by Roskilde Kommune. Prices for building site, environmental cleaning and demolition of the building are based on DEMEX empirical figures.

Price estimate

Time estimate

It is assumed that a period of one month is needed for the demolition works pertaining to the building. If the building is demolished in connection to similar works in the area, the required time may be decreased.
If the contractor is asked to speed up the process, costs are likely to increase.

Setting up and operating the building site

"Setting up and operating the building site" includes all costs for developing the site with sheds, supply of water and electricity, transport of necessary machinery etc.
The price estimate is based on an initial cost at 25,000 DKK for setting up the building site, and 5,000 DKK/month for the operation of the site. It is estimated that, on average, four persons are needed for the demolition.
Costs connected to the setting up and operation of the site can possibly be reduced if existing facilities in surrounding buildings can be used.

Removal of hazardous waste

Hazardous waste includes asbestos and problematic materials such as PCB, CFC, mercury and other heavy metals. This paper assumes that materials contaminated by radiation are treated in connection to the removal of the reactor.
Oil contamination in the cellar is estimated to be a minor problem that can be treated with normal precautions. The building has not been checked thoroughly for hazardous waste. However, the quantity of hazardous waste is estimated to be low.
The price estimate is based on the normal cost for removing hazardous waste and other problematic materials from the building.

Demolition

"Demolition" includes all costs for demolition, including the disconnection of water and electricity supplies, sewers etc., but excluding waste disposal. The current (2000) price level suggests a cost of 260 DKK per square metre for normal demolition works.

Disposal of demolition waste

Costs pertaining to the disposal of building waste depends on the types of material involved.

Concrete and tile
Assuming the necessary separation at the site, concrete and tile can profitably be recycled. The contractor is often able to dispose of such materials directly at ongoing construction projects.
The cost of disposing of concrete and tile is therefore low and sometimes positive (income for the project). This estimate uses the prices for disposing of these materials at the nearest receiving installation.
The price for disposing of concrete depends on the degree of purity in each cartload. Price parameters are: contents of other materials, size of blocks and the contents of reinforcement. The receiving installation assigned by Roskilde Kommune, Mindstrupgård, estimates a price at 42 DKK/tonne for receiving pure tile and concrete.

Metals
Metals from the demolition will mainly consist of reinforcement and construction steel. Metals are sold at market prices, which change daily. Sold for recycling the steel can be expected to bring in 150-200 DKK/tonne.

Flammable waste
Flammable materials are disposed of for combustion at an accredited installation. At "I/S KARA" the price is approximately 658 DKK/tonne including tax.

Waste for landfill:
Building waste not mentioned in the above is disposed of at an accredited landfill. At "Hedeland Losseplads", the price varies from 600 to 900 DKK/tonne including tax. The price depends primarily on the composition of the waste.

Special Waste
The price for receiving special waste (e.g. hard slabs with asbestos) is approximately 600 DKK/tonne.

Transport
Transport costs will approximately be 2.50 DKK/tonne per kilometre.
The average distance to the different installations is estimated to be 10 km.

Price estimate for the demolition

Building site			30,000
Removal of hazardous waste			5,000
Demolition	600 m^2	260 DKK/m^2	156,000

Disposal	m^3	Tonnes/m^3	Tonnes	DKK/t	DKK	
Bricks	73	1,6	117	42	4,906	
Concrete	360	2,4	864	42	36,288	
Steel and other metals (ex. Aluminium)	8	7,8	62	-200	-12,480	
Aluminium	2	2,7	5	-7,000	-37,800	
Flammable	20	0,5	10	795	7,950	
Hazardous waste	1	2	2	1,050	2,100	
Landfill	65	0,4	26	795	20,670	
Total			1,086		21,634	21,634

	Quantity [tonne]	Price [DKK/ km per tonne]	Distance [km]	
Transport	1,086	2.5	10	27,150

Total [DKK]	**239,784**

In connection with the estimated price the following should be noted:

- Tender of demolition works often reveals very differing prices, and variations up till ± 50% are not unusual. The contractor's current volume of order and/or the prospect of obtaining other works can often explain the variations. With this reservation the estimated price is an expression of the price in a normal market.
- In this estimate the price for the building site is given on the basis that this building is the only ongoing demolition project. If several buildings are demolished simultaneously a price reduction can be expected.
- A considerable reduction in the demolition cost could be expected should it be possible to co-ordinate the work with other demolition works.
- Possible gains from selling installations, such as the crane, have not been taken into account.
- The price estimate is made on the assumption that there is no PCB-oil left in the transformer. If the transformer still contains PCB an additional cost of approximately 10,000 DKK can be expected.
- Finally, it is recommended that 10 % be added to cover unforeseen expenses.

On the basis of the above-mentioned points the tender price is estimated to be between 205,000 and 270,000 DKK.

Appendix 2

Cost estimate for DR 2

Risø National Laboratory

PRICE ESTIMATE

Project Management - No Costs Allocated

ESTIMATE REPORT

Implementation Cost KR 27.928k

Waste Costs - No Costs Allocated

21-12-2000

Overall Cost KR27.928k

Local Labour Rate KR 230

"R" Task Factor 25,0

Site: Risø

Definitive List : 110020000 **Estimate Version 000**

Facility Name : DR 2

Stag	Area	WPG	WP	Description	Comp. Code	Complexity	Task Type	No.	Item Quantity	Total Quantity	Cost (KR)
0											
		0		**Reactor Hall**							
		D1	020	Removal of Pb + Al + SS from thermal column	020	3	C	1	4.170 kg	4.170 kg	48,0k
		D1	020	Removal of boron plate from thermal column	020	3	C	1	170 kg	170 kg	2,0k
		D1	020	Removal metal in lower part of reactor shield	020	2	C	1	7.900 kg	7.900 kg	81,8k
		D1	030	Removal of graphite from thermal column	030	1	C	1	4.600 kg	4.600 kg	84,6k
		D1	030	Removal of moving concrete door in thermal column	030	2	C	1	15.000 kg	15.000 kg	414,0k
		D1	030	Removal of lower concrete shield + igloo	030	2	C	1	400.000 kg	400.000 kg	11.040,0k
		D1	030	Removal of roof, floor and walls in hall incl. basement	030	2	C	1	460.000 kg	460.000 kg	12.696,0k
		D1	040	Removal of hold-up tank + pipework	040	3	M	1	3.000 kg	3.000 kg	248,4k
		D1	072	Removal of crane	071	5	M	1	20.000 kg	20.000 kg	184,9k
		D1	083	Removal of upper concrete shield	030	3	C	1	80.000 kg	80.000 kg	3.128,0k
										Area Total =	**27.927,6k**

Appendix 3

Cost estimate for DR 3

DECOMMISSIONING OF REACTOR DR3

Cost Items

Based on:
NEA/IAEA/EC "A PROPOSED STANDARDISED LIST of ITEMS FOR COSTING PURPOSES in the DECOMMISSIONING OF NUCLEAR INSTALLATIONS - *Interim Technical Document*", 1999

01 PRE-DECOMMISSIONING ACTIONS

(See item 08)

02 FACILITY SHUTDOWN ACTIVITIES

02.0100 Plant shutdown and inspection

02.0101 Termination of operation, plant stabilisation, isolation and inspection
At the end of the last operation period, the reactor has been brought to a normal shutdown in accordance with normal shutdown procedures.
The site-staff was at that time still at the normal operational level in number and capacity.
After shutdown, the operations staff should make and document a survey of all systems necessary for the coming prolonged cooling period.
Cost: 50 man weeks.
During the “cooling period” the operational staff performed continuous surveillance of the plant All parameters indicating radiation levels, contamination, cooling flows, temperatures etc. were monitored and surveyed.

02.0102 Facility reuse
Identification, isolation and conservation of systems to be reused should be performed.
Essential systems like power-supply, water and ventilation can be useful in later stages
Of the decommissioning period
Roads, plantings, buildings and other infrastructure elements should be evaluated for later uses.
Cost: 10 man weeks.

02.0200 Removal of fuel and/or nuclear-fuel materials

02.0201 Defuelling and transfer of fuel to temporary spent-fuel storage
All the fuel-elements from the reactor core have been transferred to the external storage block (outside the reactor hall).
After a cooling period in the external storage block, the spent fuel-elements should be transferred to the fuel-element cutting-pool, where the uranium containing parts of the elements are cut out, and transferred to the intermediate storage block.
Cost: 50 man weeks.

02.0202 Nuclear-fuel material inventory recovery
The nuclear fuel material is recovered in connection with the reprocessing of the spent fuel-elements, and the value is included in the economic transactions connected with the agreement concerning the transfer of spent fuel elements to USA. No special expenditures are foreseen in this connection.

02.0300 Drainage and drying or blow down of all systems not in operation
Drainage, blow down and drying of systems no longer in use should be carried out by the operations staff.

Cost: 4 men 4 weeks = 16 man weeks

02.0400 Sampling for radiological inventory characterisation after plant shutdown, defuelling and drainage and drying or blow down of systems
02.0401 Sampling for radiological inventory characterisation in the installations after plant shutdown, defuelling and drainage and drying or blow down of systems
02.0402 Subgrade soil sampling and monitoring wells to map contamination plumes.
Cost: Included in health physics-cost, cf. main report section 6.9 and Appendix 5.

02.0500 Removal of system fluids (water, oils, etc.)
This headline includes removal of all non-active fluids.
The largest volume (about 90 m3) is present in the secondary cooling system (system 04).
The system contains inhibited freshwater that should be drained to the sewage system.
Apart from the above mentioned, only small amounts of lubricating oils from pumps etc. are present (about 200 l).
Cost: 2 man-weeks.

02.0600 Removal of special system fluids (D_2O etc.)
The heavy water from the primary circuit (about 10 m^3) should be drained and filled on stainless steel drums for storage. The water contains Tritium. Further about 5 m^3 of tritiated heavy water for supply purposes are stored in stainless steel drums at the plant.
Cost: 20 man-weeks.

02.0700 Decontamination of systems for dose reduction
The primary D_2O-system may be slightly contaminated, and therefore some action with regard to decontamination may be beneficial.
(Cf. remark in Appendix 5.)
Cost: 12 man-weeks

02.0800 Removal of waste from decontamination
Considered as an ongoing process during the decommissioning period. The cost is contained under the specific item.

02.0900 Removal of combustible material
Task that comprises following:
- Small amounts of lubricating oil, diesel fuel.
- Removal of cables and electrical cabinets.

Cost: 110 man-weeks.

02.1000 Removal of spent resins
Task comprising transfer of about 50 l of spent resin from the primary circuit ion-exchanger

02.1100 Removal of other waste from facility operations
Task including small amounts of contaminated water, unused resins, paper etc.
Costs: 20 man-weeks.

02.1300 Asset recovery: Resale/transfer of facility equipment and components as well as surplus inventory to other licensed (contaminated) and unlicensed (non-contaminated) facilities
Relevant spare parts will be sold if possible to the operating sister reactors of DR 3.
No net income is calculated under this point.

03 PROCUREMENT OF GENERAL EQUIPMENT AND MATERIAL

03.0100 General site dismantling equipment

Investment and maintenance for general site-dismantling equipment, including installation, testing, licensing (operational activities to be included in decommissioning activities however), essentially comprising lifting gear, such as:
- Overhead cranes, trucks etc.

Costs: 5 million Kr
- Hiring of special lifting gear.

Costs: 3 million Kr

03.0200 General equipment for personnel/tooling decontamination
Investment for and maintenance and subsequent dismantling of additional equipment for personnel and/or tooling decontamination, including installation, testing, licensing (operational activities included in decommissioning activities, however).
Costs: 1 million Kr

03.0300 General radiation protection and health physics equipment
Included in Applied Health Physics, cf. main report section 6.9 and Appendix 5.

03.0400 General security and maintenance equipment for long-term storage
Equipment required for the surveillance of facilities either pending decommissioning or being partially dismantled, in view of minimising personnel costs during long-term storage, including installation, testing, licensing (operational activities are included in decommissioning activities, however).
Costs: 5 million kr
Security fences, including installation, testing, licensing (operational activities are included in decommissioning activities).
Costs: 5 million kr.

04 DISMANTLING ACTIVITIES

Section 04 covers all activities relating to the different actual dismantling operations.

04.0100 Decontamination of areas and equipment in buildings to facilitate dismantling
Decontamination of areas and equipment in all buildings to be placed into dormancy outside the containment building.
Outside the containment only minor tasks are considered necessary in order to decontaminate, as all these areas during operation continuously were kept in a decontaminated state. Only a throughout survey and monitoring for radioactivity should be carried out
Costs: Included in Applied Health Physics, cf. main report section 6.9 and Appendix 5.

04.0200 Drainage of spent-fuel Cutting-pool and decontamination of linings
Drainage should be done to the active water effluent system for further processing at the radioactive waste treatment plant.
Erection of plastic tent above spent-fuel pool to contain loosened contamination in the following high pressure rinsing sequence.
Costs: 10 man weeks
Decontamination of the stainless steel surfaces of the spent-fuel pool (high pressure- rinsing/washing).
Costs: 30 man weeks.

04.0300 Preparation for dormancy

04.0301 Zoning for long-term storage
Layout of dormancy control area is considered to encompass the following buildings in the DR3 complex: containment- building, the active handling bay and the office building.

To minimise the radiological controlled area requiring service and maintenance the former service building's ventilation, electrical power supply, and secondary system pump house should be decommissioned.
Decommissioning of the above mentioned buildings and systems.
Costs: 2.5 million kr. (ref.to report from DEMEX Rådgivende Ingeniører A/S)

Reorganisation, cleaning and decontamination operations in rooms, areas and equipment in order to release buildings not to be used in subsequent decommissioning period (phase 3).
Costs: 1,0 million kr.

Redesignation, and re-engineering of materials and equipment for minimum configuration, including:

- Redefinition of secured area.	Costs: -
-Modification or reconstruction of security.	Costs: 1 million kr.
-Surveillance and monitoring equipment.	Costs: 1 million kr
Costs total under this point :	2 million kr.

Costs under this point could be avoided in the 10 year decommissioning scenario.

Deenergising and isolation of non-essential systems
Costs: 1 million kr.

Replacement of equipment and systems used during operations, with more efficient or less complex services, such as:

- Ventilation	Costs: 2 million kr
- Lightning	Costs: 1 million kr
- Access control, etc.	Costs: 2 million kr.
Costs total under this point:	5 million kr.

Costs under this point could be avoided in the 20-year scenario.

04.302 Removal/ disposition of inventory not suitable for long–term storage

About 15 tons of tritiated heavy water should be sought sold. Perhaps to operating heavy-water moderated power reactors.
Cost: no net income/expenditure is calculated.

The stock of irradiated experimental facilities rigs, liners, spectrometers should, after suitable size reduction be loaded into storage casks and transferred to medium active repository.
Costs: 4 million kr.

04.0400 Dismantling and transfer of contaminated equipment and material to containment structure for long-term storage

It is not planned to bring any considerable amounts of contaminated equipment into the containment-building. Peripheral equipment and structures should be decontaminated as far as possible followed by release to non-radioactive scrap.

04.0500 Sampling for radiological inventory characterisation in the installations after zoning and in view of dormancy

Costs: Included in Applied Health Physics, cf. main report section 6.9 and Appendix 5.

04.0600 Site reconfiguration, isolating and securing structures

Site reconfiguration, isolating and securing structures are relevant activities only in the 25- and 40 years scenarios.

04.0601 Reconfiguration and maintenance of essential services and facilities
Primarily administrative arrangements considering agreements between Risø and DD.

to support long-term storage and/or decommissioning operations
Modifications to include process and electrical systems for:

- Remote operation and monitoring (alarms)
- Fire detection
- Reduced maintenance

Costs: 5 million kr.

04.0602 Site boundary reconfiguration
Physical reconfiguration of boundary to meet security requirements of dormancy.
Fences, electronic sensors, cameras etc.
Costs: 2 million kr

04.0603 Reuse of existing buildings construction of temporary enclosures, stores to support site remediation.
Costs: 10 million kr.
04.0604 Stabilisation of radioactive and hazardous waste pending remediation
Costs: 2 million kr

04.0700 Facility (controlled area) hardening, isolation or entombment
These activities are only relevant in the 35- and 50 year scenario.

04.0800 Radiological inventory characterisation for decommissioning and decontamination
Costs: Included in Applied Health Physics, cf. main report section 6.9 and Appendix 5.

04.0900 Preparation of temporary waste storage area.
I is foreseen that a temporary waste storage of 400 m^2 is needed. A covered building for such a purpose should be erected.
Costs: 3 million kr.

04.1000 Removal of fuel-handling equipment
Removal and extraction of fuel handling equipment and associated components from a temporary storage area, pool, storage blocks, viewing-boxes etc.
Not including decontamination (for that see 04.2200).
Costs: 60 man-weeks.

04.1100 Design, procurement, and testing of special tooling/equipment for remote Dismantling.

04.1101 Design and procurement of special tools for dismantling the reactor
vessel and internals

Considerations about dismantling alternatives for the reactor vessel and internals by cutting or by removal and disposal of the vessel and internals as a single package for which design and procurement of special tooling would be associated with the preparation , rigging and lifting of the package as well as shielding, weather enclosure, cradles etc., needed to transport the vessel and internals.
Cost: 80 man-weeks

Considerations about working environment an constraints such as handling requirements, packaging requirements, transportation requirements etc.
Cost: 16 man-weeks

Design of tooling, including:

- 3-D modelling of the workpiece	Cost: 48 man-weeks
- Simulations	- : 40 -

- Research an development expenditures	- :	80 -
- Consultants	- :	100 -
- Use of mock-ups	- :	120 -
- Scale models and demonstrations	- :	48 -
- Manufacturing processes	- :	200 -

Generation of design specifications, including specifications for:

- Operating and maintenance procedures	Cost :	80 man-weeks
- Hardware and software	- :	1 million kr
- Installation	- :	12 man-weeks
- Testing	- :	48 -
- Performance measurement	- :	20 -

Procurement/adaptation of tooling for remote disassembly/segmentation of complex geometric sub-components (highly automated articulated manipulators, handling equipment for remote segmentation).
Cost: 10 million kr

Procurement/adaptation of:
- Remote- viewing systems for underwater operations
- Feedback and control systems for underwater operations
- Turntables for underwater operations
- Support systems for maintaining water clarity
- Support systems for collecting cutting fines
- Support systems for controlling or eliminating the formation of explosive mixtures of gases generated in the cutting or from the dissolution of water.

Cost: 20 million kr

04.1102 Design and procurement of special tools for dismantling other components or structures
- Design of special tooling, the adaptation, modification of existing tooling for dismantling when manual disassembly is not practical or when commercially available tooling is not suitable.
- Leasing of special tooling, including concrete coring, sawing or other cutting devices, demolition hammers, etc., which, if purchased would be cost- prohibitive for smaller applications.

Cost: 20 million kr.

04.1200 Dismantling operations on reactor vessel and internals

Preparation of the work area for dismantling, extracting and packaging the waste for disposal
Cost: 40 man-weeks.

Installation of handling devices and protection systems.
Cost: 40 man-weeks.
Disconnecting of reactor vessel after retraction of horizontal liners.
Cost: 60 man-weeks.
Monitoring of the disassembly.
Cost: Included in Applied Health Physics, cf. main report section 6.9.

Operation of the segmentation tooling.
Cost: 40 man-weeks
Maintenance and change-out of support equipment (purification and ventilation filters etc.)
Cost: 8 man-weeks
Accepting and preparation of the disposal containers as well as loading and processing containers for transport.
Cost: 40 man-weeks
Removal of handling devices and protection systems.
Cost: 40 man-weeks

04.1300 Removal of primary and auxiliary systems

04.1301 Removal of primary and auxiliary systems in reactor facilities
De-energising, disconnecting, disassembly and segmentation of primary an auxiliary equipment 01-system inside D_2O -room (heat exchangers, vessels, pumps, pipes etc.)
Cost: 2,0 million kr (Amount from" UKAEA preliminary price estimate for DMTR Dounreay")
Removal, rigging/transport of the material to a local or centralised waste packaging/preparation area.
In this connection it has to be considered if the existing Hot Cells facility should be reopened or a new temporary hot cell facility should be erected in order to be able to handle radioactive components from the decommissioning of the other nuclear facilities.
Cost: 50 man-weeks (without reopening cost for Hot Cells)

04.1302 Dismantling and removal of contaminated equipment, piping, liners and internal systems outside D_2O –room but inside containment.
De-energising, disconnecting, segmentation of primary and auxiliary systems outside D_2O-room but inside containment:
- 02 He-system
- 03 Graphite CO_2 system
- 04 Secondary cooling system (inside containment)
- 05 Shield cooling system
-
- 09 Equipment for active handling incl. Internal Storage Block
- Cost: 8,1 million kr (Amount from "UKAEA preliminary price estimate DMTR
- Dounreay ").
- 10 Control elements (Coarse Control Arms, Safety Rods, Fine Control Rod)
- 11 Industrial instrumentation
- 12 Nuclear instrumentation
- 13 Interlock- and safety instrumentation
- Cost (11+12+13) : 4,5 million kr (Amount from" UKAEA preliminary price estimate DMTR Dounreay ").
- 14 Ventilation systems
- Cost: 47,3 million kr (Amount from " UKAEA preliminary price estimate DMTR Dounreay" . The amount appears very high. Perhaps due to high contamination levels of the ventilation system of the Dounreay plant).
- 15 Electrical power supply
- 16 Experimental tubes
- 25 He-collection system
- 29 Pressurised air system
- 30 Common shared equipment for irradiation experiments
- 42 –43-44 Isotope irradiation facilities, Si- irradiation facility , Mo- prod. facility
- 68 Water filled neutron scattering plug
- 81 High-temperature rig
- 90 Common shared equipment for beam experiments
- 161 Cold neutron source and neutron guide tube

Cost: 15 men during 60 weeks = 900 man-weeks

04. 1400 Removal of biological shield, secondary structures and containment.

Dismantling of the activated portion of the biological shield, including removal of the top shield, graphite reflector, boron shield, water-cooled lead shield, and surrounding steel tank.
This perhaps is the most intricate decommissioning job in the whole decommissioning programme, due to the fact that the steel tank constitutes a very intense and extended radiation source, which at the same time is moulded into the concrete of the biological shield.

It is considered necessary with implementation and extended use of remote operated heavy robots and other remote handling equipment.
Cost: 80 million kr !!
(Amount from "UKAEA preliminary price estimate for the decommissioning of the test reactor DMTR at the Dounreay Site).
Alternatively it should be evaluated if a removal of the reactor block (about 800 tons) as a whole should be possible. Possibly the weight could be reduced by suitable "slicing" of the biological shield.

04.1500 Removal of other material/equipment from containment structure and all other facilities, or removal of entire contaminated facilities

Removal of contaminated and activated materials (including structural), exceeding release levels from:
- Containment : minor task
- Other systems:
  The systems that are expected to contain considerable amounts of contamination are the active ventilation system, the active waste water systems and the systems in the active Handling Bay (AH-hallen).

Cost: Active vent. System-	5 men in 16 weeks= 80 man-weeks
Active waste water system-	10 men in 40 weeks = 400 man-weeks
Active Handling Bay (storage facilities)-	10 men in 60 weeks = 600 man-weeks
Total	1080 man-weeks

04.1600 Removal and disposal of asbestos

Not considered as a specific problem (although some quantities of "Eternit"- building plates are present)

04.1700 Removal of pool linings
No reactor –pool is present. Removal of pool linings in the AH- bay pool is evaluated above.

04.1800 Building decontamination
04.1801 Removal of contamination from areas and structures in all buildings and stacks
It is not expected that contamination is present in structures and buildings. Surveys should be carried out inside the stack and around active wastewater tanks.
Cost: 10 man-weeks.

04.1802 Removal of embedded pipes in buildings
Cost: 200 man-weeks.
04.1803 Removal of structures/facilities to gain access to radionuclides that may have breached design boundaries
Cost: no significant.

04.1900 Environmental cleanup
04.1901 Removal of embedded pipes outside building
Cost 20 man-weeks.
04.1902 Removal of structures/facilities to gain access to radionuclides that may have breached design boundaries
Not expected
04.1903 Removal of contamination from areas and structures outside all buildings and stacks
At present no contamination exists outside buildings and stacks.
During the decommissioning period some contamination may arise, caused by the many necessary transports of radioactive material. Some efforts may be necessary to remove this final contamination.

Cost: 10 men in 8 weeks = 80 man-weeks

04.2000 Final radioactivity survey
Described in main report under item 6.9.

04.2000 Final Radioactivity survey
Described in Appendix 5.

04.2100 Characterisation of radioactive materials
Described in Appendix 5.

04.2200 Decontamination for recycling and reuse
Cost: 2 men during 40 weeks during 10 years = 800 man-weeks

04.2300 Personnel training.
Cost: 50 men during 8 weeks during 10 years = 4000 man-weeks

04.2400 Asset recovery: Sale/transfer of metal or materials, and salvaged equipment or components for recycling or reuse
No asset recovery is calculated.

05 WASTE PROCESSING, STORAGE AND DISPOSAL

Costs for a final disposal facility are not included in this document.

05.0100 Waste processing, storage and disposal safety analysis
Costs: 5 man-weeks (excluding final disposal analysis).

05.0400 Processing of system fluids from facility operations
It is assumed that the waste can be treated by already existing treatment plants.
The foreseen costs for continuous operation of the plants as they operates today
3-4 million kr/year

05.1200 Processing of decommissioning waste
Waste treatment plant will deliver 1000 concrete containers.
Cost: 3 million kr.

06 SITE SECURITY, SURVEILLANCE AND MAINTENANCE

As long as working operations with free radioactive material are carried out at the site it is considered necessary to maintain surveillance with guards and automatic monitoring systems.
Cost: Included in management costs.

07 SITE RESTORATION, CLEANUP AND LANDSCAPING

07.0100 Demolition or restoration of buildings
07.0101 Dismantling of “balance-of-plant” systems and building components
This comprises the turbine-generator group and condenser system, which is not relevant to DR3.

07.0102 Dismantling of the structure
Primarily the reactor hall and adjacent auxiliary buildings
Cost: 15 million kr (amount from” UKAEA preliminary price estimate DMTR Dounreay”)

07.0103 Dismantling of the stack
Cost: 0.3 million kr

07.0200 Final cleanup and landscaping
Cost: 10 million kr

07.0300 Independent compliance verification with cleanup and/or site-reuse standards
Cost: 1 million kr
07.0400 Perpetuity funding/surveillance for limited or restricted release of property
Cost : not considered.

08 PROJECT MANAGEMENT, ENGINEERING AND SITE SUPPORT

See main report.

09 RESEARCH AND DEVELOPMENT

See main report.

10 FUEL AND NUCLEAR MATERIAL

The filled spent fuel element transport flasks should be transported abroad to reprocessing and final storage of the high radioactive inventory.
Cost: 40 man-weeks + transportation cost- 50 million kr.

The unused fuel elements should be transported and " sold "abroad.
Cost: 8 man-weeks + transportation cost- 18 million kr.

11 OTHER COSTS

See main report.

Cost estimate - DR 3

Labour cost per hour: 231
Labour cost per week: 8556

	Crew	Weeks	Man-weeks	Labour costs	Procurement	Sum 3	Sum 2	Sum 1	Remarks
01 PRE-DECOMMISSIONING ACTIONS	0		0	0		0		0	
01.0100 Decommissioning planning	0		0	0		0	0		
01.0101 Strategic studies	0		0	0		0			
01.0102 Conceptual planning	0		0	0		0			
01.0103 Detailed planning	0		0	0		0			
01.0104 Safety and environmental studies	0		0	0		0			
01.0200 Authorisation	0		0	0		0	0		
01.0201 License applications and license approvals	0		0	0		0			
01.0202 Public consultation and public inquiry	0		0	0		0			
01.0300 Radiological surveys for planning and licensing	0		0	0		0	0		
01.0400 Hazardous-material surveys and analysis	0		0	0		0	0		
01.0500 Prime contracting selection	0		0	0		0	0		
02 FACILITY SHUTDOWN ACTIVITIES	0		0	0		0		2481313	
02.0100 Plant shutdown and inspection	0		0	0		0	513375		
02.0101 Termination of operation, plant stabilisation, isolation and inspection	5	10	50	427813		427813			
02.0102 Facility reuse	1	10	10	85563		85563			
02.0200 Removal of fuel and/or nuclear-fuel materials	0		0	0		0	427813		
02.0201 Defuelling and transfer of fuel to temporary spent-fuel storage	5	10	50	427813		427813			
02.0202 Nuclear-fuel material inventory recovery	0	0	0	0		0			
02.0300 Drainage and drying or blow-	4	4	16	136900		136900	136900		

down of all systems not in operation									
02.0400 Sampling for radiological inventory characterisation after plant shutdown, defuelling and drainage and drying or blowdown of systems	0		0	0		0	0		
02.0401 Sampling for radiological inventory characterisation in the installations after plant shutdown, defuelling and drainage and drying or blowdown of systems	0		0	0		0			
02.0402 Subgrade soil sampling and monitoring wells to map contamination plumes	0		0	0		0			
02.0500 Removal of system fluids (water, oils, etc.)	1	2	2	17113		17113	17113		
02.0600 Removal of special system fluids (D_2O, sodium, etc.)	2	10	20	171125		171125	171125		
02.0700 Decontamination of systems for dose reduction	3	4	12	102675		102675	102675		
02.0800 Removal of waste from decontamination	0		0	0		0	0		
02.0900 Removal of combustible material	5	22	110	941188		941188	941188		
02.1000 Removal of spent resins	0		0	0		0	0		
02.1100 Removal of other waste from facility operations	2	10	20	171125		171125	171125		
02.1200 Isolation of power equipment	0		0	0		0	0		N/A
02.1300 Asset recovery: Resale/transfer of facility equipment and components as well as surplus inventory to other licensed (contaminated) and unlicensed (non-contaminated) facilities	0		0	0		0	0		
03 PROCUREMENT OF GENERAL EQUIPMENT AND MATERIAL	0		0	0		0		19000000	
03.0100 General site dismantling equipment	0		0	0	8000000	8000000	8000000		
03.0200 General equipment for person-	0		0	0	1000000	1000000	1000000		

nel/tooling decontamination									
03.0300 General radiation protection and health physics equipment	0		0	0		0	0		
03.0400 General security and maintenance equipment for long-term storage	0		0	0	10000000	10000000	10000000		
04 DISMANTLING ACTIVITIES	0		0	0		0		293298169	
04.0100 Decontamination of areas and equipment in buildings to facilitate dismantling	0		0	0		0	0		
04.0200 Drainage of spent-fuel pool and decontamination of linings	8	5	40	342250		342250	342250		
04.0300 Preparation for dormancy	0		0	0		0	8500000		
04.0301 Zoning for long-term storage	0		0	0	4500000	4500000			+7 mill > 20 years
04.0302 Removal/disposition of inventory not suitable for long-term storage	0		0	0	4000000	4000000			
04.0400 Dismantling and transfer of contaminated equipment and material to containment structure for long-term storage	0		0	0		0	0		
04.0500 Sampling for radiological inventory characterisation in the installations after zoning and in view of dormancy	0		0	0		0	0		
04.0600 Site reconfiguration, isolating and securing structures	0		0	0		0	17000000		> 20 years
04.0601 Reconfiguration and maintenance of essential services and facilities to support long-term storage and/or decommissioning operations	0		0	0	5000000	5000000			
04.0602 Site boundary reconfiguration	0		0	0	2000000	2000000			
04.0603 Construction of temporary enclosures, storing, structural enhancements, etc. to support site remediation	0		0	0	10000000	10000000			
04.0604 Stabilisation of radioactive and	0		0	0		0			

hazardous waste pending remediation									
04.0700 Facility (controlled area) hardening, isolation or entombment	0		0	0		0	0		> 20 years
04.0800 Radiological inventory characterisation for decommissioning and decontamination	0		0	0		0	0		
04.0900 Preparation of temporary waste storage area	0		0	0	3000000	3000000	3000000		
04.1000 Removal of fuel-handling equipment	4	15	60	513375		513375	513375		
04.1100 Design, procurement, and testing of special tooling/equipment for remote dismantling	0		0	0		0	58615063		
04.1101 Design and procurement of special tools for dismantling the reactor vessel and internals	10	89	890	7615063	31000000	38615063			
04.1102 Design and procurement of special tools for dismantling other components or structures	0		0	0	20000000	20000000			
04.1200 Dismantling operations on reactor vessel and internals	7	39	273	2335856		2335856	2335856		
04.1300 Removal of primary and auxiliary systems	0		0	0		0	70028438		
04.1301 Removal of primary and auxiliary systems in reactor facilities	5	10	50	427813	2000000	2427813			
04.1302 Dismantling and removal of contaminated equipment, piping, liners and internal systems in non-reactor nuclear facilities	15	60	900	7700625	59900000	67600625			
04.1400 Removal of biological/thermal shield	0		0	0	80000000	80000000	80000000		
04.1500 Removal of other material/equipment from containment structure and all other facilities, or removal of entire contaminated facilities	10	108	1080	9240750		9240750	9240750		

04.1600 Removal and disposal of as-bestos	0		0	0		0	0		
04.1700 Removal of pool linings	0		0	0		0	0		
04.1800 Building decontamination	0		0	0		0	1796813		
04.1801 Removal of contamination from areas and structures in all buildings and stacks	2	5	10	85563		85563			
04.1802 Removal of embedded pipes in buildings	5	40	200	1711250		1711250			
04.1803 Removal of structures/facilities to gain access to radionuclides that may have breached design boundaries	0		0	0		0			
04.1900 Environmental cleanup	0		0	0		0	855625		
04.1901 Removal of embedded pipes out-side buildings	2	10	20	171125		171125			
04.1902 Removal of structures/facilities to gain access to radionuclides that may have breached design boundaries	0		0	0		0			
04.1903 Removal of contamination from areas and structures outside all buildings and stacks	10	8	80	684500		684500			
04.2000 Final radioactivity survey	0		0	0		0			
04.2100 Characterisation of radioactive materials	0		0	0		0			
04.2101 Characterisation of radioactive materials for recycling and reuse	0		0	0		0			
04.2102 Characterisation of radioactive materials for final disposal	0		0	0		0			
04.2200 Decontamination for recycling and reuse	2	400	800	6845000		6845000	6845000		over 2 x 5 years
04.2300 Personnel training	50	80	4000	34225000		34225000	34225000		over 2 x 5 years
04.2400 Asset recovery: Sale/transfer of metal or materials, and salvaged equipment or components for recycling	0		0	0		0	0		

or reuse									
05 WASTE PROCESSING, STORAGE AND DISPOSAL	0		0	0		0		3051338	
05.0100 Waste processing, storage and disposal safety analysis	1	5	5	42781		42781	42781		
05.0200 Waste-transport feasibility studies	1	1	1	8556		8556	8556		
05.0300 Special permits, packaging and transport requirements	0		0	0		0	0		
05.0400 Processing of system fluids (water, oils, etc.) from facility operations	0		0	0		0	0		
05.0401 Processing	0		0	0		0			
05.0402 Packaging	0		0	0		0			
05.0403 Transport	0		0	0		0			
05.0500 Processing of special system fluids (D2O, sodium, etc.) from facility operations	0		0	0		0	0		
05.0501 Processing	0		0	0		0			
05.0502 Packaging	0		0	0		0			
05.0503 Transport	0		0	0		0			
05.0600 Processing of waste from decontamination during facility operations	0		0	0		0	0		
05.0601 Processing	0		0	0		0			
05.0602 Packaging	0		0	0		0			
05.0603 Transport	0		0	0		0			
05.0700 Processing of combustible material from facility operations	0		0	0		0	0		
05.0701 Processing	0		0	0		0			
05.0702 Packaging	0		0	0		0			
05.0703 Transport	0		0	0		0			
05.0800 Processing of spent resins from facility operations	0		0	0		0	0		
05.0801 Processing	0		0	0		0			
05.0802 Packaging	0		0	0		0			

05.0803 Transport	0		0	0		0			
05.0900 Processing of other nuclear and hazardous materials from facility operations	0		0	0		0	0		
05.0901 Processing	0		0	0		0			
05.0902 Packaging	0		0	0		0			
05.0903 Transport	0		0	0		0			
05.1000 Storage of waste from facility operations	0		0	0		0	0		
05.1001 Preparation of storage facility	0		0	0		0			
05.1002 Waste storage	0		0	0		0			
05.1003 Storage of radioactive waste from facility operations	0		0	0		0			
05.1004 Decontamination of storage facili-ty	0		0	0		0			
05.1005 Dismantling/disposal of storage facility	0		0	0		0			
05.1100 Disposal of waste from facility operations	0		0	0		0	0		
05.1101 Preparation of disposal site	0		0	0		0			
05.1102 Waste disposal	0		0	0		0			
05.1103 Disposal of radioactive waste from facility operations	0		0	0		0			
05.1104 Disposal of non-radioactive waste from facility operations	0		0	0		0			
05.1200 Processing of decommission-ing waste	0		0	0		0	3000000		
05.1201 Processing of radioactive de-commissioning waste	0		0	0		0			
05.1202 Processing of non-radioactive de-commissioning waste	0		0	0		0			
05.1203 Waste containers	0		0	0	3000000	3000000			
05.1300 Packaging of decommissioning waste	0		0	0		0	0		
05.1301 Packaging of radioactive decom-	0		0	0		0			

missioning waste									
05.1302 Packaging of non-radioactive decommissioning waste	0		0	0		0			
05.1400 Transport of decommissioning waste	0		0	0		0	0		
05.1401 Transport of radioactive decommissioning waste	0		0	0		0			
05.1402 Transport of non-radioactive decommissioning waste	0		0	0		0			
05.1500 Storage of decommissioning waste	0		0	0		0	0		
05.1501 Preparation of storage facility	0		0	0		0			
05.1502 Waste storage	0		0	0		0			
05.1503 Storage of radioactive decommissioning waste.	0		0	0		0			
05.1504 Decontamination of storage facility	0		0	0		0			
05.1505 Dismantling/disposal of storage facility	0		0	0		0			
05.1600 Disposal of decommissioning waste	0		0	0		0	0		
05.1601 Preparation of disposal site	0		0	0		0			
05.1602 Decommissioning waste disposal	0		0	0		0			
05.1603 Disposal of radioactive decommissioning waste on disposal site	0		0	0		0			
05.1604 Disposal of non-radioactive decommissioning waste	0		0	0		0			
06 SITE SECURITY, SURVEILLANCE AND MAINTENANCE	0		0	0		0		0	
06.0100 Site security operation and surveillance	0		0	0		0	0		
06.0200 Inspection and maintenance of buildings and systems in operation	0		0	0		0	0		
06.0300 Site upkeep	0		0	0		0	0		
06.0400 Energy and water	0		0	0		0	0		

06.0500 Periodic radiation and environmental survey	0		0	0		0	0		
07 SITE RESTORATION, CLEANUP AND LANDSCAPING	0		0	0		0		26300000	
07.0100 Demolition or restoration of buildings	0		0	0		0	15300000		
07.0101 Dismantling of “balance-of-plant” systems and building components	0		0	0		0			N/A
07.0102 Dismantling of the structure	0		0	0	15000000	15000000			
07.0103 Dismantling of the stack	0		0	0	300000	300000			
07.0200 Final cleanup and landscaping	0		0	0	10000000	10000000	10000000		
07.0300 Independent compliance verification with cleanup and/or site-reuse standards	0		0	0	1000000	1000000	1000000		
07.0400 Perpetuity funding/surveillance for limited or restricted release of property	0		0	0		0	0		
08 PROJECT MANAGEMENT, ENGINEERING AND SITE SUPPORT	0		0	0		0		0	
08.0100 Mobilisation and preparatory work	0		0	0		0	0		
08.0101 Mobilisation of construction equipment and facilities	0		0	0		0			
08.0102 Mobilisation of personnel	0		0	0		0			
08.0103 Set-up/construction of temporary facilities	0		0	0		0			
08.0104 Construction of temporary utilities	0		0	0		0			
08.0105 Temporary relocations	0		0	0		0			
08.0200 Project management and engineering services	0		0	0		0	0		
08.0201 Project manager and his staff	0		0	0		0			
08.0202 Planning and cost control	0		0	0		0			
08.0203 Quality assurance and quality surveillance	0		0	0		0			
08.0204 Procurement, warehousing, and	0		0	0		0			

materials handling									
08.0205 General/subcontractor administration	0		0	0		0			
08.0206 Documentation and records control	0		0	0		0			
08.0207 Engineering support	0		0	0		0			
08.0300 Public relations	0		0	0		0	0		
08.0400 Support services	0		0	0		0	0		
08.0401 Housing, office equipment, site services	0		0	0		0			
08.0402 Computer support	0		0	0		0			
08.0403 Decommissioning support including chemistry, decontamination and field supervision	0		0	0		0			
08.0404 Waste-management support	0		0	0		0			
08.0500 Health and safety	0		0	0		0	0		
08.0501 Health physics	0		0	0		0			
08.0502 Radiation protection and monitoring	0		0	0		0			
08.0503 Industrial safety	0		0	0		0			
08.0600 Demobilisation	0		0	0		0	0		
08.0601 Removal of temporary facilities	0		0	0		0			
08.0602 Removal of temporary utilities	0		0	0		0			
08.0603 Demobilisation of construction equipment and facilities	0		0	0		0			
08.0604 Demobilisation of personnel	0		0	0		0			
09 RESEARCH AND DEVELOPMENT	0		0	0		0		0	
09.0100 Research and development of decontamination, radiation measurement and dismantling processes, tools and equipment	0		0	0		0	0		
09.0200 Simulation of complicated work on model	0		0	0		0	0		
10 FUEL AND NUCLEAR MATERIAL	0		0	0		0		68410700	
10.0100 Transfer of fuel or nuclear ma-	3	16	48	410700	68000000	68410700	68410700		for 5 years

terial from facility or from temporary storage to intermediate storage									
10.0200 Intermediate storage	0		0	0		0	0		
10.0201 Wet intermediate storage	0		0	0		0			
10.0202 Dry intermediate storage and containers	0		0	0		0			
10.0300 Dismantling/disposal of temporary storage facility	0		0	0		0	0		
10.0301 Decontamination of temporary storage facility	0		0	0		0			
10.0302 Dismantling/disposal of temporary storage facility	0		0	0		0			
10.0400 Preparation of transfer of fuel or nuclear material from intermediate storage to final disposition	0		0	0		0	0		
10.0500 Dismantling/disposal of intermediate storage facility	0		0	0		0	0		
10.0501 Decontamination of intermediate storage facility	0		0	0		0			
10.0502 Dismantling/disposal of intermediate storage facility	0		0	0		0			
11 OTHER COSTS	0		0	0		0		0	
11.0100 Owner costs	0		0	0		0	0		
11.0101 Implementation of transition plan	0		0	0		0			
11.0102 Capital expenditures	0		0	0		0			
11.0200 General, overall (not specific) consulting costs	0		0	0		0	0		
11.0300 General, overall (not specific) regulatory fees, inspections, certifications, reviews, etc.	0		0	0		0	0		
11.0400 Taxes	0		0	0		0	0		
11.0500 Insurances	0		0	0		0	0		
11.0600 Overheads and general administration	0		0	0		0	0		
11.0700 Contingency	0		0	0		0	0		

11.0701 Risk, financial assurance versus inherent uncertainties	0		0	0		0			
11.0702 Escalation of high-risk cost elements	0		0	0		0			
11.0800 Interest on borrowed money	0		0	0		0	0		
11.0900 Asset recovery: Resale/transfer of general equipment and material	0		0	0		0	0		
Investments at Waste Plant					25000000	25000000	25000000	25000000	
Total			**8.747**	**74.841.519**	**362.700.000**	**437.541.519**	**437.541.519**	**437.541.519**	

Cost estimate, produced by DEMEX Consulting Engineers A/S, for demolition of DR 3 buildings (available in Danish only)

Indledning

Med henvisning til bestilling nr. 2000012794 af 28.11.2000 gives følgende prisoverslag for nedrivning af 7 bygninger og prydbassin, RISØ DR3.

Prisoverslaget er baseret på udvendig og indvendig visuel besigtigelse, samt udleveret tegningsmateriale

Tal for affaldsmængder er udregnet på baggrund af tegningsmaterialet, suppleret med DEMEX erfaringstal for enhedsvægte.

Priserne for bortskaffelse af affald er beregnet på baggrund af prisoplysninger fra modtageanlæg, som er godkendt af Roskilde Kommune. Priserne for etablering af byggeplads, miljøsanering og nedrivning er beregnet ud fra DEMEX erfaringstal.

Bygningsmassen som er beskrevet i nedenstående består af servicebygninger, værksteder kontorer, lagre og nødkontrolrum placeret omkring DR-3. Endvidere er beskrevet et gammelt køletårnsfundament, samt et bygværk til udledning og indtag af kølevand fra fjorden.

Bygningernes etageareal er opgjort til:

Bygning 214/214a	2.500 m^2
Bygning 215	590 m^2
Bygning 216	100 m^2
Bygning 217	1.050 m^2
Bygning 218	65 m^2
Bygning 226	315 m^2
Bygning 247	225 m^2
Prydbassin	380 m^2
Samlet etageareal	5.225 m^2

Hvor intet andet er beskrevet, er byggeriet projekteret og opført i slutningen af 50'erne. Størstedelen af bygningerne er opført i beton og tegl, med få undtagelser, hvis bærende konstruktion er af stål.

Prisgrundlag

Etablering og drift byggeplads

Etablering og drift af byggeplads i forbindelse med bygge- og anlægsarbejder inkluderer etablering og drift af anlægsområde, herunder etablering af skurby med forsyninger af vand og el, transport af maskiner til pladsen, opsætning af hegn, skilte mm.

Der budgetteres med gennemsnitligt 25.000 kr. til etablering af byggeplads etc., og 5.000 kr. pr. måned i driftsomkostninger. Det vurderes, at der bør regnes med en samlet daglig mandskabsstyrke på gennemsnitlig 3-5 personer.

Det er ved udregningerne forudsat, at der afsættes én måned pr. bygning til nedrivningen. Ved nedrivning af alle bygninger under ét vil perioden kunne reduceres til ca. tre måneder. Såfremt entreprenøren presses i tid vil dette kunne forventes at medføre meromkostninger.
Omkostninger til etablering og drift kan muligvis reduceres, såfremt der findes mulighed for at bentte faciliteter i eksisterende bygninger.

Miljøsanering

Miljøsanering omfatter asbestsanering og udsortering af andre problematiske stoffer, herunder PCB, CFC, kviksølv og andre tungmetaller. Der er ikke foretaget detaljeret kortlægning af miljøproblematiske stoffer. Mængden af miljøproblematiske stoffer vurderes generelt at være begrænset. Enkelte af bygningerne indeholder dog større mængder af eternitplader med formodet indhold af asbest

Udgiften til miljøsanering af de enkelte bygninger er fastsat som den forventede udgift til nedtagning af eternitplader og fjernelse af sædvanlige forekomster af andre skadelige stoffer.

Nedrivning

Nedrivning omfatter alle omkostninger til nedrivning, inkl. afbrydelse af forsynings- og afløbsledninger, men ekskl. bortskaffelse af affald. Sædvanligvis regnes med en gennemsnitspris på ca. 260 kr. pr. m^2 baseret på aktuelle erfaringer fra tilsvarende danske nedrivningsprojekter. For betonkontruktioner (fx prydbassinet) regnes med ca. 350 kr. pr. m^3 beton.

Bortskaffelse af nedrivningsaffald

Omkostninger til bortskaffelse af nedrivningsaffald afhænger af typen af fraktioner.

Beton og tegl:
Såfremt der foretages fornøden kildesortering af nedrivningsmasserne, vil tegl og beton med fordel kunne bortskaffes til genanvendelse. I mange tilfælde vil entreprenøren kunne finde afsætning for materialerne direkte til et igangværende anlægsprojekt. Udgiften til bortskaffelse vil derfor være lav og i nogle tilfælde positiv. I dette budget benyttes priser for bortskaffelse til modtageanlæg.

Prisen for bortskaffelse af beton afhænger af, om der er tale om rene eller blandende fraktioner samt om betonen er med eller uden armering. Endelig vil prisen afhænge af det konkrete modtageanlæg. Mindstrupgård som anvises af Roskilde Kommune regner med 42 kr. pr. ton for modtagelse af ren tegl og beton.

Metal:
Metallet fra nedrivningen vurderes hovedsagelig at bestå af konstruktionsstål samt armeringsjern. Stålet forventes at kunne indbringe ca. 15-30 øre pr. kg. ved salg til genbrug.

Brændbart:
Brændbare materialer bortskaffes til forbrænding. Pris for bortskaffelse til forbrænding hos I/S KARA er ca. 658 kr. pr. ton (inkl. statsafgift).

Deponi:
Øvrigt byggeaffald bortskaffes til deponi. Prisen for bortskaffelse af affald til deponi (fx til Hedeland losseplads) er ca. 600-900 kr. pr. ton (inkl. statsafgift) og er afhængig af sammensætningen.
Specialaffald:
Prisen for modtagelse af specialaffald (fx hårde eternitplader med asbest til Mindstrupgård) ligger på ca. 600 kr. pr. ton (inkl. statsafgift).

Transport

Omkostninger til transport af affaldet vurderes at udgøre ca. 2,50 kr. pr. ton pr. km. Den gennemsnitlige transportafstand for bortskaffelse af affald vurderes at være ca. 10 km.

Budget for nedrivning af bygninger

Bygning 214/214a
Bygningen har 3 formål: Kontor, hovedadgang til DR3 samt værksteds- og afkølingsområde.

Bygningen er opført i et plan med bærende konstruktion i stål.

Etagearealet udgør 2500 m^2.

Etablering			30.000
Miljøsanering			25.000
Nedrivning	2500 m^2	260 kr./m^2	650.000

Bortskaffelse		m^3	t/m^3	T	kr./t	kr.	
	Tegl	311	1,6	498	42	20.899	
	Beton	906	2,4	2.174	42	91.325	
	Metal	21	7,8	161	- 450	- 72.450	
	Brændbart	10	0,5	5	795	3.975	
	Specialaffald	9	2,0	18	1.050	18.900	
	Øvrigt deponi	20	2,3	46	795	36.570	
	Total			2.902		99.219	99.219

Transport	2.902 t	2,5 kr./km/t	10 km	72.550
Total				**876.769**

Bygning 215
Bygning 215 har været anvendt som driftsbygning med generatorer til nødforsyning af kølesystemet. Bygningen er opbygget i beton, med facade i tegl.

Bygningen har enkelte pumpefundamenter og fuld kælder

Etagearealet udgør 590 m^2.

Etablering			30.000
Miljøsanering			5.000
Nedrivning	590 m^2	260 kr./m^2	153.400

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Tegl	66	1,6	106	42	4.435	
	Beton	448	2,4	1.075	42	45.158	
	Metal	0,7	7,8	5	- 450	- 2.250	
	Brændbart	5	0,5	3	795	2.385	
	Øvrigt deponi	15	2,3	35	795	27.825	
	Total			1.224		77.553	77.553

Transport	1.224 t	2,5 kr./km/t	10 km	30.600
Total				**296.553**

Bygning 216
Bygværket er opført i beton og har været anvendt til indtag og udledning af kølevand ved fjorden.

Etagearealet udgør 100 m^2.

Etablering			30.000
Miljøsanering			0
Nedrivning	75 m^3	350 kr./m^3	26.250

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Beton	75	2,4	180	42	7.560	7.560

Transport	180 t	2,5 kr./km/t	10 km	4.500
Total				**68.310**

Bygning 217
Bygningen har været anvendt som pumpehus for køle og udluftningssystem.

Bygningen svarer til bygning 215 hvad angår materialer og udseende. Endvidere har bygning 217 en tilbygning opført i stål.

Etagearealet udgør 1050 m^2.

Etablering			30.000
Miljøsanering			5.000
Nedrivning	1050 m^2	260 kr./m^2	273.000

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Tegl	161	1,6	258	42	10.836	
	Beton	701	2,4	1.682	42	70.644	
	Metal	1,8	7,8	14	- 450	- 6.300	
	Brændbart	16	0,5	8	795	6.360	
	Deponi	18	2,3	41	795	32.595	
	Total			2.003		114.135	114.135

Transport	2.003 t	2,5 kr./km/t	10 km	50.075
Total				**472.210**

Bygning 218
Bygningen har været anvendt som nødkontrolrum. I kraft af bygningens funktion har den særligt tykke dæk og vægge.

Bygningen er opført som massiv beton med yderside i tegl. Bygningen forudsættes funderet i 2 meter.

Etagearealet udgør 65 m^2.

Etablering			30.000
Miljøsanering			5.000
Nedrivning	65 m^2	260 kr./m^2	16.900

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Tegl	5	1,6	8	42	336	
	Beton	110	2,4	264	42	11.088	
	Deponi	5	2,3	12	795	9.540	
	Total			284		20.964	20.964

Transport	284 t	2,5 kr./km/t	10 km	7.100
Total				**79.964**

Bygning 226
Bygningen består af en lagerbygning samt en mindre tilbygning, som indeholder back-up batterier til køleanlægget.

Bygningen er opbygget i stål med udfyldende tegl. Tilbygningen er opført med facader i eternit.

Etagearealet udgør 315 m^2.

Etablering			30.000
Miljøsanering			10.000
Nedrivning	315 m^2	260 kr./m^2	81.900

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Tegl	68	1,6	109	42	4.578	
	Beton	171	2,4	410	42	17.220	
	Metal	3	7,8	23	- 450	10.350	
	Specialaffald	0,6	2,0	1	1.050	1.050	
	Øvrigt deponi	10	2,3	23	795	18.285	
	Total			566		51.483	51.483

Transport	566 t	2,5 kr./km/t	10 km	14.150
Total				**187.533**

Bygning 247
Bygning 247 er en relativt ny bygning, der anvendes til kontorformål.

Bygningen er opført i stål og tegl med tagkonstruktion i træ og stålplader.

Etagearealet udgør 225 m^2.

Etablering			30.000
Miljøsanering			0
Nedrivning	225 m^2	260 kr./m^2	58.500

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Tegl	25	1,6	40	42	1.680	
	Beton	176	2,4	422	42	17.724	
	Metal	1	7,8	8	- 450	- 3.600	
	Brændbart	10	0,5	5	795	3.975	
	Øvrigt deponi	10	2,3	23	795	18.285	
	Total			498		38.064	38.064

Transport	498 t	2,5 kr./km/t	10 km	12.450
Total				**139.014**

<u>Prydbassin</u>
Det nuværende prydbassin er oprindelig opført som fundament for tidligere køletårne.

Bassinet består udelukkende af beton.

Etagearealet udgør 380 m^2.

Etablering			30.000
Miljøsanering			0
Nedrivning	170 m^3	350 kr./m^3	59.500

Bortskaffelse		m^3	t/m^3	t	kr./t	kr.	
	Beton	170	2,4	408	42	17.136	17.136

Transport	408 t	2,5 kr./km/t	10 km	10.200
Total				116.836

Vurdering af samlet budget for nedrivning:

Bygning 214/214a	876.769 kr.
Bygning 215	296.553 kr.
Bygning 216	68.310 kr.
Bygning 217	472.210 kr.
Bygning 218	79.964 kr.
Bygning 226	187.533 kr.
Bygning 247	139.014 kr.
Prydbassin	116.836 kr.
Total	2.237.189 kr.

Til de anførte budgetpriser for nedrivning bemærkes, at der i forbindelse med udbud af nedrivningsopgaver i praksis ses i meget svingende priser, og variationer på op til 50% mellem de bydende er ikke usædvanligt. Disse variationer kan oftest forklares ved de bydende entreprenørers aktuelle ordresituation, eller deres forventning til yderligere arbejde i forlængelse af entreprisen. Med forbehold for disse udsving kan den beregnede pris tages til udtryk for prisen i et 'normalt' marked.

Prisen for etablering af byggeplads er i beregningerne fastsat udfra, hvad der må forventes, såfremt bygningerne nedrives enkeltvis. Der kan således forventes en besparelse såfremt bygningerne nedrives samlet.

Ligeledes kan der forventes en betydelig mængderabat på selve nedrivningen såfremt bygningerne nedrives samlet.

Endelig anbefales, at regne ca. 10% til uforudsete udgifter til diverse arbejder.

Med baggrund ovenstående bemærkninger vurderes den samlede pris for nedrivning at ligge i intervallet fra 1.700.000 - 2.500.000 kr.

Appendix 4

Cost estimate for Hot Cells

Risø National Laboratory

PRICE ESTIMATE

Project Management - No Costs Allocated
Implementation Cost KR 24.669k
Waste Costs - No Costs Allocated

ESTIMATE REPORT

22-12-2000

Overall Cost KR 24.669k

Local Labour Rate Applied KR 230
"R" Task Factor 25,0

Site: Hot Cell
Definitive List : 350010000 **Estimate Version 000**
Facility Name : Hot Cell

Stag	Area	WPG	WP	Description	Comp.	Complexity	Task	No	Item Quantity	Total Quantity	Cost (KR)
20											
		0		**Hot Cells**							
		00	000	Establish venting and electricity	000	0	A	1			50,0k
		B1	031	Build new external facilities	000	0	A	1			10.000,0k
		D1	010	Remove hot sources	010	3	R	5	0,01 kg	0,05 kg	90,6k
		D1	050	Remove ventilation	050	1	C	1	200 kg	200 kg	97,3k
		D1	071	Remove workbenches	071	1	R	6	300 kg	1.800 kg	55,9k
		D1	071	Remove crane	071	3	C	1	10.000 kg	10.000 kg	232,9k
		D1	072	Remove ventilation	072	1	C	1	3.500 kg	3.500 kg	29,0k
		D1	072	Remove shutters	072	1	C	6	2.000 kg	12.000 kg	99,4k
		D1	072	Remove Cable drum house	072	1	C	1	3.000 kg	3.000 kg	24,8k
		D1	072	Remove carousell	071	1	C	1	1.000 kg	1.000 kg	12,4k
		D1	072	Decontamination of carousell	072	2	C	1	1.000 kg	1.000 kg	13,2k
		D1	081	Decontamination of top of cells	081	2	C	1	300 m2	300 m2	263,9k
		D1	081	Decontamination of inside of cells	081	3	R	1	1.000 m2	1.000 m2	4.140,0k

Risø National Laboratory

Site: Hot Cell

Definitive List : 350010000 Estimate Version 000

Facility Name : Hot Cell

Stag	Area	WPG	WP	Description	Comp.	Complexity	Task	No	Item Quantity	Total Quantity	Cost (KR)
		D2	070	Remove cells	082	5	R	1	300 m2	300 m2	6.210,0k
		D2	070	Disposal of concrete	320	1	M	1	1.000 m3	1.000 m3	110,0k
		D4	031	Reestablish building	170	5	M	1	1.000 m2	1.000 m2	3.240,0k
										Area Total =	**24.669,3k**
										STAGE TOTAL =	**24.669,3k**

2 of 2 22-12-2000 **Risø National Laboratory**

Appendix 5

Health physics aspects related to the costing items of the NEA/IAEA/EC list

02.0400 Determination of the inventory of radioactive materials inside the nuclear installations and in their immediate surroundings

02.0401 Radioactive inventory inside the nuclear installations at Risø

The activity content in the structural materials in the reactors DR 1, (DR 2) and DR 3 will be determined in two ways: (1) from model calculations of activation processes and (2) from γ-spectrometric analyses of destructive and non-destructive samples. The model calculations are possible only if the material composition is known in detail. Activation analyses, *e.g.* in DR 1, of small material samples are therefore necessary to determine the material composition in the construction parts.

The activity content in the concrete cells (especially in cells 1, 2 and 3) in the Hot Cell plant will be determined from (1) measurements of radiation fields with TL-dosemeters and subsequent modelling and (2) α- and γ-spectrometric analyses of smear samples and other samples taken from the cells. Ventilation facilities and breathing protection facilities, *e.g.* frogman suit equipment, should be re-installed in the Hot Cell plant before entering the cells. In addition, changing rooms, bath facilities and instrumentation for measurements of contamination on clothes and in air should be established.

Health physics expertise and manpower is needed to determine the activity inventory in the nuclear installations. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

02.0402 Sub-grade soil sampling and monitoring wells to map contamination plumes

During the operation of the nuclear installations at Risø, radioactive materials were routinely released in the atmospheric and aquatic environments. Incidents during operation might also have released activity to the environments, and underground installations like tanks might have leaked due to corrosion. Therefore, the activity levels in soils and sediments need to be documented through an extensive sampling and analysis programme before the single sites can be released with green field status.

These measurements will be performed when all other decommissioning tasks are near completion. Manpower needs are difficult to estimate but they will include expertise on measurement of low-level nuclide-specific activity content in soil and other environmental samples. The Nuclear Safety Research Department at Risø has presently this expertise, which should be kept at least at the present level, so the foreseen post-decommissioning environmental tasks can be implemented.

Health physics expertise and manpower is needed to determine the extent of any contamination of the environment. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

02.0700 Facilities for decontamination of systems for dose reduction

Decontamination of facilities and systems in the nuclear facilities, *e.g.* the concrete cells in the Hot Cell plant, will often be a justified and necessary measure to reduce the radiation fields before decommissioning tasks can be implemented. Decontamination is here defined in a broad sense and includes separation of active and non-active parts of bulky components like the top shield of reactor DR 3. Remote handling facilities like water basins containing cutting tools are needed for such large-scale decontamination operations. Detailed dummy operations in a real scale will be crucial. The manpower resources and investments for decontamination operations will be among the largest of the decommissioning costs.

Health physics expertise and manpower is needed to determine the extent of decontamination operations. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

02.0800 Removal of waste from decontamination operations

The radioactive waste produced by decontamination operations - both for dose reduction purposes and in waste segregation operations during the decommissioning operations - has to be removed for temporary storage at the Risø area. Much of the waste has a high radionuclide concentration with a resulting high radiation level.

Health physics surveillance of the waste removal operations is therefore needed. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

03.0300 General radiation protection and health physics equipment

A new laboratory for γ-spectrometric analyses of large components should be built in an area, where the background radiation from activity in the nuclear facilities is close to zero. This facility is needed for the documentation and release of bulky materials and large components as non-radioactive waste. The costs are approximately:

- Laboratory building: 5-6 Mkr.
- Gamma-spectrometric equipment: 2-3 Mkr.
- Handling equipment (crane, conveyor system etc.): 1-2 Mkr.
- **Total costs: 8-11 Mkr.**

In addition to the personal TL-dosemeters there will be an increased need for instantaneous dose readings with digital dosemeters. A new and more reliable digital dosemeter system, *e.g.* Siemens Electronic Personal Dosemeter, should replace the existing rather old ALNOR digital dosemeter system. Estimated total costs for a new digital dosemeter system from Siemens are:

- Dosemeters, readers, control terminals, TeleTrak transmitters and calibration unit: **1-1.5 Mkr.**

An important feature of the Siemens digital dosemeter system is the possibility to survey the dose meter accumulations remotely using radiotransmitters.

04.0100 Decontamination of areas and equipment in buildings to facilitate dismantling and release

Buildings and equipment that should be released for general use, excluding work with radioactive materials, should be decontaminated to a level below the release level for buildings and equipment. After decontamination, detailed radiation surveys, γ-spectrometric analyses of smear samples and destructive samples as well as model calculations of doses to the users of the building are necessary to document and demonstrate that the buildings and equipment in fact can be released. In any case, the radiation protection authority should approve on a case by case basis the release of buildings and equipment.

If the buildings and equipment are to be dismantled and the building waste are to be released as non-radioactive waste, decontamination before dismantling to below release levels for building rubble and metal scrap is needed. After decontamination, detailed radiation surveys, γ-spectrometric analyses of representative samples as well as model calculations of doses to the population for selected exposure scenarios are necessary to document and demonstrate that the building waste in fact can be released after the approval from the authority.

Health physics expertise and manpower is needed to document that buildings/equipment or the building waste can comply with the release levels and that the resulting doses after a release are acceptable. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

04.0500 Sampling for radiological inventory characterisation in the installations after zoning and in view of dormancy

The activity inventory in buildings and equipment that should be left sealed in dormancy needs to be detailed documented so future decommissioning personnel are able to prepare plans for the dismantling and decommissioning of the buildings and equipment without the implementation of a new programme for inventory characterisation. The characterisation of the inventory will be based upon radiation surveys, γ-spectrometric analyses of non-destructive and destructive samples as well as model calculations.

Health physics expertise and manpower is needed to document the inventory of radionuclides in installations, on equipment, and on surfaces. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

04.0800 Radiological inventory characterisation for decommissioning and decontamination

In order to determine radiation exposures and cleanup of the nuclear facility sites, suitable measurement strategies and criteria are required to identify the extent of site contamination and to determine the required clean-up. Clean-up criteria would generally be expressed in individual dose in excess of background to the average member of the critical group. Criteria should also be derived for decontamination of buildings and subsequent release of the buildings for other purposes or for decommissioning.

Because of the difficulties in directly measuring dose to humans, especially as a result of internal exposure, and because the dose criteria may not be readily or directly measurable because they are often very low, these criteria must generally be converted into more readily measurable quantities, operational quantities, such as mass activity concentration (Bq/kg or Bq/l), dose rate (μSv/h) and surface contamination density (Bq/m^2) in the contaminated media. The operational quantities are calculated as follows:

$$\text{operational quantity} = \frac{\sum_{\text{all pathways}} \text{annual dose after clean-up}}{\sum_{\text{all pathways}} \text{annual dose per unit operational quantity}}$$

Such calculations require a detailed understanding of the nature and extent of the contaminated area, environmental factors for the area like transfer factors to food and location factors, the reasonably possible pathways by which humans may be exposed to radiation from this area, and the scenarios that describe how the site will be used after clean-up.

Health physics expertise and manpower is needed to derive criteria and operational quantities for release of sites and buildings/building waste. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

04.2000 Final radioactivity survey

After completion of the cleanup/decontamination of the nuclear facility sites/buildings/equipment a comprehensive radiation/contamination survey should be implemented. The purpose is to document that the decontaminated sites/buildings can comply with the release criteria. A carefully planned survey should be based on a sampling strategy that allows a proper averaging of the measured data. Implementation of the measurements requires proper instrument selection, calibration, measurement techniques, and the recording of data. The sensitivity of the instruments directly affects the number of measurements needed.

Health physics expertise and manpower is needed to make the necessary surveys for compliance with the criteria. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

04.2100 Characterisation of radioactive materials

The waste produced during the decommissioning of the nuclear facilities and during decontamination of sites and buildings should be characterised in terms of radionuclide specific activity concentrations. A detailed measurement programme is therefor needed. Such a programme would include γ-spectrometric analyses of representative samples of the waste and gross γ-measurements/γ-spectrometric measurements on the waste units (drums, containers etc.) to document the activity content in the waste units, which should be stored in a temporary shielded waste storage facility at Risø. The final disposal of the radioactive waste units would hereafter be a relatively simple operation. Of equal importance is the separation/segregation of radioactive and non-radioactive waste, which would result in a significantly reduced volume of radioactive waste.

Health physics expertise and manpower is needed to make the necessary characterisation of the radioactive/non-radioactive waste produced during the decommissioning of the nuclear facilities. The health physics expertise and manpower needed for all health physics tasks in the decommissioning project are discussed in Section 6.9.1.

Appendix 6

Summary of the review seminar

RISØ

21-02-2001, Kurt Lauridsen

Summary of discussions at the review seminar 14 February 2001

Participants:

Reviewers:

Roger Andres (RA), Paul Scherrer Institut, Wolfgang Pfeifer (WP), Forschungszentrum Karlsruhe, John Williams (JW), UKAEA

Danish Decommissioning

Knud Larsen (KnL), managing director

Risø National Laboratory:

Jørgen Kjems (JK), managing director, Lisbeth Grønberg (LG), Knud Brodersen (KB), Mogens Bagger Hansen (MBH), Benny Majborn (BM), Lisbeth Warming (LW), Nils Hegaard (NH), Klaus Iversen (KI), Kurt Lauridsen KL)

The discussions at the seminar were structured in order to have the reviewers' comments to the following items:

- the approach to the work
- the scenarios - possibly recommendations as to which one to choose
- the technical decommissioning tasks identified
- the estimated costs
- how to proceed in refining the cost assessment

plus, of course, other items the reviewers wished to comment on.

The approach to the work

The reviewers agreed that the report is well written and structured, easy to navigate in.

JW noted that his main issue of concern is that more emphasis should be put on the waste issues, including the approval and licensing aspects for the treatment and disposal, which to his experience may be very time and resource consuming. The other reviewers agreed to this point of view.

RA drew the attention to the need for acceptance criteria for waste containers, and WP pointed to the necessity of identifying a repository for radioactive waste, as the specifications of the waste containers depended on the repository design.

Furthermore, RA recommended that the Danish regulatory system be described in the report, also for the benefit of readers in neighbouring countries.

The scenarios - possibly recommendations as to which one to choose

There was general agreement between the three reviewers that a short decommissioning scenario should be preferred, even though radiation levels will be higher than after a long cooling period (but for the timescales for the three scenarios the differences will not allow significantly different dismantling procedures to be used). Among the arguments for a short scenario the following were underlined:

- the availability of skilled personnel (especially relevant in Denmark, since there are no other places in the country working with nuclear technology)
- the waste management facility needs to have a reasonably continuous flow of work without prolonged periods with little or no influx of waste

In addition, WP mentioned that the larger costs for the long scenarios may be even larger than indicated in the report if discounting of the costs were applied, and as JW added, the Danish authorities need to maintain relevant knowledge as the liability still exists.

RA mentioned that some cooling time would be achieved even in the short scenario, since it would take at the least 10 years to dismantle and check the external equipment at DR 3.

The reviewers expressed general concern about the possible loss of expertise in long scenarios, and shared the common view, that utilisation of foreign expertise sometime in the future would be much more expensive, than using the existing expertise now. The reviewers recommended the short scenario. JW suggested considering a common approach for the DIDO/PLUTO users in Germany, Australia, the UK and Denmark.

The technical decommissioning tasks identified

<u>DR 1:</u>
JW noted that the only major problem in his opinion would be the disposal of the fuel solution. RA and WP agreed. WP suggested that the fuel might be treated as radioactive waste.

<u>DR 2:</u>
RA noted that disposal of beryllium, lead and aluminium might pose problems, which should be considered. In this context WP mentioned that experience exist at the FZK with conditioning aluminium waste in concrete.

For DR 2 as well as for the other two reactors it was noted that the disposal of graphite could be difficult. RA mentioned that a Swiss patent exists on the disposal/waste treatment of graphite.

<u>DR 3:</u>
WP mentioned that release of tritium during the decommissioning work might pose a problem.

JK asked whether heavy water could be considered a tradable item. JW informed that the UKAEA had not received any income from the disposal of the heavy water from DIDO, PLUTO and SGHWR. The high tritium content of the Risø Heavy Water may pose a problem, preventing the re-use in power plants without treatment. The receiver of the UKAEA’s Heavy Water provided new drums for it to be transported and paid for the transport. The UKAEA packaged the Heavy Water, made the safety cases and disposed of the wastes arising from the transfer and storage process within the UK. Overall the disposal was a cost the UKAEA with no income for the ‘sale’ of the Heavy Water. The Risø Heavy water might have a lower tritium content than the UKAEA’s so it might have some value. The UKAEA collected all its redundant Heavy Water together so only one disposal was made.

WP found the use of the IAEA/OECD/EC list of costing items very useful. But at the FZK the cost estimates usually are made directly in term of money rather than in terms of man-hours. The reason is that it is not FZK personnel but contractors that perform the demolition work; FZK personnel plans the work and is responsible for the safe performance of it.

JW advised to be cautious when dismantling electrical installations; the UKAEA has had some bad experiences with work on electrical cables that were thought to be dead but turned out not to be so.

Replying to a question from KI, both WP and JW confirmed that radioactive material had been found in unexpected locations during decommissioning at FZK and UKAEA.

JK asked the reviewers to inform about their strategy and procedures of radiation protection, for instance with a view to external contractors.

For the UKAEA, JW informed, the people from outside contractors have to be suitably qualified (and trained) with respect to working on any UKAEA licensed site but the UKAEA still has the responsibility for the safety and could not pass on this responsibility to contractors. However, some of the radiation protection service is provided by (different) contractors, but the UKAEA ensured that they satisfied all the safety requirements. JW noted that very much monitoring is necessary.

RA informed that people entering a nuclear facility in Switzerland must be properly instructed in both radiation protection and quality control. He was satisfied to see that an adequate health physics workforce had been foreseen in the report, and advised that the health physicists be involved in the planning of decommissioning operations as well as in the surveillance of the work being performed. He found that Danish dose limits and the ALARA and optimisation principles ought to be mentioned in the report.

WP informed that the organisation in charge of the health physics at FZK is strictly separated from the dismantling staff. There are health physics managers present at all facilities where decommissioning work is going on. Most of the radiation protection work is carried out by external contractors.

KnL put the question to the reviewers: "If you had personnel in-house, would there still be arguments for outsourcing the work?" WP found that it depends on the situation of the staff - also seen from a social point of view. He advised not to be afraid of employing contractors. RA, on the other hand, strongly advised to use own personnel as long as it is available, as they are much cheaper.

RA observed that the report does not mention a quality management system. KL explained that this was due to the fact that the new organisation, Danish Decommissioning, which is being established at present, is in the process of setting up a quality management system. This subject has, therefore, been kept out of the report. RA informed that the PSI uses the ISO system for quality management. JW told that the UKAEA has set up their own, very large, system, which is computer based and available to all staff on all its sites. The FZK also has its own system; it does not put so much emphasis on documentation, since the final goal of the work is the removal of the facilities.

<u>The Isotope laboratory and the Fuel fabrication facility</u>
The reviewers had no comments to the report concerning these small facilities.

<u>Hot cells:</u>
RA found the estimated radiation doses for the Hot cell dismantling very high.

JW commented that using water jets for decontamination of surfaces will not be an adequate method, because it may lead to contamination being moved into cracks and crevices; dry cleaning methods should be used.

JW and WP informed that both the FZK and the UKAEA have much experience with decontaminating and decommissioning hot cells. JW specifically mentioned one case of a hot cell that had been contaminated with hydraulic oil in addition to a large number of cobalt pellets being spilled. This cell suite had been cleaned completely and demolished.

RA pointed to the difficulty in measuring plutonium contamination; this demands a very low background radiation from other radionuclides.

JW drew the attention to the fact that using robots raises the cost with more than just the price of the robot itself. Mock-up facilities and training of the operators adds substantially to the cost. Furthermore, WP added, robots need to be adapted to the individual facility. RA was not very much in favour of using robots at all, but pointed to the ALARA optimisation of cost versus dose (in Switzerland a value of 3000 CHF per person-mSv is used).

WP concluded that there would probably not be any need for actual robots at Risø; normal, fixed manipulator systems will suffice. JW will provide information on the decommissioning of two UKAEA hot cell facilities which involved the use of special handling systems, one being a NEATER Robot which was adapted from a Unimation Commercial Robot and the other an ARTISAN heavy duty hydraulic manipulator.

Waste storage and handling:
WP commented that the decommissioning tasks seemed to be relatively easy. Concerning the possible investments listed, e.g. sandblasting equipment, he drew the attention to the existence of mobile equipment that can be rented. RA added that also mobile facilities for clearance measurements might be rented.

JW commented that, based on UK experience, recovery of waste from underground storage facilities might prove difficult if there is corrosion of the waste containers.

WP mentioned that old documentation has been evaluated due to changed demand on waste containers. (4 million DM / year for re-inspection and re-evaluation)

JK asked for advice as to what to do with the 233 kg of hot cell waste, containing fuel. RA mentioned that a waste management subgroup under the European Atomic Energy Society was planning a meeting in Mol, trying to set up a common strategy for such questions. KB informed that Anne Sørensen from the Waste Management Plant is going to participate in the meeting.

The estimated costs

WP asked whether the cost estimates include waste treatment; at the FZK waste treatment and packaging gives about 20% of the total cost. KB confirmed that waste treatment costs had been taken into account.

Both WP and RA found that the cost estimate for DR 3 was too high; RA mentioned that the cost of decommissioning of the DIORIT reactor will be around 100 million DKK, and WP mentioned that the decommissioning for the MERLIN reactor in Jülich was around 200 million DKK.

Concerning the management costs it was recommended to specify these in more detail, because they comprise activities, which are not normally considered as management, e.g. health physics services.

Furthermore, it was recommended to add more transparency to the costs given in chapter 10, Total costs.

WP presented a comparison of the cost estimate for decommissioning of Risø's waste management plant with that for the corresponding facility at the FZK; a good agreement was found between the costs per ton (Risø: 2600 DKK/t, FZK: 5000 DKK/t), considering the differences in complexity between the two facilities.

As a general remark to the cost estimates JW noted that all estimates in the decommissioning field have large uncertainties, in particular in an early stage. For instance, the UKAEA had experienced that responses to calls for tenders had shown very large ranges of bids from different contractors.

RA mentioned that a major risk to the estimating of the time and cost schedule was delay in response from authorities and politicians.

How to proceed in refining the cost assessment

WP recommended for further iterations that costs be divided into cost blocks, such as preparation, dismantling, waste treatment, health physics etc., and that a work breakdown structure be made for each decommissioning project.

In general it was recommended that the costs for DR 3 should be scrutinised in more detail in order to reduce uncertainty in the estimates.

Other items the reviewers wish to comment on

Referring to chapter 6.9.3, "Research and development of decontamination, radiation measurement and dismantling processes, tools and equipment", RA stated that there in his opinion would not be a need for actual research in order to perform the decommissioning; maybe some development. But R&D-like activities and the maintenance of international contacts and collaboration might be necessary in order to keep the staff motivated.

Regarding the estimated personnel doses WP commented that the estimate for DR 3 seemed very high. In his experience, the doses accumulated during decommissioning in general are lower than those during operation, because work can better be planned and there is less time pressure in the performance of the tasks. Adding to this, JW told that, the UKAEA experience is that doses can be reduced by ensuring only the staff undertaking the work are in the vicinity of the work place and that video systems are used to observe much of the work from a distance. Video recording and training had been a good tool to improve this situation with supervisors able to show the operational staff what they had to do without always having to enter the high radiation areas.

PAUL SCHERRER INSTITUT

Review of a report:

Decommissioning of the nuclear facilities at Risø National Laboratory: Descriptions and cost assessment (K. Lauridsen, ed.); Draft

Roger Andres
Paul Scherrer Institut
Switzerland

February 2001

Contents

1. Method of review

Before the onset of the review, a few decisions regarding the procedure of review have been taken. The four main points are:

- It was felt to be advisable that the report should give all the information relevant to the general decisions needed now. It will probably be read by a variety of people not necessarily fully informed in the respective areas of knowledge. It should therefore contain all the relevant information. Consequently, no information in addition to that in the report has been sought regarding the specific situation at the Risø National Laboratory and in Denmark in general, in order to make the review of the document as impartial as possible.
- The contents of the report are compared to the situation in Switzerland, not because this should represent a particularly valuable benchmarking, but it is a comparison to a working system, well known to myself.
- Questions I have asked were, amongst other: Is the report consistent, comprehensive, understandable and correct?
- The issues regarding nuclear fuel have not been reviewed, since my corresponding experience is very limited, and therefore of insufficient relevance.
- It seems a very difficult, if not impossible task, to review such a report without at the same time having my own opinion on certain procedures proposed in the document. No effort was undertaken to filter these opinions out of the statements concerning exclusively the report itself.

2. General comments

2.1. Structure and contents of the report

The report represents a very comprehensive source of information about the situation at the nuclear facilities in Risø. It addresses all the relevant points of the task of decommissioning and dismantling these installations. The most important issues treated in the report are

- Technical procedures and management
- Health physics
- Regulatory issues
- Finance
- Etc.

Reference is made, whenever possible and appropriate, to international guidelines (European Commission, IAEA, etc.) and experience is cited from other places, that have gone through similar dismantling procedures. This seems eminently sensible, as it would definitely not be economical to “re-invent” everything.

The report is clearly structured, and leads the reader through the information along a well-defined route. Chapters 1 to 5 set out the basic information, the methods used in the analysis of the task and sources of information. The rest of the report then describes the results of the deliberations and offers options for decisions to be made.

A strict numerical system of chapters, subchapters etc. makes navigation and cross-referencing in the report easy. It clearly is the product of a well lead and motivated team.

2.2. Level of detail

The level of detail given in the different chapters varies widely:

- Very detailed information is found regarding technical issues and finance.
- Issues of health protection and management are treated competently, but in less detail.
- The regulatory situation, questions of interim and final storage (not part of this report, but impinging on it) are dealt with in a rather general fashion only.

The decommissioning and dismantling of several complete nuclear installations, as planned at Risø National Laboratory, is a very complex task, for which detailed information regarding all aspects will be needed at a certain point in time. The pattern of the levels of detail as described above is not unusual, it simply represents the frame of mind of scientists and engineers. However, it might be possible that other organizations, e.g. policy makers, have other points of view, and their respective needs might later have to be addressed in greater detail. At the current stage of debate, the information found in the report is certainly sufficient to start a fruitful discussion, leading to a first set of decisions.

3. Specific comments

3.1. Technical points

In chapter 6 (and the respective annexes) a description is given of the tasks to be undertaken to dismantle each of the facilities. The DR 3 reactor is correctly identified as the big task in all respects. However, it seems not very productive for me to comment on details, without having had the opportunity to see the situation. As a general rule it seems wise to keep up all safety related ancillary equipment, because the subsequent dismantling operation result in an increased risk relative to normal operations. The authors have correctly identified this need as an unavoidable obligation for the future.

The very detailed description of the dismantling operation of DR 3, given in Annex 3, is prove of a careful analysis of the situation. If the dismantling work begins it will be necessary to continuously adapt and fine-tune the work plan, as it will be inevitable that unexpected hurdles or opportunities arise. This is true also for all the other facilities under consideration.

It seems unnecessary to further elaborate the technical aspects. Some of the tasks are not very demanding and some might lay in the rather distant future anyway.

The technical aspects of the report are certainly commensurate with the needs of the present stage of the decision process.

3.2. Health physics

Issues relating to health physics permeate the whole report, which is prove of the importance of this subject to the authors. However, because radiation protection has both political and ethical dimensions, the report might profit from a somewhat firmer commitment to definite limits for the doses incurred by the workers and the public, and from a clear demonstration of the will to keep these low. International guidelines are cited, that sometimes have the character of very general advice only, but constitute good "checklists" to check the completeness of health physics concepts.

Some specific points worth mentioning are:

- At several places throughout the report ICRP-reports are cited. Modern health physics systems make extensive use of this advice. A key feature is the concept of optimization. In the report this concept is use solely in the framework of exemption and release of materials, but not in the context of operations. In particular, the goal of ALARA ("as low as reasonably achievable") is not mentioned explicitly.
- For the dismantling of the DR 3 a total dose of 2000 person mSv is estimated. This seems very high, in particular when compared to other decommissioning experience such as PLUTO and DIDO. Some optimization of the procedures might be indicated.
- It is gratifying to see, that adequate manpower is planned for the completion of health physics tasks. From experience in Switzerland it is evident that the operational tasks of the health physics personnel are not very demanding, if the dismantling crew is adequately instructed in operational radiation protection rules. However, a very big task is the measurements for clearance, because it is clearly more difficult to prove the absence than the presence of contamination. In addition, a data bank has to be kept for future demonstration of compliance with regulations.
- At several places a concept is mentioned to distribute dose to a numerous workforce. Although a common practice, this procedure seems ethically justified only if all reasonable efforts have been taken to reduce the dose through careful optimization.
- At various places throughout the report the problem of releases is mentioned, including limits not to be exceeded (e.g. inhalation dose to public less than 10 μSv). As the issue of off-site doses to persons has delicate national (and possibly international) dimensions, it is recommended that this subject be treated carefully and in a somewhat firmer fashion before major dismantling activities are started.

3.3. Finance

It is apparent that the authors of this report have very carefully addressed all financial aspects. The use of the PRICE program, explicitly shown for DR 3, is an eminently suitable way to estimate the costs of dismantling. As has been stated by the authors, care should be taken not to over-interpret these numbers. At this stage of discussion they cannot represent financial information with a degree of confidence usually found in a business environment. Nevertheless, it is recommended to improve the transparency and clearly specify what will (and what will not) be delivered with the funds requested.

Not unexpectedly, dismantling of DR 3 again is the major single contributor of costs. The financing of the central parts of this endeavor results in a peak of monetary needs. This, in my opinion, has consequences to be taken into consideration on a political and management level, and this issue will be addressed below.

The funds requested are high by any standards. Rather than wishfully reducing them with unrealistic optimism, it would be more useful at this stage to confirm the willingness - and show possibilities - for continuous cost control and possibly, reduction, during the lifetime of the whole operation.

Apart from all the above qualifications, the numbers are adequate for the decisions needed now, and the best one can deliver at this time.

3.4. Intermediate storage and final repository

It is explicitly stated that the subject of a final repository is not within the scope of this report, but this issue nevertheless has consequences for the actions described in the document.

It is inferred from the report that no fixed criteria exist for the acceptance of waste containers with regard to their size, weight, structure, stability and radiological parameters. At some point in the future such a definition of the acceptability of containers is needed, in order give guidance to the crew involved in the actual dismantling. Repacking of containers due to shifting acceptance parameters is not a viable option, both from a financial and a health physics point of view.

No special consideration is given to the disposal of chemically problematic and radioactive materials like lead, aluminum and beryllium. It is recommended that this issue should be addressed.

3.5. Regulatory oversight

In accordance with international guidelines, regulatory oversight is mentioned in the report. However, the document makes no clear reference to a strong regulatory authority, charged with the task of independent scrutiny of the decommissioning and dismantling work. But it can be questioned, whether the establishment of a full regulatory body is advisable, in particular in light of the fact that Denmark has no nuclear power plants. Still, this question has to be answered at an early point in time, as the potential strengthening of the existing organization takes a considerable time, mostly due to the delicate national political issues involved that may also have international dimensions.

3.6. Quality management

No reference is made throughout the report to a formalized quality management system, for instance based on EN ISO 9001 and EN ISO/IEC 17025. Such a system is audited and approved at regular intervals by independent experts. Having such a formal system in operation is very useful when questions by concerned stakeholders have to be faced regarding credibility, traceability and trust.

It is recommended to study the usefulness of having a formalized quality management system in place when major dismantling operations are commenced.

4. Management issues

When looking at the report (and the dismantling projects) as a whole, several important points come to mind. The authors have addressed most of these. They still will be raised here, as the statements below may represent useful aspects from a slightly different point of view. It is well known from past business and public endeavors, that the quality of management is one of the most decisive factors determining the ratio of “quality and quantity of work” to “resources used”. Some possibly helpful thoughts are outlined below.

4.1. Flexibility

Risø National Laboratory possibly is at the beginning of a completely new phase with a major reorientation towards decommissioning and dismantling activities. A few of the parameters important for this work are not firmly set. They are amongst other:

- No fundamentally different dismantling techniques (at least for DR 3) will be possible, even after prolonged cooling times.
- No major differences are expected in costs for rapid or delayed dismantling (mainly DR 3).
- No fixed sequence of dismantling for most of the individual facilities is demanded.
- No separate regulatory authority exists.
- No fixed framework exists for the acceptance of waste containers with regard to their size, weight, structure, stability and radiological parameters.

The room for maneuvering therefore is considerable. This represents both an opportunity and a risk. The latter lays in the possibility of prolonged controversial discussions to fix the parameters, delaying the whole process.

4.2. Opportunities

The freedom of action as described above permits the optimization of the process with respect to other parameters, most notably resources needed and (radiation) safety. The management can therefore fully take advantage of interleaving various activities in order to share on-site experience and equipment. It is recommended that this is a continuous management process, once the decision to dismantle is made. In this way it might also be possible to smoothen out the peaks of financial needs, which (at least in Switzerland) are difficult to communicate to political funding authorities.

In any case, there presently is room for pragmatism that could be advantageously used in many respects.

4.3. Timeliness

Despite the great operational freedom for the management there probably is a rather well defined sequence of decisions to be taken and actions to be initiated. At the onset of the major dismantling activities an outline of this sequence should probably be elaborated, possibly with the help of modern electronic planning tools. In this way a smooth overall process and an optimized use of resources is likely to be achieved.

In the absence of other decisive factors (see 4.1.) it seems advisable to start early with dismantling. The main reason is the limited time during which experienced operators of the respective facilities are still available as part of the workforce. It also is ethically questionable whether major dismantling liabilities should be left to future generations.

4.3. Research

At several places in the report the need for research is mentioned. In my opinion no research is needed to handle the problems that will arise during the dismantling activities. The information found in the scientific literature seems wholly adequate for the needs of dismantling operations. In addition, most institutions that have successfully completed such projects are willing to share their experience. However, there certainly is need for development of site-specific methods and procedures.

There might be another reason to do real research: For the motivation and the giving of professional perspectives to senior staff. In a field, where the long-term outlook is so clearly limited, this might be the only way to maintain a knowledge base adequate to the task. If this is the reason for mentioning research in this report, it is recommended to clearly state is so in the interest of credibility.

5. Conclusions

It is my strongly held opinion that the report as a whole is

- Consistent with regard to contents and form
- Comprehensive, as it addresses all essential aspects
- Understandable due to a successful and strict editorial process
- Correct, according to my knowledge and experience

It is the correct format and size to serve as guidance for the upcoming basic decisions, and it represents an excellent base for future, more detailed considerations, decisions and reports.

I congratulate the team for an impressive piece of work, and I am thankful for the opportunity to have had a chance to review this report.

Roger Andres February 2001

First review of report on decommissioning of the nuclear facilities at Risø National Laboratory

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3. Description of the nuclear facilities

- documentation of plant-installations
- activity inventories
- remaining reactivate residues from operation

All three points are described in the necessary details to get a good view about the nuclear facilities.

4. Relevant experience from other countries

In Germany there are similar reactors as

- University of Hannover
- DKZ Heidelberg (TRIGA-Reaktor)
- PTB – Braunschweig (1 MW)
- Merlin FRJ1 (10 MW) Jülich
- DDO FRJ2 (23 MW) Jülich

Some reports can be delivered or connections can be made to the above mentioned institutions.

The Karlsruhe reactors with an average of 100 MW_{th} are much larger compared to the RISØ reactors.

5. Approach to cost assessment

- scenario

  I would prefer the short-term scenario.

  Reasons are:

  - availability of skilled own personal
  - waste management facilities would have continuously work
  - to push consideration to establish acceptance criteria's for a repository
  - after the decision for decommissioning DR3 giving the public the feeling to abandon nuclear-activities as far as possible
  - the DR3 can be dismantled within the next 10 year

- Taking into account the escalated costs; the cost comparison between the 3 scenarios would be more different

- the management costs with 450 MiDKK are very high compared to the total costs. From this point of view one request should be o start decommissioning as soon as possible will all projects.
- a work breakdown structure should be compatible with time schedule and working process and costs.

6. Decommissioning work

- DR1
  - Is it possible to declare the fuel solution as radioactive waste and how can it be treated?
  - What a about the activated graphite blocks, which processing is intended?

- DR2
  - It is not clear if dose rate of some reactor parts demand remote handled equipment for cutting and filling in shielded containers
  - What a about beryllium as a toxic material, how it is treated and does it have an effect on the basic criteria's for the foreseen repository.

- DR3
  - The total costs for DR3 with 560 million DKK seems quite high compared to our own experiences; does it include costs for waste treatment?
  - Some estimated costs including waste-management from the MZFR which have been found realistic are:

- activated steel	600 KDKK/Mg
- contaminated materials	200 KDKK/Mg
- activated concrete	60 KDKK/Mg
- demolition of inactive building structure	4 KDKK/Mg

Fig. 1 shows the structure of project costs of the MZFR without repository and fuel-element costs.

- Waste management plant
  - the RISØ waste management plant can be compared in some way with FZK-plant for treatment of radioactive liquid effluents (HDBplant).

    Fig. 2 shows the waste categories and Fig. 3 the total costs.

    In Fig. 4 a cost comparison is made.

- Waste management plant

  There is no breakdown to waste categories and how the waste will be treated. For example: There must be some burnable waste like clothes etc.

Other questions are:

- What can be transported to treatment facilities outside Denmark; for example to Studsvik, Sweden

- For which kind of radioactive residues an interim storage is foreseen. In some cases treatment and conditioning will be done when acceptance criteria for a repository are established.

- 6.9.2 Investments

  A lot of mentioned equipment exists in other countries (frogman area)
  - think about to order mobile facilities like sand blasting, tree realizing measurement
  - shielded and unshielded containers exist and must not be developed
  - Measurement can be made in laboratories in France, UK etc.

- 6.9.3 Research and development

  At FZK we never saw the necessary for research in the decommissioning field. Most equipment, which is needed can be planned and manufactured by engineering companies. In some special cases it is recommended to proof equipment in a mock up facility.
  Research costs time and time is money and the results will always come too late.

7. Environmental aspects

Considering EU regulations; as a part of a decommissioning license the environmental aspects must be proved within the regulations; a public hearing is needed.

8. Licensing aspects

- for clearance of radioactive materials we can provide you with experiences we made
- For exemption limits the IAEO-BSS requirements are not practicable. The EC-regulations are much better to handle
- the report does not give any ideas about the licensing procedure. This part is very important for time and cost estimation.

9. Project management

If you are interested we can provide you with information about the used tools.

10. Total costs

- From the point of costs do decommissioning as quick as you can especially under the aspects you will not be longer involved in the nuclear field.

- 10.2 Radiation doses

  The estimated total personnel dose for DR3 is quite high compared to the MZFR reactor which will be in a range of 1000 mSv in total (see report; Radiation Exposure of the personal during dismantling of the primary system of the Karlsruhe MZFR).

11. Requirements to a repository

- take the demolition waste to refill grounds on site

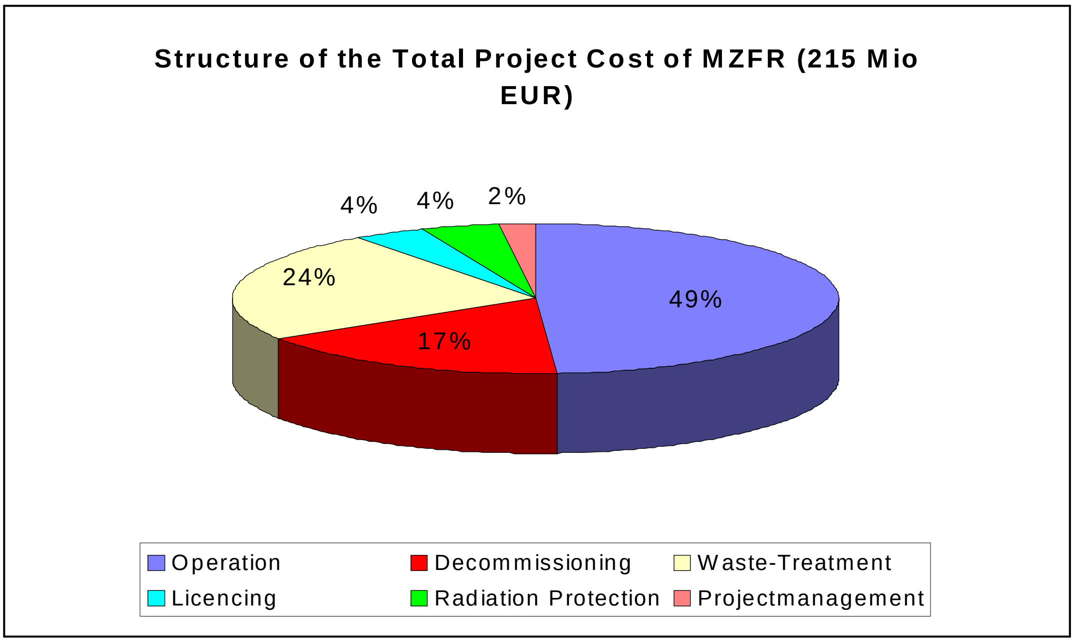

Fig 2 Initial waste masses and categories of the LAW Evaporation and Cementing

Waste categories	mass of conditioned waste (Mg)	recycled material after decontamination and free release measurement (Mg)	total (Mg)
Process components	26	59	
Contaminated systems	34	78	
Contaminated equipment	61	133	
Construction masses	88	10271	
Steel components	10	37	
total	**217**	**10578**	**10795**

Area of the building 16510 m³
Volumes of the building 9800 m³

Fig 3 Survey of Costs and Funds Required

No.	Activity	Costs (TDM)
1.	**Planning (contracts)**	**1640**
	Planning	820
	Expert opinions and licenses	820
2.	**Dismantling of processing systems and demolition of buildings (contracts)**	**8560**
	Operation media and supply	690
	Preparation	820
	Radiation protection	1100
	Dismantling of processing systems	1800
	Dismantling of measurement and control systems	420
	Decontamination, packaging and removal	1660
	Demolition of buildings and cleaning of sites	2070
3.	**Own staff**	**3620**
	Engineering and dismantling expenses in total	**13820**
4.	**Waste treatment and conditioning (HDB)**	**29100**
	Equipment decontamination	17500
	LAW scrapping	9500
	Incineration	1500
	Release measurement laboratory	600
5.	**Interim storage**	**300**
6.	**repository**	**3300**
Total costs		**46520**

Fig. 4 Comparison of cost-estimation of RISØ waste management plant to FZK LAW-plant (without costs of waste treatment and final disposal of)

	total mass (Mg)	area m³	Volume m³	total cost KDKK	Cost/Mg KDKK
RISØ	1000	1800		2594	2,6
FZK	10000	16500	9800	50000	5

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Edited by Kurt Lauridsen

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Abstract (max. 2000 characters)

The report is the result of a project initiated by Risø National Laboratory in June 2000 on request from the Minister of Research and Information Technology. It describes the nuclear facilities at Risø National Laboratory to be decommissioned and gives an assessment of the work to be done and the costs incurred. Three decommissioning scenarios were considered with decay times of 10, 25 and 40 years for the DR 3 reactor. The assessments conclude, however, that there will not be much to gain by allowing for the longer decay periods; some operations still will need to be performed remotely. Furthermore, the report describes some of the legal and licensing framework for the decommissioning and gives an assessment of the amounts of radioactive waste to be transferred to a Danish repository.

Descriptors INIS/EDB

ACTIVITY LEVELS; COST ESTIMATION; DECOMMISSIONING; DR-1 REACTOR; DR-2 REACTOR; DR-3 REACTOR; HOT CELLS; LEGAL ASPECTS; LICENSING; NUCLEAR FACILITIES; RADIOACTIVE WASTE FACILITIES; RADIOACTIVITY; RISOE NATIONAL LABORATORY

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