Nordisk kernesikkerhedsforskning
Norrænar kjarnöryggisrannsóknir
Pohjoismainen ydinturvallisuustutkimus
Nordisk kjernesikkerhetsforskning
Nordisk kärnsäkerhetsforskning
Nordic nuclear safety research

NKS-165
ISBN 978-87-7893-230-3

Cost Calculations for Decommissioning and Dismantling of Nuclear Research Facilities

Inga Andersson[1], Steinar Backe[2], Anna Cato[3],
Henrik Efraimsson[4],Klaus Iversen[5], Staffan Lindskog[3],
Seppo Salmenhaara[6] and Rolf Sjöblom[7]

[1] Studsvik Nuclear AB, Sweden
[2] Institute for Energy Technology, Norway
[3] Swedish Nuclear Power Inspectorate, Sweden
[4] Swedish Radiation Protection Authority, Sweden
[5] Danish Decommissioning, Denmark
[6] VTT Technical Research Centre of Finland, Finland
[7] Tekedo AB, Sweden

July 2008

Abstract

Today, it is recommended that planning of decommission should form an integral part of the activities over the life cycle of a nuclear facility (planning, building and operation), but it was only in the nineteen seventies that the waste issue really surface. Actually, the IAEA guidelines on decommissioning have been issued as recently as over the last ten years, and international advice on finance of decommissioning is even younger. No general international guideline on cost calculations exists at present.

This implies that cost calculations cannot be performed with any accuracy or credibility without a relatively detailed consideration of the radiological prerequisites. Consequently, any cost estimates based mainly on the particulars of the building structures and installations are likely to be gross underestimations.

The present study has come about on initiative by the Swedish Nuclear Power Inspectorate (SKI) and is based on a common need in Denmark, Finland, Norway and Sweden.

The content of the report may be briefly summarised as follows. The background covers design and operation prerequisites as well as an overview of the various nuclear research facilities in the four participating countries: Denmark, Finland, Norway and Sweden.

The purpose of the work has been to identify, compile and exchange information on facilities and on methodologies for cost calculation with the aim of achieving an 80 % level of confidence.

Key words

decommissioning, cost calculations, nuclear, research facilities, fund, nordic

NKS-165
ISBN 978-87-7893-230-3

Electronic report, July 2008

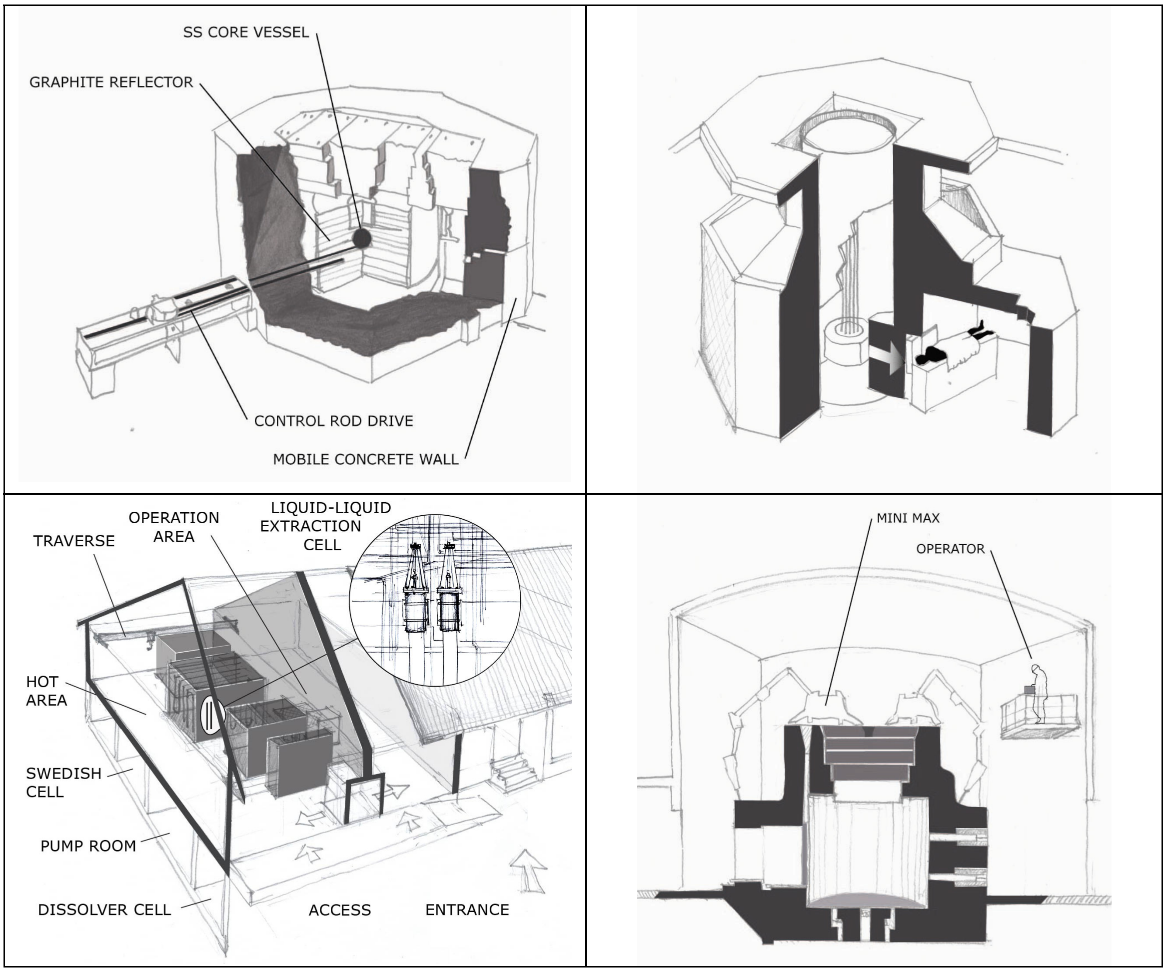
The report can be obtained from
NKS Secretariat
NKS-776
P.O. Box 49
DK - 4000 Roskilde, Denmark

Phone +45 4677 4045
Fax +45 4677 4046
www.nks.org
e-mail nks@nks.org

COST CALCULATIONS FOR DECOMMISSIONING AND DISMANTLING OF NUCLEAR RESEARCH FACILITIES

Volume I, Main Report

Inga Andersson, Studsvik Nuclear AB
Steinar Backe, Institute for Energy Technology
Anna Cato, Swedish Nuclear Power Inspectorate
Henrik Efraimsson, Swedish Radiation Protection Authority
Klaus Iversen, Danish Decommissioning
Staffan Lindskog, Swedish Nuclear Power Inspectorate
Seppo Salmenhaara, VTT Technical Research Centre of Finland
Rolf Sjöblom, Tekedo AB



May 2008

Perspektiv

Bakgrund

Detta uppdrag har finansierats gemensamt av Dansk Dekommissionering, Institutt for energiteknikk (IFE, Norge), Nordisk Kärnsäkerhetsforskning, Statens Kärnkraftinspektion (SKI, Sverige), Tekniska Forskningscentralen (VTT, Finland). Projektet är initierat av SKI som också bidrar med perspektivet nedan.

SKI presenterar den 1 september varje år ett förslag till regeringen om avgifter för det kommande året inom ramen för den s.k. Studsvikslagen . En viktig del i detta arbete är att avgöra om det finns en jämvikt mellan vad som är fonderat i kärnavfallsfonden och de framtida åtagandena för dekontaminering och nedläggning av vissa kärnteknisk verksamhet som bedrivits vid Studsvik.

I arbetat med att analysera och värdera fondens utveckling är de framtida kostnaderna den väsentligaste variabeln. För de flesta objekt rör det sig om belopp på 10-tals miljoner kronor eller mer och dessa belopp kräver att detaljerade kostnadsberäkningar skapas, analyseras och evalueras. I föreliggande projekt görs ett försökt till att utveckla mera ändamålsenliga metoder för att verifiera att en korrekt skattning ligger till grund för beräkning av de totala framtida kostnaderna, och den därpå följande fonderingen, av äldre kärntekniska anläggningar.

Syfte

Detta forskningsprojekt har haft till syfte att utveckla en metod för en värdeneutral och tydlig beräkning av kostnaderna för dekontaminering och nedläggning av äldre kärntekniska anläggningar kan göras i ett tidigt skede. Uttrycket tidigt skede refererar till att beräkningar skall göras idag för kostnader som infaller i en avlägsen framtid. Det kan till och med vara så att kalkylen omfattar en tidsrymd på upp emot ett halvt sekel.

Då flera av de nordiska länderna har, eller har haft, forskningsreaktorer som endera har rivits eller kommer att rivas så finns det fördelar till ett aktivt kunskapsutbyte från ett samnordiskt perspektiv. Att utveckla en modell för beräkning av de framtida kostnaderna i syfte att skapa tillförlitligare och robustare uppskattningar av kostnaderna i ett tidigt skede, i vissa fall innan avvecklings- och rivningsprocessen har inletts, är en angelägen uppgift.

Resultat av studien

I denna rapport ges explorativa beskrivningar av vunna erfarenheter från tidigare nordiska projekt. Genom att beskriva hur dekontaminering och avveckling av äldre kärntekniska anläggningar tidigare har gjorts kan ett underlag skapas för fortsatt analys och diskussion kring hur kostnadsberäkningar på bästa sätt kan utvecklas..

Effekter av SKI finansierad forskningsverksamhet

Genom att utveckla metoder för att skapa en god praxis för kalkylering av kostnader i ett tidigt skede i planeringsprocessen för avveckling och rivning av kärntekniska anläggningar är det möjligt att tillse att nutida generationers användning av nukleärt alstrad elenergi verkligen bär sina kostnader. Detta leder i sin tur till att framtida generationer inte behöver ta något konsumtionsutrymme i anspråk för dessa frågor, utan kan istället ägna sig att lösa de specifika frågor som de framtida generationerna kommer att möta.

SKI kommer att använda resultatet från denna studie i den årliga granskning som göras av den kostnadsberäkning som AB SVAFO lämnar in i enlighet med ”Studsvikslagen”. Denna kostnadsberäkning ingår som en central del i det förslag till avgifter som SKI:s styrelse lämnar till regeringen. Denna forskningsrapport kommer att ingå i det granskningsmaterial som SKI analyserar i samband med framställningen av ett förslaget till avgifter för år 2008..

Behov av fortsatt forskning

De empiriska beskrivningarna som presenteras i rapporten kan ligga till grund för en konstruktion av en modell för beräkning av framtida kostnader i de nordiska länderna. Genom att sedan validera de beräkningsresultat som modellen genererat kan en utvärdering göras av modellens reliabilitet och validitet. En sådan jämförande analytisk utvärdering kan endast göras om flera länder deltar i forskningsprocessen. I ett andra steg bör en gemensam modell tas fram.

Projektinformation

På SKI har Staffan Lindskog varit ansvarig för att samordna projektet. Forskningsarbetet har koordinerats av Rolf Sjöblom på TEKEDO AB.

SKI referens: 2005/584/200509079

Content

Volume I

Volume II

APPENDICES

Summary

Today, it is recommended that planning of decommission should form an integral part of the activities over the life cycle of a nuclear facility (planning, building and operation).

It was only in the nineteen seventies that the waste issue really surfaced, and together with it to some extent also decommissioning. Actually, the IAEA guidelines on decommissioning [1-5] have been issued as recently as over the last ten years, and international advice on finance of decommissioning is even younger [5-6]. No actual international guideline on cost calculations exists at present.

Intuitively, it might be tempting to regard costs for decommissioning of a nuclear facility as similar to those of any other plant. However, the presence of radionuclide contamination may imply that the cost is one or more orders of magnitude higher as compared to a corresponding inactive situation, the actual ratio being highly dependent on the level of contamination and later use of the facility in question.

This implies that cost calculations cannot be performed with any accuracy or credibility without a relatively detailed consideration of the radiological prerequisites. Consequently, any cost estimates based mainly on the particulars of the building structures and installations are likely to be gross underestimations.

The present study has come about on initiative by the Swedish Nuclear Power Inspectorate (SKI) and is based on a common need in Denmark, Finland, Norway and Sweden.

It was found in various studies carried out on commission by SKI (see e g [7-11] where [10] is included in the present report in the form of Appendix F) that the intended functioning of a system for finance requires a high precision even in the early stages of cost calculations, and that this can be achieved only if the planning for decommissioning is relatively ambitious. The following conclusions were made:

- IAEA and OECD/NEA documents provide invaluable advice for pertinent approaches.
- Adequate radiological surveying is needed before precise cost calculations can be made.
- The same can be said about technical planning including selection of techniques to be used.
- It is proposed that separate analyses be made regarding the probabilities for conceivable features and events which could lead to significantly higher costs than expected.
- It is expected that the need for precise cost estimates will dictate the pace of the radiological surveying and technical planning, at least in the early stages.

- It is important that the validity structure for early cost estimates with regard to type of facility be fully appreciated. E g, the precision is usually less for research facilities.
- The summation method is treacherous and leads to systematical underestimations in early stages unless compensation is made for the fact that not all items are included.
- Comparison between different facilities can be made when there is access to information from plants at different stages of planning and when accommodation can be made with regard to differences in features.
- A simple approach is presented for "calibration" of a cost estimate against one or more completed projects.
- Information exchange and co-operations between different plant owners is highly desirable.

The present report represents a realisation of the above thoughts in a Nordic context. At present, the content of the report may be briefly summarised as follows.

A relatively ambitious background is provided since it is essential that the design and operation prerequisites and particulars are reasonably well understood when – at a much later stage – decommissioning is to be carried out. The background also comprises an overview of the various nuclear research facilities in the four participating countries: Denmark, Finland, Norway and Sweden.

The purpose of the work has been to identify, compile and exchange information on facilities and on methodologies for cost calculation with the aim of achieving an 80 % level of confidence.

The scope has been as follows:

- to establish a Nordic network
- to compile dedicated guidance documents on radiological surveying, technical planning and financial risk identification and assessment
- to compile and describe techniques for precise cost calculations at early stages
- to compile plant and other relevant data

A separate section is devoted in the report to good practice for the specific purpose of early but precise cost calculations for research facilities.

A separate section is also devoted to techniques for assessment of cost.

Examples are provided for each of the countries of relevant projects. They are as follows:

- Research reactor DR1 in Denmark
- The TRIGA research reactor in Finland
- The uranium reprocessing plant in Norway
- Research reactor R1 in Sweden

1 Background

1.1 Introduction

Today, it is recommended [1-6,12-13] that planning of decommission should form an integral part of the activities over the life cycle of a nuclear facility (planning, building and operation). It is further recommended that funding of decommission should be a part of the overall planning and funding of the facility.

This recommendation did not exist in the nineteen forties when man-made radionuclides were generated in significant quantities for the first time in conjunction with utilization of chain reactions and associated neutron activation in nuclear reactors and nuclear explosives. It was only in the nineteen seventies that the waste issue really surfaced, and together with it to some extent also decommissioning. Actually, the IAEA guidelines on decommissioning [1-4,13] have been issued as recently as over the last ten years, and international advice on finance of decommissioning is even younger [5-6]. No actual international guideline on cost calculations exists at present, but there exists an ASTM standard and an IFRS accounting standard.

This situation contrasts to that of radiation protection, where the need for it was actually realized from the very beginning of nuclear technology.[14-16] The x-rays had been discovered half a century earlier and had become utilized on a grand scale virtually overnight. Application of x-rays in medicine improved diagnoses and thereby also treatment immensely, but lack of appropriate protection also led to many cases of health detriment. Consequently, a lot of experience and knowledge was available in the nineteen forties as well as methodology for radiation protection.[14-16]

Thus, focus was kept on radiation protection during operation of the facilities, and little or no precautionary measures were taken to facilitate the waste management and decommissioning. Eventually, and in the course of events, it was realized that the undertakings and costs for waste management and decommissioning would be substantial.

Intuitively, it might be tempting to regard costs for decommissioning of a nuclear facility as similar to those of any other plant. However, the presence of radionuclide contamination may imply that the cost is one or more orders of magnitude higher as compared to a corresponding inactive situation, the actual ratio being highly dependent on the level of contamination and later use of the facility in question.

This implies that cost calculations cannot be performed with any accuracy or credibility without a relatively detailed consideration of the radiological prerequisites. Consequently, any cost estimates based mainly on the particulars of the building structures and installations are likely to be gross underestimations.

There are a number of reasons why cost estimates for decommissioning are considerably more difficult to make for old nuclear research facilities as compared to modern nuclear power plants:

- Plans for decommissioning do not exist
- They were not designed with regard to decommissioning
- They are small (which means that investigations can become expensive in relation to the total cost)
- They are very different in character
- The types of contamination are different, e g with regard to radionuclides and activity levels (which relates to detectability / penetration of the radiation), spatial distribution, surface or bulk, wet/dry, soluble/non-soluble e t c
- Different methodologies for decontamination and dismantling are appropriate depending on the circumstances
- The buildings were constructed and operated at a time when the regulations were considerably less strict than today
- Incomplete documentation of the operation history, accidents and incidents causing contamination
- Institutional memory has been lost and people who know what took place may no longer be alive
- The efficient and economical application of methodologies developed for large scale applications at nuclear power plants

Accordingly, general figures on the international nuclear legacy are difficult to find and do not exist with any precision. It was presented recently[17] that the environmental management cleanup cost for Department of Energy in the US amounted to 6.2 G$ for the fiscal year 2004. It was said in the presentation that it might be expected that this effort will be continued for a few decades.

It seems plausible that the international nuclear legacy associated with nuclear research, development and defence may exceed 1 T$. This figure is comparable to that of the gross national product of the Nordic countries combined (0.91 T$ in the year 2003).

However, there exists valuable information from a large number of decommissioning projects that have been completed. Many of those have been successful in technical as well as financial terms. A general feature of those projects is that they have included appropriate planning and consideration of the specifics of the facility in question. This experience forms the basis for the present day recommendations mentioned above on planning for decommissioning throughout the various phases of the life cycle of a facility.

Several countries have requirements on collection of funds during the operation of a facility. In such cases the overall planning might be prompted and promoted by the financial requirements.

1.2 General international development

The early developments of nuclear technology in the Nordic countries were strongly influenced by the preceding international events.

Nuclear fission was discovered just before the start of the Second World War. It was soon realized the effect might be utilized for very powerful explosives. This led to the initiation of the Manhattan project in the United states and the subsequent bombing of Hiroshima (a bomb based on U-235) and Nagasaki (a bomb based on Pu-239).

The Manhattan project involved enormous resources and had a very tight time schedule. When the decision was taken on the project it was not known what, if any, route might lead to a functioning bomb. Therefore, alternative methods were being developed in parallel.

The abundance of U-235 in natural uranium is around 0.7 %. This would have to be increased to above around 80 % to be feasible in a bomb (actually much higher enrichment of uranium-235 was used).[1]

The plutonium-239 was obtained from reprocessing of natural uranium fuel used in a graphite moderated nuclear reactor. It is essential that the fuel has a low burn-up so that the transuranium isotopes formed consist almost entirely of plutonium-239.

The United States had no access to heavy water in the Manhattan Project, so only graphite was used as a moderator in the reactors.[2]

The nuclear technology underwent continued rapid growth during the post-war years. The cold war meant further development of nuclear weapons technology. The access to enriched uranium made way for the development of very compact light water reactors for use in submarines.

Various civilian uses were investigated, including ship vessel propulsion, but it was nuclear reactors for electricity generation that became the dominating application. Three

---

[1] Two methods were applied for the enrichment: mass spectrometry and gas diffusion. In mass spectrometry ionic species of uranium are accelerated in vacuum and subjected to a strong magnetic field. The deviation of the trajectories in this field is slightly different for the two isotopes, and they can be collected at different target areas. The diffusion process is based on the fact that the diffusion is slightly different for gaseous species of uranium. (Uranium hexafluoride is used for this purpose, and fluorine has the advantage of having only one isotope).

[2] A moderator slows down the neutrons formed in the fission process. Low energy neutrons (thermal neutrons) are much more efficient for fission processes than fast neutrons and are essential for the neutron economy.

In a nuclear reactor, moderation competes with absorption. Carbon atoms have a mass that is considerably higher then that of a neutron and graphite is therefore a less efficient moderator than heavy water or light water. Light water is the most efficient moderator, but absorbs neutrons to some extent and can therefore only be used in conjunction with fuel that is somewhat enriched in uranium-235. Since large volumes of graphite are required in a graphite moderated reactor, it is essential that the graphite is very pure so that the absorption of neutrons is sufficiently small.

types of moderators are used in civilian reactors today: light water, heavy water and graphite. Most reactors use light water, but graphite moderated reactors were designed and used in the former USSR, and heavy water reactors are used in Canada. The high efficiency of the moderation of the light water enables the corresponding reactors to use a pressurized vessel for the entire reactor. For the other moderators, pipe designs are common. The pipes surround the fuel but not the main part of the moderators, and thus the fluid in the pipe can be pressurized and also take up the very most of the energy released.

The pressurized light water reactor used widely today for electricity generation has a design that is similar to that of the early submarines. Alternative reactor design principles were studied intensely internationally in the early days of nuclear technology, but have with few exceptions[3] received little attention during the last several decades. However, an number of studies have dealt with the thorium cycle[see e g 18] for several reasons including less long-lived transuranics and non-proliferation. Heavy water moderation constitutes a significant part in these studies.

There are a number of other reactor types that have been studied, e g Magnox and AGR reactors (gas cooled reactors) as well as breeder type of reactors. They are not dealt with here because they have not had any influence of any magnitude on the nuclear development in the Nordic countries.

Waste management (together with reactor safety) has been a dominating issue since the nineteen seventies. It was realized that attention had to be paid also to protection of the environment ant to the long-term safe disposal of nuclear waste.

Perhaps somewhat later came the full realization of the significance of the nuclear legacy in terms of decommissioning and dismantling.

1.3 Nuclear technology development in the Nordic countries

It was realized also in Germany during the war that it might be possible to utilize controlled nuclear chain reactions as well as nuclear explosives.

Essential in this regard is the availability of uranium and a moderator. It has already been said that heavy water is more efficient than graphite, and thus a more compact reactor might be designed if heavy water is available.

Through the occupation of Norway, Germany had access to the heavy water generated as a byproduct at the Norsk Hydro A/S water electrolysis plant at Rjukan.[4] The plant was, however sabotaged through a combined action of the Norwegian resistance movement and allied forces. Nonetheless, a shipment of 614 litres went underway to Germany, but was sabotaged and sunk deep in a the lake Tinnsjø. It has been

---

[3] E g nuclear reactors for space ships.

[4] There is a strong isotope effect in electrolysis. Enrichment of heavy hydrogen can therefore be achieved in an electrolysis plant for water by applying appropriate "logistics" for the water used.

assessed[15] that this quantity might have been just what was needed in order for the Germany to succeed in her experiments on a nuclear reactor.

After the war it was realized that the heavy water could provide an important basis for a domestic Norwegian Nuclear programme[14-16,19]. The first Nordic research reactor was commissioned at Kjeller in Norway already in July 1951, preceded only by facilities in Canada and the four great powers United States, The Soviet Union, Great Britain and France.[19] It was clearly stated that "*the project should be open and without any secrecy arrangements*" and that the Institute for Atomic Energy, IFA, should aim at establishing co-operation with other countries having similar approaches, e g Sweden and France. (In 1980 the Institute for Atomic Energy, IFA, changed its name to Institute for Energy Technology, IFE.)

The five Nordic countries became active participants when new international organisations were planned in the nineteen fifties and it was in Norway that the first international nuclear conference was organised already in 1953.[20] This was two years before the conference on the Peaceful Uses of Atomic Energy (*The Geneva conferences*) held by the United Nations.

At the time of the commissioning of the JEEP 1 reactor (in Norway) in 1951, the great powers had control over most of the uranium available. Nonetheless, IFE managed to purchase uranium from the Netherlands. This contract also included co-operation, which continued in various forms for a long time. The moderator and medium for heat transfer used in the core of the JEEP 1 reactor was heavy water, which was obtained domestically. The core was surrounded by a reflector made of graphite that was obtained from France.

The first Swedish nuclear research reactor was located at the Royal Institute of Technology in Stockholm and was commissioned in 1954 (see Appendix E and Reference [21]). The moderator consisted of heavy water and the natural uranium for the fuel (three tonnes) was "borrowed" from France.[16,20] Sweden has huge natural resources of uranium. At the time, uranium-bearing shale was mined for oil production. An auxiliary mineral in this shale is "kolm" the ash of which contains percentage quantities of uranium. Such uranium was beneficiated from 1953 at a capacity of five tonnes per year.

Self-sufficiency was important and Denmark (Kvanefjeldet, Greenland), Norway (Einerkilen) and Sweden (Kvarntorp and Ranstad) had domestic programmes for uranium mining, beneficiation and processing. Iceland had natural resources in terms of hydropower which relates to beneficiation of heavy water.[20]

Denmark acquired two reactors from the United States in 1956, and a larger one from Great Britain in 1957.[20] They all used enriched uranium in the fuel. The small training reactor used uranium dissolved in a liquid homogeneous liquid reactor, and this concept was subsequently studied in Denmark for power generation purposes.

Finland started its nuclear technology in 1956 by a subcritical pile, which used natural uranium as fuel and light water as moderator. Next step was the purchase of a TRIGA reactor from USA and to balance the political situation small amount of enriched fuel for the subcritical pile was bought from the Soviet Union in order to increase the reactivity of the subcritical pile. In both purchases there was a third party, IAEA in the agreements. The TRIGA reactor went critical in 1962 and has been in operation since that time.

Initially, the purpose of the research and development work in the Nordic countries was very broad, and military applications were not excluded until around the late nineteen fifties. Civilian applications included ship vessel propulsion, although no specific reactors were tested for such purposes.

Important prerequisites for the work included independence with regard to the resources required, and to keep options open with regard to e g reprocessing, enrichment and moderator requirements (absorption to moderation ratio, and moderator efficiency).

In Sweden, "the Swedish strategy" ("den svenska linjen") was established and applied. It consisted of use of heavy water (from Norway) as a moderator and natural uranium, mined and processed domestically. In addition, reprocessing was included, and comprehensive research and development work in this area was carried out at IFA in a Nordic collaboration. The pilot plant for reprocessing ("Uranrensanlegget") at IFA was commissioned 1962 and decommissioned in 1968.

Further research and development facilities in the Nordic countries include the JEEP 2 (2 MW) and the Halden (25 MW) heavy water reactors in Norway. In Sweden, the R2 (50 MW) light water reactor was commissioned in 1961 and shut down in 2005.

The first reactor for energy generation in the Nordic countries was the Ågesta heavy water reactor (65 MW, 10 MW for electricity generation and 55 MW for district heating) in the southern part of Stockholm. It was commissioned in 1963 and shut down in 1973.

All in all there is a fair number of facilities that have been commissioned and operated at different stages in the overall progress and for various purposes. The locations of the main nuclear sites in the Nordic countries are presented in Figure 1-1. An overview of the various major facilities at these sites is presented in Section 2.

The early work on nuclear technology development included a lot of co-operation between the various research establishments in the Nordic countries, and further information on this can be found in [20, see also 14-16,19,22]. This situation contrasts to that of power generation in the larger facilities commissioned from 1970 in Sweden and Finland, which mainly concerns these two countries.

Nordic co-operation in the fields of nuclear technology and safety have kept on in new areas of common interests, see [20].

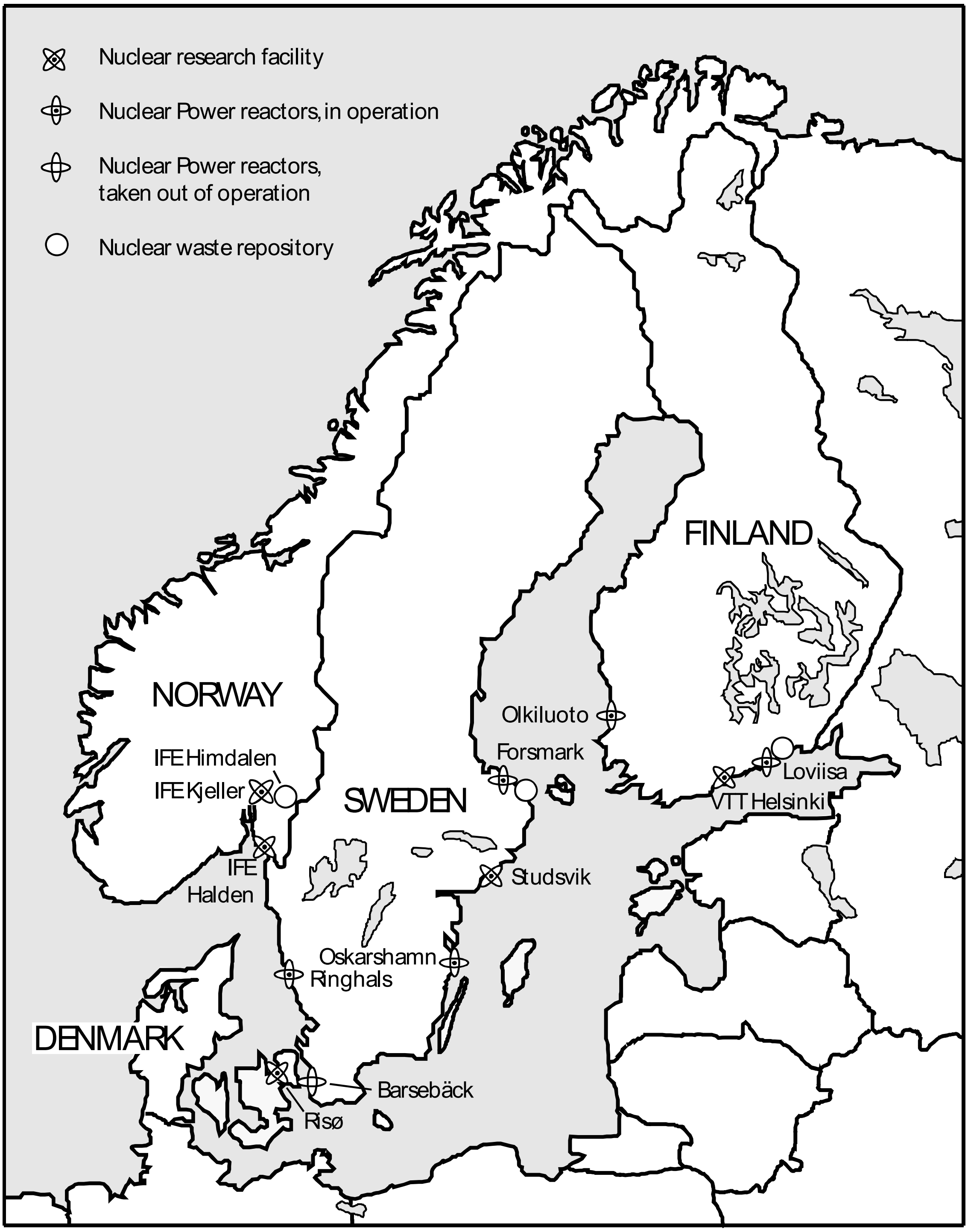


Figure 1-1. The locations of the main nuclear sites in the Nordic countries.

1.4 Present systems in the Nordic countries for funding decommissioning of nuclear research facilities

1.4.1 Denmark

In Denmark the only existing nuclear facilities are the above mentioned research facilities at the Risø National Laboratory. The Risø National Laboratory is owned by the state, and therefore the decommissioning costs will be paid by the state. The following text is taken from Reference [23] which is included in full in Appendix A, see also Appendix B and Section 5.

As part of Risø's strategic planning in 2000, it was taken into account that the largest research reactor, DR 3, was approaching the end of its useful life, and that the decommissioning question was becoming relevant. Since most of the other nuclear activities at Risø depended on DR 3 being in operation, it was decided to decommission all nuclear facilities at Risø National Laboratory once the reactor had been closed. Therefore, a project was started with the aim to produce a survey of the technical and economical aspects of the decommissioning of the nuclear facilities.

The survey should cover the entire process from termination of operation to the establishment ofa "green field", giving an assessment of the manpower and economical resources necessary and an estimate of the amounts of radioactive waste that must be disposed of. The planning and cost assessment for a final repository for radioactive waste was not part of the project. Such a repository is considered a national question, because it will have to accommodate waste from other applications of radioactive isotopes, e.g. medical or industrial.

In September 2000 Risø's Board of governors decided that DR 3 should not be restarted after an extended outage. The outage was caused by the suspicion of a leak in the primary system of the reactor, and followed after the successful repair of a leak in a drainpipe earlier in the year. Extensive inspection of the reactor tank and primary system during the outage showed that there was not any leak, but at the same time some corrosion was revealed in the aluminium tank. According to the inspection consultant the corrosion called for a more frequent inspection of the tank. Therefore, the management judged that the costs of bringing the reactor back in operation and running it would outweigh the benefits from continued operation in the remaining few years of its expected lifetime.

The closure of DR 3, of course, accentuated the need for decommissioning planning and for the results of the above-mentioned project. By the end of February 2001, the project report [24] was published. The study was followed by other studies in order to prepare a proposal for legislative action by the parliament to provide funding for the decommissioning. Among other aspects, possible decommissioning strategies were evaluated. Two overall strategies were considered, (1) an irreversible entombment where the nuclear facility is covered by concrete and thereby transformed into a final repository for low- and medium level waste, and (2) decommissioning to 'green field' where all buildings, equipment and materials that cannot be decontaminated below established clearance levels are removed. The entombment option was rejected rather quickly as not being acceptable, among others for ethical reasons ("each generation should take care of its own waste"). Instead, three different decommissioning scenarios were considered with 'green field' as the end point, but with different durations, viz. 20, 35 and 50 years, respectively.

After thorough preparations, including an Environmental Impact Assessment, the Danish parliament in March 2003 gave its approval to funding the decommissioning of all nuclear facilities at Risø National Laboratory to "green field" within a period of time up to 20 years. The decommissioning is to be carried out by a new organisation, Danish Decommissioning (DD), which is independent of Risø National Laboratory, thus avoiding any competition for funding between the decommissioning and the continued research activities at Risø.

In the year 2000 the Minister of Research and Information Technology requested that a survey be conducted which comprises the entire process of decommissioning from termination of the operations to the establishment of "green field" conditions. As a result, a report was published in 2001 [24] with descriptions of the above mentioned facilities together with cost calculations. During the project it became evident, however, that for many of the decommissioning tasks the extent of the work and the costs can only be assessed with considerable uncertainty (± 30 %) at that stage. More detailed assessments of the decommissioning costs are to be conducted during the more detailed planning of the decommissioning projects for each facility.

1.4.2 Finland

The nuclear waste management plan is based on immediate dismantlement after the final shutdown of the reactor. Experienced personnel will still be available to conduct the decommissioning work. It is intended that the decommissioning waste should be disposed of in the repository constructed in the bedrock of the Loviisa nuclear power plant site at the depth of 110 m. At the moment preparatory work has been done to clarify the possible problems of the decommissioning waste of the TRIGA research reactor (cf Section 1.4.2) in the surroundings of decommissioning waste of the nuclear power plant. The Finnish goal is to work out an agreement between VTT and the Loviisa NPP about the final disposal of decommissioning waste in the said repository.

The decommissioning waste studies concentrate mainly on the long term safety of the decommissioning waste disposal. The main part of the active reactor components will be packed in concrete packages in the waste disposal facility, which means an additional barrier against the ground water flow. Among others the amount and behaviour of some long-lived radioactive isotopes like ^{14}C belong to these studies. TRIGA reactors typically have plenty of irradiated components consisting of graphite.

In Finland, the producer of nuclear waste is fully responsible for its waste management. The financial provisions for all nuclear waste management have been arranged through the State Nuclear Waste Management Fund. The cost estimate of the nuclear waste management is to be sent annually to the authorities for approval. Based on the approved cost estimate, the authorities are able to determine the assessed liability and the fees to be paid to the Fund [25]. The main objective of the system is that at any time there shall be sufficient funds available to take care of the nuclear waste management measures associated with the waste produced up to that time. The details can be found in the Finnish legislation [26]. The funding system is applied also to government institutions like the FiR 1 research reactor operated by the VTT.

1.4.3 Norway

There exist no funding for decommissioning of Norwegian nuclear research facilities today. It is IFE:s opinion that this is a national responsibility in Norway. The question of funding of decommission of these facilities will be elucidated by the Norwegian Ministry of Trade and Industry.

1.4.4 Sweden

It has been described in Section 1.3 (see also Section 1.4) that substantial development work was carried out before and in conjunction with the introduction of nuclear power in Sweden, and much of it took place in the facilities at the Studsvik site. Consequently, it has been decided that that it is those who benefit from the electricity generated by the nuclear power plants who shall pay the costs for the decommissioning, decontamination, dismantling and waste management which is required when the old research facilities at are no longer needed.

Thus, the Law on financing of the management of certain radioactive waste e t c (SFS 1988:1597) states (§1) that "*fee shall be paid to the Government in accordance with this law as a cost contribution*" to amongst other things "*decontamination and decommissioning of*" a number of facilities listed in the law.

The Ordinance (SFS 1988:1598) on financing of the handling of certain radioactive waste e t c states (§4) that the funds collected should be paid to cover the costs incurred. It also states (§4) that "*payment will be carried out only for costs which are needed for*" the decontamination and commissioning "*and which have been included in the cost estimates*" required.

According to the Law on financing of the management of certain radioactive waste e t c (SFS 1988:1597, §5), cost calculations shall be submitted to the Swedish Nuclear Power Inspectorate (SKI) each year. They shall comprise estimates of the total costs as well as the costs expected to be incurred in the future with special emphasis on the subsequent three years.

The Swedish Nuclear Power Inspectorate (SKI) has the responsibility (SFS 1988:1598, §5) to review the cost estimates and to report to the Government if there is a need to change the level of the fee. The SKI also has the responsibility (SFS 1988:1598, §4) to decide on the payments to be made.

It might be added that according to its instruction (SFS 1988:523, §2) SKI also has the responsibility "*in particular … to take initiative to such … research which is needed in order for the Inspectorate to fulfil its obligations*". The participation in the present project is an example of such an undertaking by SKI.

The legislation referred to above can be downloaded from SKI's website (www.ski.se) or from Rixlex (www.riksdagen.se/debatt/).[5]

1.5 Rationale for Nordic co-operation on decommissioning

The present study has come about on initiative by the Swedish Nuclear Power Inspectorate (SKI) and is based on a common need in Denmark, Finland, Norway and Sweden.

It was found in various studies carried out on commission by SKI (see e g [7-11] where [10] is included in the present report in the form of Appendix F) that the intended functioning of a system for finance requires a high precision even in the early stages of cost calculations, and that this can be achieved only if the planning for decommissioning is relatively ambitious. The following conclusions were made:

- IAEA and OECD/NEA documents provide invaluable advice for pertinent approaches.
- Adequate radiological surveying of a facility is needed before precise cost calculations can be made.
- The same can be said about technical planning including selection of techniques to be used.
- It is proposed that separate analyses be made regarding the probabilities for conceivable features and events which could lead to significantly higher costs than expected.[6]
- It is expected that the need for precise cost estimates will dictate the pace of the radiological surveying and technical planning, at least in the early stages.[7]
- It is important that the validity structure for early cost estimates with regard to type of facility be fully appreciated. E g, the precision is usually less for research facilities as compared to nuclear power plants.[8]
- The summation method is treacherous and leads to systematic underestimations in early stages unless compensation is made for the fact that not all items are included at early stages (since they cannot be identified then).

---

[5] The latest development on the so-called Studsvik fund can be found in the proceedings of the upcoming meetings: "*the 2008 Avignon International Conference on decommissioning, dismantling, decontamination and reutilization which will be held in Avignon, France, September, 28th to October, 2nd, 2008*", and "*Environmental economics 2008. Second International Conference on Environmental Economics and Investment Assessment 28 - 30 May, 2008, Cadiz, Spain*".

[6] In practice, in most cases discovery of unexpected features leads to additional costs.

[7] This is clearly the case in countries where funds are collected far in advance of the decommissioning operations. Otherwise, pace may be dictated by the technical planning and the associated cost estimates.

[8] This has to do with the research facilities being more different in comparison with each other which makes it less efficient to apply previous experience. They are also smaller which makes it more difficult to rationalize the work.

- Comparison between different facilities can be made when there is access to information from plants at different stages of planning and when accommodation can be made with regard to differences in features.
- A simple approach was presented [9-10] for “calibration” of a cost estimate against one or more completed projects.
- Information exchange and co-operations between different plant owners is highly desirable.

These conclusions are in concordance with and are supported by a recent report by an expert group at the IAEA[5].

Denmark is presently moving ahead with the implementation of the decommissioning of its old research facilities and have already completed the work on their first reactor. A thorough planning – including cost calculations – was carried out before the practical work was started. The experience from this approach is very positive.

The pre-studies carried out in Finland and Norway, as well as the previously completed decommissioning of the Uranium Reprocessing Pilot Plant (“Uranrensanlegget”) at Institutt for Energiteknikk (IFE), also clearly indicate the necessity of appropriate technical and financial planning. The work at the Norwegian pilot plant also showed the importance of associated development work.[19,27]

Information exchange and co-operation on decommissioning of old nuclear research facilities – among owners, contractors, and authorities – will improve the efficiency of the planning and implementation processes. For such systems for finance where funds are to be collected now and costs are to be incurred in some future, such interactions are even necessary prerequisites since experience and data on finished and on-going projects are needed for assessments regarding future ones. (This is explained further in Section 4.2.2.)

1.6 Purpose

The purpose of the present work is to identify what knowledge and methodology is required for sufficiently precise cost calculations for decommissioning of nuclear research facilities. The purpose is also to exchange and compile[9] such information, data and methodology so that they become available in a suitable format. Furthermore, the purpose is to establish a Nordic network for information exchange and co-operation.

The work is to be carried out during a period of three years, and the present report presents the findings from the first and second year.

The emphasis has been on networking, collection and compilation of data and guidance documents as well as on schemes of calculation. The focus during the third year is on the establishment of a searchable database.

---

[9] I e make searchable and comparable.

It has been assessed [8-10] that a confidence level of 80 % might be attained even at a relatively early stage. It is highly important in this regard that differentiation is made with regard to stage of planning, cf [1,28].

1.7 Scope

The scope of the present work is as follows:

1 Establishment of a Nordic network in the field including an Internet based expert system

2 A guidance document for the prerequisites for precise cost calculations, including
   - radiological surveying
   - the technical planning
   - financial risk identification

3 Descriptions of techniques that may be applied at early stages of calculations and assessments of costs

4 Collection and compilation of data for plants, state of planning, organisations, e t c.

2 Present status of major Nordic facilities for nuclear technology development

The nuclear technology development that lead to the establishment of the present facilities and sites is described in Sections 1.3 (the Nordic countries) and 1.2 (international development in general). The locations of the main nuclear sites in the Nordic countries are presented in Figure 1-1.

2.1 Denmark

Facilities of interest to consider for the proposed information exchange e t c, cf below, are as follows. (It is not expected that each participant will include all of its facilities listed in the project work).

Risö, Denmark

- DR 1. A 2 kW thermal homogeneous, solution type research reactor which uses 20 % enriched uranium as fuel and light water as moderator.
- DR 2. A tank type, light water moderated and cooled reactor with a power level of 5 MWth. It was finally closed down in 1975 and was later partially decommissioned.
- DR 3. A research reactor built to test materials and new components for power reactors. It uses ≈ 20 % enriched uranium and is moderated and cooled by using heavy water. The power output is 10 MWth.
- Fuel fabrication facility (for the DR 3 reactor)
- Isotope laboratory. Management of irradiated samples.
- Hot cell laboratories. Six concrete cells used for post irradiation investigations. The facility has been partially decommissioned.
- Waste management plant and storage facilities

All the heavy nuclear research facilities in Denmark have been taken out of ordinary operation.

The reseach reactor DR1 was decommissioned during 2005 and the reactor building and site area have been free released without restrictions by the Danish nuclear authorities. The research reactor DR2 is presently (2008) at al alte stage of decommissioning and the site is planned to be free released without restrictions around the first quarter of 2009.

Further information on the Danish programme can be found in Appendices A and B.

2.2 Finland

Otaniemi, Espoo, Finland

- FiR 1. A 250 kW TRIGA research reactor, operated since 1962. A special U - ZrH_x - fuel, uranium enrichment 20 %. Light water moderated. The main purpose of the operation of the reactor is BNCT (Boron Neutron Capture Therapy) as well as isotope production.
- Radiochemical laboratory
- Hot cell laboratory with e g testing of irradiated steel samples from nuclear power plants, especially samples from pressure vessels

In particular, an environmental impact assessment work of the decommissioning of the reactor is planned to be carried out next year.

Further information on the TRIGA research reactor can be found in Appendix C.

2.3 Norway

The *Institute for Atomic Energy* (IFA) in Norway changed its name to *Institute for Energy Technology* (IFE) in 1980.

2.3.1 Overview

The major nuclear facilities in Norway in operation or decommissioned are:

- JEEP I, a 450 kWth research reactor at IFE, Kjeller.
- The NORA zero-effect research reactor at IFE, Kjeller.
- The Uranium Reprocessing Pilot Plant at IFE, Kjeller
- The Halden Boiling Water Reactor (HBWR) a 25 MWth research reactor at IFE, Halden.
- JEEP II, a 2 MWth research reactor at IFE, Kjeller.
- The radioactive waste treatment plant and storage facilities.
- Metallurgical laboratory II for post irradiation investigations of test specimens of fuel and other materials.

Short descriptions of these nuclear facilities are given below. According to the licence for operation of existing facilities, the Norwegian Radiation Protection Authority (NRPA) has required preparation of decommissioning plans for each of these facilities. IFE has thus prepared decommissioning plans according to IAEA:s recommendations for “ongoing plans” during the operation of the facilities and to “stage 1: Storage with surveillance” or “stage 2: Restricted site use” as long as this is not in conflict with storage of spent nuclear fuel and long lived intermediate level radioactive waste. Recently the NRPA has asked IFE to take another step forward and extend these decommissioning plans to “green field”, and this was conducted in the year 2007.

2.3.2 Decommissioned facilities

JEEP I

The Dutch-Norwegian co-operation in the field of atomic energy was established in April 1951. The aim of the co-operation was at the time to complete the heavy water uranium reactor constructed at IFA, Kjeller in Norway. It was decided that a Joint Commission, consisting of three Norwegian members and three Dutch members, should lead further work in atomic energy in the two countries. The establishment at IFA, Kjeller, was included a Dutch-Norwegian organisation called Joint Establishment for Nuclear Energy Research (JENER). [29]

Operation started:	June 1951
Operation terminated:	December 1966
Thermal power from 1951 to 1956:	100 kW
Thermal power from 1956 to 1966:	450 kW
Fuel:	Natural metallic uranium, 2448 kg
Moderator and cooling:	Heavy water
Moderator temperature:	Around 50 °C at 450 kW
Pressure:	Atmospheric pressure

In 1956 the heat exchanger was replaced with a larger one and the capacity of the cooling of the light water system was improved by installation of a cooling tower. The thermal power of the reactor could then be increased to 450 kW. [30]

In April 1960 a leakage in the heavy water circuit was detected, necessitating the replacement of the reactor vessel. The reactor was started up again in October 1960 with a new reactor vessel. [30]

Today the reactor has been emptied of fuel and heavy water. The spent fuel is stored at IFE, Kjeller. The reactor vessel including the biological shielding is still not dismantled. The building containing the reactor is now used for housing a ^{60}Co irradiation facility.

There were several purposes of the JEEP I reactor. Atomic energy was a new and promising energy source in the 1940s and 1950s and reactor operation and reactor physics were two major fields of study.

Before JEEP I was built Norway had to import radioisotopes for medical and industrial use. Long delivery time, high transportation costs and problems with short-lived nuclides made it desirable to start production of radioisotopes in Norway. Research on production of radioisotopes for medical use and reactivation of radioisotopes for industrial use started in 1951-1952. During the period of 1952-1962 the production of radioisotopes increased tenfold and more than 75 % of the production was for medical use. The other Nordic countries showed at an early stage great interest in the Norwegian isotope production and exports of these products increased steadily. In addition to export of radioisotopes to the Scandinavian countries IFA also exported some products to the Netherlands and to a lesser extent to other European countries. [30]

After the start in 1951 it was possible to take up studies of neutron physics first by measurements of reactor characteristics and neutron- and γ-spectrometry. After building neutron diffractometers, fundamental studies of solid-state physics could be conducted. [30]

The NORA reactor

Based on the experiences for operation of the JEEP I reactor it was soon realised that its possibilities for reactor physics studies were limited and that flexibility is of greatest importance in this field. A plan for a "zero-effect" reactor (only a few watts), the NORA reactor, was therefore worked out in the course of 1958.

In January 1960 an agreement was signed between IFA and the International Atomic Energy Agency (IAEA) to put the NORA reactor at IAEA's disposal for a common reactor physics program. The IAEA contribution was to provide a fuel charge for the common operation. NORA also made it possible to continue and extend the work carried out with the ZEBRA-assembly in Stockholm by a joint Swedish-Norwegian-Dutch team. [30]

Operation started:	1961
Operation terminated:	1966
Thermal power:	Zero-effect (50 W)
Fuel:	UO_2 enriched to 3.41 wt% in ^{235}U
Weight of fuel in fuel element:	1598 $\pm$ 15 g U_2O
Moderator and cooling:	H_2O/D_2O (sometimes mixed)
Moderator temperature	Room temperature
Pressure	Atmospheric pressure

Variable core configuration, number of reference core configurations: 4
Configuration 1: Number of fuel elements = 248,
Configuration 2: Number of fuel elements = 240
Configuration 3: Number of fuel elements = 348
Configuration 4: Number of fuel elements = 424

This reactor would serve as an instrument for the reactor physicists in their work on the determination of fundamental physics problems and physics parameters for planned core geometries and fuel elements for both light water and heavy water reactors.

The reactor was housed in the "NORA" building which now is connected to the JEEP II reactor-building complex. The reactor is now completely decommissioned.

The Uranium Reprocessing Pilot Plant at IFA, Kjeller

Operation started:	1961
Operation terminated:	1968

The emphasis of this Norwegian-Dutch reprocessing pilot plant was on experimental reprocessing of natural uranium fuel elements from the research reactor JEEP I, and testing of the "Purex" process equipment, instrumentation and various flow sheets, especially for Eurochemic in Mol, Belgium. Another objective was to obtain operation experience and know-how for the design of a full-scale plant. The Swedish "AB Atomenergi" completed an additional facility in 1964 with the intention to study a separation process using a silica gel column. The Norwegian –Dutch "Purex" part and the Swedish "Silex" part were connected in 1964 to increase the purification capacity.

In the operation period about 1200 kg of uranium was processed, and plutonium and fission products separated by means of liquid-liquid extraction. The plant comprised a tube system of more that 6000 meters and a total of 50 tanks, evaporators and extraction columns.

The plant was shut down and partly decontaminated in 1968. The dismantling was delayed due to economic constraints and re-started in 1982 for one-year period. The decommissioning was resumed in 1989 and continued during the period 1989-1993 [31]. The purpose of the decommissioning was to remove radioactive and contaminated materials so that the building could be used for radwaste work. This required decommissioning to "Stage 2: Restricted site use" and "Stage 3: Unrestricted site use" according to IAEA nomenclature.

2.3.3 Facilities in operation

The Halden Boiling Heavy Water Reactor (HBWR) at IFA, Halden

The Halden Boiling Water Reactor (HBWR) was built by the Norwegian Institute for Energy Technology during the years 1955-1958 (as Institute for Atomic Energy) after a resolution by the Norwegian parliament and government. A photograph from the reactor is shown in Figure 2-1. From 1958 the Halden Reactor Project was established as a joint undertaking of the OECD Nuclear Energy Agency. An agreement was drawn up between nuclear organizations of different OECD countries sponsoring an experimental research programme to study the HBWR concept. The Institute for Energy Technology is the owner and operator of the reactor installation. The reactor operation is thus solely governed by Norwegian laws and regulations.

The HBWR does not produce any electricity but delivers process steam to the nearby paper mill (Norske Skog Saugbrugsforeningen).

Today the Halden Research Project has 17 member countries with more than 100 participating organisations. The project is operated in three- year programme periods.

Operation started:	June 1959
Operation terminated:	Still in operation
Thermal power:	25 MW
Standard fuel:	UO_2 enriched to 6 wt% in ^{235}U
Moderator and cooling:	14 tons of heavy water
Operation temperature:	240 °C
Pressure.	33.6 bar

The Halden Boiling Water Reactor (HBWR) started up in June 1959 and is still in operation. The core consists of standard fuel assemblies and test assemblies. The total number is in the range 80 – 120, of which around 20-35 are test assemblies. The standard fuel assemblies consist of UO_2 fuel rods with 6 wt % ^{235}U enrichment. The total mass of fuel in the core depends of the test program and will be in the range 400 – 600 kg. The reactor is located in a mountain hall that also serves as containment for the reactor. [32]

The main purpose of the HBWR is to carry out experiments to gain knowledge of optimal and safe operation of reactors and power plants over extended periods of time. Instrumentation of the test fuel assemblies has made it possible to make advanced studies in fuel-, material- and corrosion technology. Since the Swedish R2 reactor at Studsvik has been closed down an agreement between IFE and Studsvik has been signed for using the HBWR for experiments.

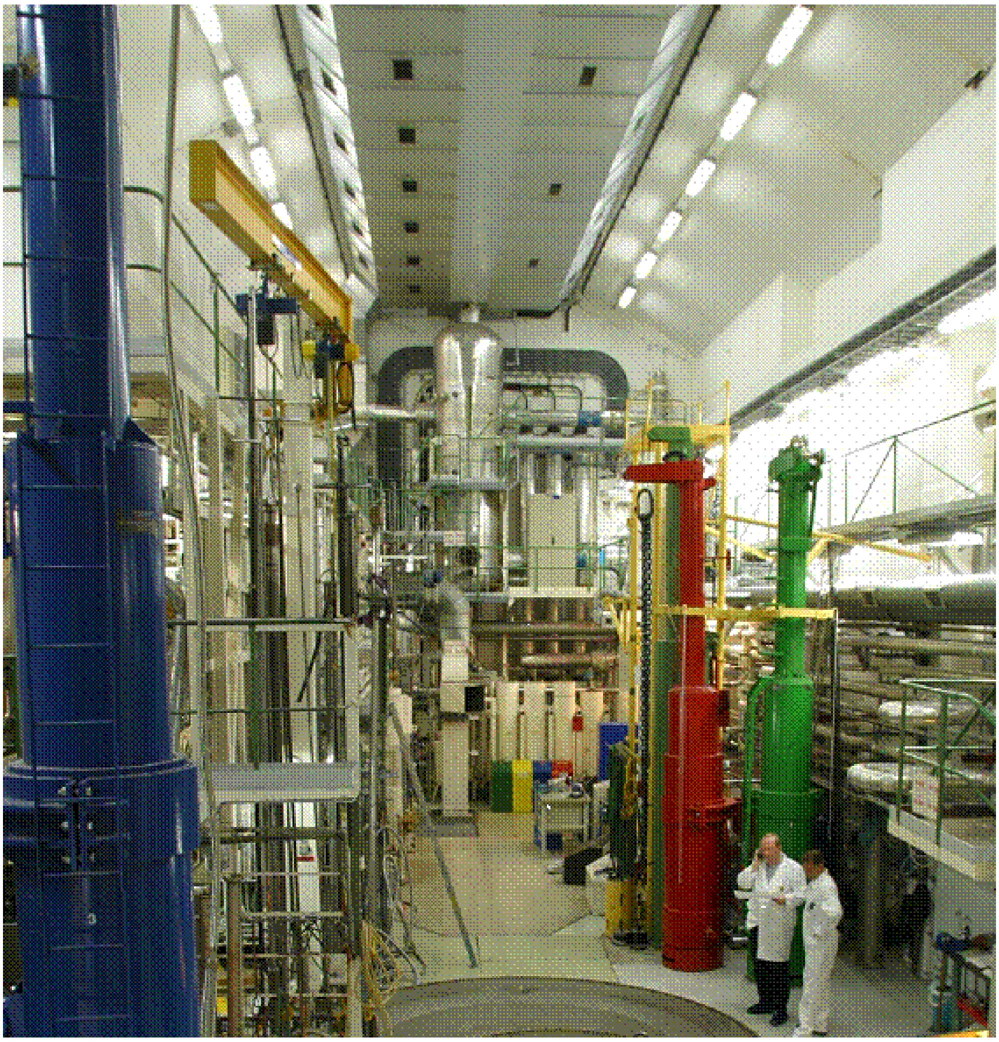


Figure 2-1. The Halden Boiling Heavy Water Reactor (HBWR) at IFA, Halden, Norway.

This licence period for operation the HBWR will terminate 31. December 2008. IFE has applied for a 10 years licence period for operation of the HBWR from 2009.

The JEEP II reactor at IFE, Kjeller

At the end of 1960 the JEEP I reactor had been in operation for about 10 years and a more modern research reactor with greater experimental possibilities was required. The dominant demand was for a higher neutron flux for the neutron physics work which was carried out at IFA, Kjeller, forming the main line of the academic research activity. This work was limited by the low neutron flux and the inadequate number of beam channels for physics experiments. The planning of the new research reactor, the JEEP II, was therefore started in 1959.

Operation started:	June 1967
Operation terminated:	Still in operation
Thermal power:	2 MW
Fuel:	UO_2 enriched to 3.5 wt% in ^{235}U, 250 kg
Number of fuel assemblies	19
Moderator and cooling:	5 tons of heavy water
Operation temperature:	55 °C
Pressure.	Atmospheric pressure

The reactor is housed in a steel containment and is operated approximately 10 months each year. This licence period for operation of JEEP II will terminate on 31st of December 2008. IFE has applied for a 10 years licence period for operation of JEEP II from 2009.

A photograph of the reactor is shown in Figure 2-2.

The core of the reactor has 51 vertical channels for fuel assemblies, control rods and for experiments, and 9 positions in the reflector for irradiation of silicon crystals and for isotope production. The reactor also has 10 horizontal beam channels where neutrons can be utilised for physics experiments outside the biological shield of the reactor.

The reactor is extensively used for doping of silicon crystals to produce semiconductors. Doping by use of neutrons gives a more homogenous doping throughout the crystals than other methods. Up to summer 2000 only silicon crystals having diameters of 3 " or less could be irradiated. In the autumn 2000 the reactor was stopped and a new top lid was built in order to enable irradiation of silicon crystals with diameters up to 5 ".

The reactor is also used for production of radioactive sources for industrial and scientific use. Radioactive isotopes can be used as tracers for studies of physical and chemical processes. Tracers are extensively used in detection of movements of fluids in oil reservoirs. Radioactive isotopes for use in nuclear medical diagnostic examinations are also produced in the reactor. Another use of the reactor is neutron activation analysis. This is a much-used method in environmental technology and pollution studies.

One of the main uses of the JEEP II reactor is to supply neutrons for studies of static and dynamic structures in solid materials and liquids. The method used is neutron scattering and has many advantages in studies of materials of high importance for storage of hydrogen and studies of nano-particles.

Figure 2-2. The JEEP I reactor at IFE, Kjeller, Norway.

The Radioactive Waste Treatment Plant at IFE, Kjeller

The production of radioactive isotopes for medical use form 1951 resulted in radioactive waste products. The operation JEEP II also resulted in some radioactive waste. Up to 1954 this waste was collected and stored. In 1954 IFA was grated the permission from Statens Radilogisk-Fysiske laboratorium (now Norwegian Radiation Protection Authority) to discharge specified amounts of liquid radioactive waste to Nitelva river close to IFAs facilities at Kjeller in Norway. Unfortunately IFA had applied for permission to the wrong authority and this wrong authority had granted the permission. The discharge of liquid radioactive waste had therefore to be stopped in 1957 and the liquid waste must once again be collected and stored at IFA.

Planning of a radioactive waste treatment facility was started in 1957. The radioactive waste treatment facility was tested in 1961 and taken into ordinary use from 1962. The facility treated liquid radioactive waste to reduce radioactivity levels before discharges to Nitleva in accordance with discharge permissions given by the authorities. The facility also treated and stored solid radioactive waste. The present licence period for operation of the Radioactive Waste Treatment Plant will terminate on December 31, 2008. IFE has applied for a 10 years licence period for operation of the HBWR from 2009.

Today the Radioactive Waste Treatment Plant receives waste from IFEs activities and from other users of radioactive materials and sources in Norway. It has been estimated that the volume of solid radioactive waste treated is 110 – 120 drum equivalents (equal 210 litre drums) per year. For IFEs own activity this comprises 80-90 drum equivalents and approximately 30 drum equivalents from other waste producing activities in Norway.

In 1970 the storage area for treated solid radioactive waste was filled to capacity. IFA was therefore granted the permission to establish a repository in clay at its premises at Kjeller in Norway. The repository contained 997 drums including 166 drums containing 35 grams of plutonium in a clay bed 2-3 meters below a lawn. Leakage from the repository was supervised by taking water and mud sampled from a drain sump at one end of the repository. Water from the repository running though the drainage sump was collected and treated in the Radioactive Waste Treatment Plant.

When the decision was made in the Norwegian Parliament to build a new storage and repository in Himdalen it was required that the old repository at IFE should be retrieved, the waste drums repacked into new drums and moved to the new repository in Himdalen. This operation was carried out in 2001. The free release limits for the clay bed were specified by the Norwegian Radiation protection Authority to 100 Bq/g dry weight for ^{137}Cs and 10 Bq/g dry weight for the sum of ^{239}Pu, ^{240}Pu and ^{241}Am. Testing of clay from the drums and in the clay bed showed levels of radioactivity below the free release limits. 200 m^3 of sediments from a clean up-operation at the end of an old discharge pipeline in Nitelva carried out in 2000 were filled into the empty clay bed. It had been proved that these sediments contained contamination levels below the free classification limits.

The Metallurgic Laboratory II

The Metallurgic Laboratory II (Met.Lab.II) at IFE, Kjeller, was built in the period 1961-1963 and has been in continuous operation since. A photograph from the laboratory is shown in Figure 2-3. The Nuclear Materials Technology department (NMAT) of the sector for nuclear safety and reliability at IFE operates the laboratory. The current licence period for operation the laboratory will terminate on December31, 2008. IFE has applied for a new10 years licence period for operation from 2009.

The main activities in the laboratory are:

- Production of UO_2-pellets and fuel rods for the two Norwegian test reactors JEEPII and HBWR.
- Production of instrumented experimental test fuel rods for the HBWR by refabrication and instrumentation of irradiated fuel rods and by encapsulation of MOX-fuel (Mixed Oxide Fuel).
- Post-irradiation examination of irradiated experimental fuel assemblies and rods.
- Examination of irradiated construction material samples.
- Management and storage of spent fuel and high-level radioactive waste.

The main part of the work at the laboratory is Post Irradiation Examination (PIE) of fuel rods and irradiated structural components tested in the HBWR.

The main installations in the Met.Lab.II are:

- A pilot production plant for experimental nuclear fuel rods with a complete line for fuel pellet production.
- A Hot Laboratory. The hot laboratory has several hot cells for the handling of high-level radioactive materials and sources. The hot laboratory has three concrete shielded cells with 1 m thick concrete walls and 4 windows with 1 m thick lead glass incorporated in the front wall of the caves. The cells are furnished with a periscope and movable equipment for non-destructive (NDT), destructive tests (DT), and benches for re-fabrication/instrumentation. Additionally there are separate lead shielded cells (4 + 1 + 1) with lead-glass windows furnished with various movable equipment for DT PIE, namely cutting devices, equipment for metallographic and chemical sample preparation, a macroscope, optical microscopes, etc. Work in the hot cells is done by using mechanical and electrical manipulators.
- Laboratories with glove boxes for work with non-irradiated fuel and MOX.
- Laboratories with fume hoods/boxes and partly shielded equipment for work with non-irradiated fuel and low radioactive materials.
- Auxilary installations such as an unloading bay for shipping flasks, storage pits, decontamination rooms, maintenance room for active components etc.
- A dry storage area for spent fuel from the JEEP II reactor, experimental fuel from the Halden reactor and high level radioactive waste. The storage consists of 84 vertical steel pipes in a concrete block blow the ground. The pipes are locked and shielded by lead plugs.

Nuclear materials stored at the laboratory are under continuous control and inspection by the International Atomic Energy Agency (IAEA) and by the Norwegian Radiation Protection Authority.

Figure 2-3. The Metallurgic Laboratory II at IFE, Kjeller, Norway.

2.4 Sweden

2.4.1 Overview

The major nuclear facilities in Sweden in operation or decommissioned are:

- The R1 research reactor at the Royal Institute of Technology
- The active Central Laboratory
- The R2 Research reactor
- The storage for old intermediate level waste
- The interim store for spent nuclear fuel
- The Hot Cell Laboratory
- The scrap melting facility

Short descriptions of these nuclear facilities are given below.

2.4.2 Decommissioned facilities

The R1 research reactor at the Royal Institute of Technology

The R1 research reactor at the Royal Institute of Technology is described in Appendix E, and the decommissioning work is described in Section 6.

The active Central Laboratory

The Active Central Laboratory (ACL) was commissioned in 1964 and was taken out of operation in 1997.

The facility was a qualified general purpose active laboratory and the use included the following:

- analysis of cladding and other materials
- decontamination and repackaging of glove boxes
- pyrolysis of ion exchange resin
- manufacturing of Sr-90-radiation sources
- mechanical workshop for radioactive components
- experiments with "radiation knife" for treatment of cancer tumours
- experiments with eluation of radioactive elements from ion exchange and the subsequent absorption on inorganic ion exchange material (zeolites)
- compaction of waste drums
- leach tests of glass from reprocessing
- storage and handling of fissionable and other radioactive material
- storage of uranium hexafluoride
- manufacturing of equipment for concrete solidification

- filter tests
- testing of materials
- manufacturing of isotope batteries and overvoltage surge protection
- laboratory for reactor chemistry
- gammacell for irradiation
- experiments with iodine in fuel
- etc

The facility is decommissioned and declassified.

Various international co-operation has taken place including OECD/NEA and the Nordic countries.

Further information can be found in [33].

2.4.3 Facilities taken out of operation, awaiting decommissioning

The R2 Research reactor

The reactors R2-0 and R2 were commissioned in 1960 and were taken out of operation in 2005. They have been used mainly for materials and fuel testing purposes, isotop generation and silicon doping.

The reactor building comprises reactor hall for the reactors and a cellar for auxiliary equipment. There are three pools, one for each of the two reactors and one for interim fuel storage.

The R2 reactor was of a tank type and had light water as moderator. The neutron flux was high and so was the level of enrichment. The thermal power was 50 MW.

The R2-0 reactor was of pool-type. Maximum power was 1 MW and it was cooled by natural convection.

The use of the R2 reactor has mainly been geared towards nuclear power generation issues and the incentive for Nordic co-operation has consequently been small.

Three alternatives are planned for the decommissioning. *Alternative 1* implies that the R2 building and auxiliary buildings, including the centre for isotope production are evacuated before the service operation for the decommissioning is incepted. *Alternative 2* includes emptying of the pool of the R2 reactor as well as the R2 building itself, but no further evacuation. *Alternative 3* implies continued operation of the systems for the R2 reactor including the maintenance of the integrity of the pool system for the purpose of radiation protection.

All three alternatives include the removal of the reactor fuel as well as active fuel specimens from the interim pool storage as a first step. Also a thorough cleaning and radiological surveying are included.

The special facility for spent fuel in pool storage need be prepared for receiving the fuel. Assessments need be made for fuel test pins as to whether they should be regarded as waste and managed for final direct disposal, or what should be stored for other dispositions, and where the appropriate storage is to take place.

References on the R2 reactor are [34] and [35].

The storage for old intermediate level waste

The storage for old intermediate level waste (SOILW) was erected in 1960 and taken into operation in 1961. All of the waste that was earlier stored in the facility has been removed and treated for continued storage in another facility in Studsvik. Contaminated parts and minor amounts of remaining radioactive material will be removed at a later stage. According to current plans, the facility will be used for investigation and reconditioning of historical waste. Decommissioning is planned for 2036-2039.

The main floor of the store is at ground level. The store includes pipe positions as well as concrete cells, all well shielded relative to the floor above. The atmosphere at the various positions is at a slight underpressure and the air is evacuated through a slit in the concrete construction underneath the storage positions.

There has been no Nordic co-operations related to this facility.

Further information can be found in [36].

The interim store for spent nuclear fuel

The interim store for spent nuclear fuel (ISSNF) was taken into operation in 1965 and has been used until recently for interim storage of spent fuel from the R1 and other reactors.

The facility is hosed in a separate building together with an auxiliary building. It comprises water filled pools for storage of irradiated fuel.

There are no plans at present to discontinue the operation of the facility.

There have been no Nordic co-operation projects.

The license of operation extends to the year 2014. In the planning for decommissioning and the associated cost calculations it is assumed that the decommissioning takes place in the year 2034.

Further information can be found in [37].

2.4.4 Facilities in operation

The Hot Cell Laboratory

The Hot Cell Laboratory was commissioned in 1960 and is still in operation. The Laboratory is important for the continued operation of the Swedish Nuclear Power plants and there are no plans for discontinuing the operation.

The Laboratory is used for investigation of radioactive material such as fuel elements, fuel rods and core components. It is designed for work with specimens having a high level of gamma radiation.

In the plan for decommissioning and the associated cost calculations it is assumed that the decommissioning of the facility will start in the year 2031.

There has been a conference around Hot Cells in the Nordic countries, and nowadays there is a European co-operation on the topic.

Further information can be found in [38].

The scrap melting facility

The plant was commissioned in 1960 for reprocessing of heavy water. During the 1970'ies it was exhaust gas laboratory under the auspices of the Swedish Environmental Protection Agency. In 1985 the scrap melting facility was taken into operation. The facility was substantially extended in 2005.

There are no plans for discontinuing the operation of the facility.

The plant is being used for handling and melting of low active scrap metal from the nuclear industry with the purpose of free release, recycling and volume reduction (of material that is to be stored).

The plant has facilities for sorting, fractioning, mechanical decontamination and melting of scrap metal. The operation is batchwise.

There exists a decommissioning plan.

There has been no Nordic co-operation in connection with this facility.

Further information can be found in [39].

3 Good practice

3.1 Strategy and planning

The overall purpose of decommissioning is actually the protection of man, the environment and natural resources. In the case of Sweden, the basis for this is defined in a law called "*The Environmental code*" (SFS 1998:808) . According to part one, chapter one, section one of this code, it "*shall be applied in such a way as to ensure that human health and environment are protected against damage and detriment, ... biological diversity is preserved, ... the use of land ... is such as to secure a long term good management ... and reuse and recycling ... raw materials and energy is encouraged*". This is further specified in the Swedish radiation protection law SSI FS 1988:220 which has the following corresponding wording (1§): "*The purpose of this Act is to protect people, animals and the environment against the harmful effects of radiation*".

The strategy and legislation is similar in all of the Nordic countries.

Planning for the financing - including the establishment of reliable cost estimates – is a part of this strategy, c f section 1.5. Cost calculations can, however, not be performed as an isolated or incidental event. They must be part of an integrated strategy and planning involving all relevant aspects over the life cycle of a plant. Cost calculations are required in all the Nordic countries in all stages of planning, c f Section 1.5. Therefore, sufficient strategic decisions and technical planning must exist at all times.

For practical purposes this implies that the mainly technical staff that in practice performs the planning for decommissioning must set their objectives based on non-technical – economical - needs and criteria. It is essential in this regard that clear functional requirements are set as to the tolerable levels of uncertainties in the cost calculations and that their implications are fully communicated, realized and considered.

Ideally, decommissioning should start already at the design phase of a plant and be part of the overall long-term planning and management. By including decommissioning aspects from the beginning, the actual cleaning and dismantling operations can be carried out very efficiently and with insignificant impact on health, environment and natural resources.

Conversely, if no provisions and preparations for decommissioning were made in the design and construction phase of a facility, it is imperative that planning is being commenced "*as soon as possible*"[1], and that it also includes "*the costs of the decommissioning and the means of financing it*"[2]. In such a case, the extent of efforts required might be rather fortuitous, depending on e g what design features were actually chosen, and what foresight has been applied during the operation. This applies also to the possibility to assess the extent of efforts required.

Nonetheless, the increasing realisation of these prerequisites in the international nuclear communities has lead to the establishment of certain procedures and development of tools to manage the situation. In this regard, the IAEA has compiled the vast international experience into a number of Safety Guides [1-4] dealing primarily with management, safety and technical matters. National guidelines include [12, 40]. Strategy and costs are discussed in e g reports from IAEA[5] and OECD/NEA[6,41], but no international guideline on how to achieve requirements on cost calculations has been identified in the present work[10].

Sections 3.2 – 3.4 summarizes good practice needed as a basis for cost calculations. The sources for the account include the above references as well as experience from the organisations of the present authors. The proposed practice is based on a requirement on precision in the cost calculations of ± 20 %. The word "precision" has the meaning that there should be a 65 % probability that a cost estimate would fall within ± 20 % of the actual cost as incurred after the project has been completed. This figure was put forward in [8, see also 10] as being achievable for decommissioning of nuclear research facilities. This requirement is in reasonable concordance with the figure of ± 15 % mentioned in [41], the ± 20 for 60 % probability in [42] and the ± 20 % in [43] for nuclear power reactors.

It was mentioned in both of these cases[8,41] that such a level of precision can be achieved for a decommissioning project only if the approach is rather ambitious. This includes the actual calculations as well as the basis for them. Thus, following the international standards [1-4] e t c is highly recommended but will not be sufficient in general. The good practice described in the following is intended to fill in this gap, at least partially.

It should be pointed out that the precision of ± 20 % might not be attainable – or rather reasonable to aim at achieving – for some systems. However, the requirements of accuracy in the cost calculations in general still apply. Consequently, deviations should be accepted only when justified, when the reasons for them are properly accounted for, and when an estimate or at least a verbal description of the level and nature of the uncertainty is documented. Such information will constitute part of the basis for assessment of pertinent levels of fees as well as for transparency around the finance system.

A prerequisite for the high precision is that management and staffing is adequate, see e g [44]. It might be indicated, though, that proper management is imperative, and that staffing should preferably include people having experience in operation of the plant in question as well as in previous decommissioning projects. Since these experiences mainly rest with different individuals it is an important management task to promote the appropriate integration between the two.[44]

[10] Quote T. S. LaGuardia in [41]: *"An international organization such as the International Atomic Energy Agency (IAEA) or OECD/NEA, or both need to re-establish a committee to promote the standardization of cost estimation guidelines and methodology. The committee should seek adoption of cost estimating guidelines and methodology, and provide training as required for implementation of its use. Similarly, the committee should be directed to continue to accumulate actual decommissioning costs and convert them into a form that does not compromise proprietary information. From this data base, consensus can be achieved."*

3.2 Methodology selection

It might be tempting to make the selection of technology straight from knowledge of the equipment and building construction in combination with experience from conventional cleaning and disassembly operations. In such a case, it will most likely be realized sooner or later in the project that other techniques will have to be or should have been applied due to the implications of the radiological contamination.

At first sight, the statement just made might appear as self-evident or even commonplace. However, it is frequently difficult even for experienced people and specialists in the area to fully apprehend its implications. For instance, a certain technique might appear appropriate, considering the amount of efforts estimated initially. However, it might become apparent through the course of the work that this estimate is in error, and thus another method would be preferable. In such a case, it may be imperative that an alternative and supplementary technology it available, at the time when it is needed, and that those responsible are prepared to reconsider their selection of technology on a continual basis.

Actually, no rational selection of technology for decommissioning of a nuclear facility can be made without a sufficiently comprehensive radiological survey (cf Section 3.3). Even when such a survey exists, it may not be sufficient for all of the needs. For instance, some of the activity may not be possible or feasible to measure before certain sources or bulky components and/or structures have been removed. Such cases call for contingencies in terms of alternative plans and methodologies.

Actually the graphite in the R1 reactor (cf Section 5.3.1) is an excellent example of this. The radiological survey preceding the decommissioning included sampling and measurement of the graphite neutron reflector around the core. However, it was not appropriate for radiological reasons to make the sampling and characterization comprehensive (and give rise to an increased dose to the staff), and thus some uncertainty remained. It turned out that the rest of the graphite was more radioactive than the sample taken, and consequently the work had do be carried out somewhat differently and therefore took some more time. (The over-all outcome was very good, however, see below).

It is sometimes thought that decommissioning of a nuclear facility requires the availability and use of novel techniques that have to be developed in conjunction with a project. Indeed, it is a good idea to carry out research and development work on decommissioning in order to come up with safer and more efficient methods and also to improve the planning and operation as well as the cost calculations. However, the general experience is that the technologies for decontamination, dismantling, demolition, size reduction and assaying and packaging need not be nearly as sophisticated as those used for the construction of the plant.[44] It is important to use proven technology which will provide for reliable planning and costing rather than theoretical approaches with advanced technology and potential – but not necessarily actual - cost reductions.

Further support for such an approach can be found in [45] where an evaluation is made of the availability of technologies and where it is concluded that most of the techniques required are widely available at present. Rather, it is the interfacing between techniques in combination with the radiological prerequisites that constitute the challenge.[45]

However, availability on the world market in general does not necessarily mean that a technology is readily available for use in decommissioning at a nuclear research facility. The deregulated markets enable companies to invest in development of techniques to be used in commercial decommissioning operations. This gives rise to a selection of vendors and techniques as well as competitive prices.

The other side of the coin is that each vendor will defend its information and may only participate in projects on its own conditions. This might not be suitable for small projects with research facilities where is not feasible to call in staff of a supplier from another part of the world to undertake minor tasks. Conversely, methods which have been used successfully in the past and which are familiar to the existing staff might not be the best choice in a new situation.

Thus, many considerations apply when methodologies and their interfaces are to be selected, and the analysis of the best choices might be complex. In order for a selection of technology to be systematic, transparent, integrated, and defensible in retrospect e t c, it is a good idea to use some kind of systematic approach. There are a number of books available on the principles of decision making and References [46-47] represent the analytic hierarchy process methodology. Application of such a systematic approach means that the selection process can be described, and thus be communicated to interested parties and stake holders. It also substantially reduces the risk of bias including the risk of others suspecting that bias is involved.

Much of the material needed for such evaluation and comparison can be found in the literature. This includes the methods themselves and their specifics as well as various projects that have been carried out. It is of special value if it is possible to find a plant that is similar so that the experience is particularly relevant.

An example of this can be found in [8-9] on an intermediate level waste storage facility at Studsvik where a similar but largely completed project was found at the Argonne National Laboratory in Illinois, USA. The experience with the drilling rig included difficulties with drilling with sufficiently high precision as well as loss of drilling liquid and potential contamination of the drill fluid due to voids in the concrete.

No plan or selection should be made without extensive contacts with people at other similar facilities. Nothing can replace such input. There are many lessons learned and much is published in the literature, but the benefit will be much larger if such studies are combined with plant visits and meeting the staff. There is an overrepresentation of success stories, and they have a high value as good examples, but it is equally important to learn from mistakes or difficulties, and such aspects may be easier to communicate on an informal basis.

When the R1 reactor at the Royal Institute of Technology in Stockholm was to be decommissioned by Studsvik in the early 1980's, three persons went literally around the globe and visited a large number of facilities. This caused a few eyebrows to be raised among the colleagues, including those of one of the present authors, but it can safely be said in retrospect that this was completely warranted. It is also in concordance with advice generally given in the literature.

It is assessed as likely by the present authors that much of the success in the R1 project (c f section 5.3.1) is due to the careful planning and the ability to find and make use of experience from other facilities.

3.3 Radiological surveying

It has been said already that the cost for decommissioning of a nuclear research facility with typical levels of contamination may be two or more orders of magnitude higher than for a corresponding (hypothetical) non-radioactive plant.

The presence of radioactivity gives rise to increased cost in a number of ways:

- The practical work will have to be conducted with the precautions necessary with regard to the radiological health hazard (remote handling, radiation monitoring, dust control, e t c)
- The sources containing most of the radioactivity will have to be removed and managed separately
- The general contamination will have to be reduced by decontamination
- The residual levels will have to be determined to be sufficiently low as to allow reasonable management of the waste

However, major radioactive sources might not be possible to remove until bulky components have been taken apart. In some cases novel and somewhat sophisticated techniques might be applied to at least allow the major sources to be characterized, e g to insert radiation probes into pipes.[45]

Radiological surveying for decommissioning work is very different from that of ordinary operation of a facility. The main reason for this is that the purpose is different. For the ordinary work, it is the general level of radiation together with the potential for contamination that constitutes the health hazard. For decommissioning, knowledge is needed also on concealed radionuclides that might not even show up on the readings of the instruments.

Examples of such concealed activity may be surface contamination that has become stabilized by means of paint. In such cases, smear tests will not unveil its presence. Other cases include deposits on the inner surfaces of pipes and other equipment, and deposits in fissures and fractures. A special case of concealed radiation sources is where components have become activated in their interior, which may be the case for items that have been exposed to radiation by neutrons.

The prospect of finding concealed activity is related to the ability of the radiation in question to penetrate. Here alpha and beta emitters have a short range, especially in condensed matter, and the penetration range of gamma rays is highly dependent on their energy (which is different for different radionuclides).

Also the potential health hazard varies highly between external exposure, respiratory intake and oral intake, which in turn are different for different radionuclides.

With time experience will develop as to what to look for, and efficient means of controlling the radiological hazard have been developed for facilities that are either large or many of a kind (or both). Thus, in light water reactors with little fuel damage[11], activation products from outside the fuel (but including the outer surfaces of the fuel pins) dominate the hazard, and among them cobalt-60. It has a half life of around five years which is sufficiently long for it not to decay in a short time, and yet sufficiently short in order for the unstable nuclei formed to transform to a stable state at a considerable rate. In addition, the energy of the gamma rays emitted is high, and so the radiation is quite penetrating.

Consequently, much of the time simple instruments measuring cobalt-60 can be used, and the hazard of other radionuclides can be evaluated by inference (e g transuranics).

For a nuclear research facility, such commonplace features might not necessarily apply. For instance, if alpha radiating specimens without accompanying gamma emitters have been handled, contamination might be very difficult to find since the alpha radiation is very easily shielded. Another example might be standard assumptions used in order to determine the amount of activity inside a pipe. If the calculation is based on cobalt-60 while the actual radiation is something else, then it is likely that the inventory is underestimated since the radiation from cobalt-60 is more penetrating than for most other sources.

Thus, a radiological survey of a nuclear research facility for the purpose of decommissioning should start with a recapitulation of what the facility was used for and an analysis of what might be expected in terms of radionuclides and contamination levels. The next step would be a general survey including hot spots, potential hidden activity and known sources.

The strategy, planning, methodology selection and uncertainty analyses are highly dependent on the results of the radiological survey. Most likely, such work based on a general survey will give rise to specific questions on the radiological situation. Thus an iterative approach should be applied and supplementary and specific surveys conducted.

Such iteration initiated work should include planning for the radiological follow-up of the decommissioning operation as well as the measurements intended for waste and for material to be released (unconditionally or otherwise).

[11] In the case of fuel damage cesium-137 and strontium-90 will be of interest as well. Cesium-137 is also a gamma emitter albeit the energy is lower and the penetrability less than those of cobalt-60.

In some cases, it might be difficult to measure sufficiently well in order to achieve the requirement of ± 20 %. One reason might be if it is difficult to avoid dose to staff. Such cases should be documented and the associated uncertainty assessed. In this way, the total cost commitment may still be estimated and possible to find limits for. Hopefully, various uncertainties may even out and make the total uncertainty acceptable nonetheless.

In other cases, it might be warranted to carry out a limited amount of work prior to the actual decommissioning in order to be able to obtain good radiological data for the various other planning activities and for the cost calculations. Such work may include removal of sources and hot spots, emptying containers (e g with ion exchange resin) cleaning, e t c.

Sampling for the purpose of radiological characterization is a natural part of the radiological surveying and should be conducted to the extent needed and appropriate. Sampling may also include a certain but limited amount of decommissioning work, e g core drilling in concrete, or (perhaps temporary) removal of shielding or other entities in order to take samples.

Since the radiological work serves several purposes and concerns various groups of people and is carried out iteratively, it is important that there exists plans for this work and that they are properly updated. Similarly, it is important that the results are properly documented.

The basics of radiation and radiation protection are not explained in the above, and the reader is referred to the standard literature on the subject, see e g [48].

3.4 Uncertainty analysis

It has been pointed out in the previous sections (3.1-3.3) that the aim of ± 20 % in uncertainty might not be achieved for all systemss even if appropriate planning, methodology selection and radiological surveying is carried out. The knowledge needed for such a precision might not be reasonably achievable.

For such cases it is imperative that assessments are made regarding the possible size of the issue and the probability of various outcomes. As a minimum this should be carried out verbally with scenarios for various types of outcomes. It is also important that an upper bound of the magnitude of each case is stated.

Such uncertainty analyses can then be integrated in total assessments where the total uncertainty typically can be shown to be less than those of the constituents. Such conclusions can be made only if the various cases involved do not have common causes.

However, experience tells us that such analyses will only bring to attention part of the total uncertainties. If no further analyses are made it is likely that "surprises" will appear during the course of the work. Experience also tells us that such surprises are more likely than not to give rise to increases in cost.

Thus, some sort of extended uncertainty investigation and analysis need be made in which further features, processes and events which might cause increased cost can be identified.

Such risk identifications and assessments can be made using tools which are available from the area of technical risk analysis and which are described extensively in the literature, see e g [49-51]. Even when such an extended uncertainty analysis has been made, there may still be features which have not been identified and which constitute a residual risk. Such uncertainties can be managed by means including another factor for contingency.

An extended uncertainty analysis should start with a system description together with a definition of the boundaries for the analysis, which defines the border between internal and external features and events. If the parts of the work described in Sections 3.1-3.4 are well underway, much of what is needed has already been compiled. The two types of descriptions are not identical, however. For the extended uncertainty analysis it is beneficial to structure and analyse the systems in terms of the following[52-53]:

- the parts of the system in which or between which the different processes take place together with the relevant properties (*features*)
- initiating internal as well as external *events*
- the *processes* that occur during these events

After the system has been identified and described including its interdependencies, the next step should be to identify potential uncertainties, and especially all types of risks. Different sources should be consulted in order for the compilation to be as complete as possible. It is highly desirable that individuals with different kinds of competence and experience are involved in this work. A few examples of what might be attempted are given in the following:

- a systematic analyses of the various aspects of the facility
- brainstorming
- follow standard check lists
- review literature
- utilize feed-back from previous projects
- networking internationally

The assessment of the various types of uncertainties identified relates to the following questions:

- Where might there be deviations?
- How likely is it?
- What would be the consequences (including worst case)?

There are a number of methods available for risk / uncertainty analysis. They can be divided into inductive or deductive. For deductive methods assumptions are made on the final outcome and the task of the staff is to attempt do describe events that might lead to such a consequence. For inductive methods, some sort of error is assumed and the task is to foresee what consequences this might lead to. Methods that can be applied include the following[51]:

- Preliminary Hazard Analysis (PHA)
- What-if analysis
- Hazard operability analysis
- Failure modes and effects analysis
- Fault tree analysis
- Event tree analysis
- Cause-consequence analysis

It is important that the work is carried out in steps, and that checks are made from time to time to evaluate what level of effort is warranted. It is anticipated that for most purposes it will be sufficient with uncertainty identification together with expert judgement and assessment rather than a full analysis.

The result should be identifications of uncertainties together with assessments of their probabilities and consequences.

An example of an identification of a potential uncertainty was made in [9-10] where it was found that a pool for wet storage of spent fuel did not have the double containment that modern facilities do. Thus, conceivable leakage to the underlying rock and soil constitutes an uncertainty with regard to cost. The uncertainty was identified from systematic searches and studies in the literature of facilities. The probability and consequence were not evaluated, although it was assessed that the most probable case is an intact containment. In the case appearing in the literature, leakage had occurred and contamination had spread outside the facility, however.

It is important that the uncertainty analysis is properly documented. This will enable future analyses to start from where the previous ones ended. It will also make the process for financing transparent and thereby also credible to stake holders and interested parties outside the sphere of experts.

4 Techniques for assessment of cost

4.1 Cost structuring

Decommissioning is the final phase of the life cycle of a nuclear facility and is thus highly dependent on the design, operation, documentation and planning, etc. Nonetheless, it has been shown in a number of projects [54] on various types of facilities that technical methods and equipment are available today to dismantle safely nuclear facilities of whatever type and size.

Decommissioning projects for various types have also demonstrated that costs can be managed. However, comparisons of cost estimates for different individual facilities may show relatively large variations[54], even at late stages of planning, and both in relation to cost calculations for other facilities and to incurred costs.

In the past, cost estimates have been based on the world-wide experience from decommissioning projects as well as maintenance and repair work at facilities in operation. This experience has been compiled and utilized in the form of either costs for various tasks and / or unit costs for various basic decontamination and dismantling activities.[54]

A number of differences exist between the various facilities and projects constituting the original base for such per item data. Moreover, the prerequisites for extracting such per item data vary considerably since the method of calculation and the structuring of the cost items may also be very different.

Such errors may be strongly reduced if a common "standard" is applied on the structuring of the costs as well as on the schemes for calculation. This topic has been dealt with by OECD/NEA in collaboration with IAEA and EU and the resulting "*proposed standardised list of items for costing purposes in the decommissioning of nuclear installations*" has been documented in [54]

The group undertaking this work found that it is essential when cost figures from a project are to be used that the real content, i e what is actually behind the figures, be investigated and analysed. Numbers taken at their numerical value, without regard to the specific context, can namely easily be misunderstood and misinterpreted.

Consequently, the group has also come up with a compilation of definitions of the technical cost groups, cost elements, and cost factors.

The document [54] consists mainly of listings of the various cost items. It is very detailed and extends over more than a hundred pages. Obviously, this structuring corresponds to the summation method

4.2 Cost estimation methodology

The OECD/NEA document [54] (cf section 4.1) does not say anything about how it should be applied with regard to the stage of planning. It is obvious from the document, however, that an underlying assumption is that estimates can be made on an item to item basis. This actually presupposes that a relatively detailed planning has been carried out (cf *appropriate planning* in Section 3.1) including methodology selection (cf Section 3.2), radiological surveying (cf Section 3.3) and uncertainty analysis (cf Section 3.4).

4.2.1 Cost calculations for new industrial plants in general

The topic of cost calculations in early versus late stages of planning has been dealt with in the literature on cost calculations for industrial plants in general [55]. Actually, early cost calculations may call for approaches that differ from those of late ones. State of the art in this area might be briefly summarized as follows.

As soon as the final process-design stage is completed, it becomes possible to make accurate cost estimations because detailed equipment specifications and definite information are available. However, no design project should proceed to the final stage before costs are considered. In fact, cost estimates should be made throughout the various stages of planning, development and design in spite of the fact that complete specifications are not available.

Thus, cost estimates can be made even at the earlier stages and are then referred to as predesign cost estimations. If the design engineer is well acquainted with the various estimation methods and their accuracy, it is possible to make remarkably close cost estimations even before any detailed specifications are given. Such cost estimates frequently form the basis for the management in their decisions on investments.

Five categories of cost estimates have been identified to be applied to the successive stages in a large chemical plant project[55]. These are as follows:

1. Order of magnitude (ratio estimate) based on similar previous cost data; probable accuracy of estimate over +/- 30 percent.
2. Study estimate (factored estimate) based on knowledge of major items of equipment; probable accuracy of estimate up to +/- 30 percent.
3. Preliminary estimate (budget authorization estimate; scope estimate) based on sufficient data to permit the estimate to be budgeted; probable accuracy of estimate within +/- 20 percent.
4. Definitive estimate (project control estimate) based on almost complete data but before completion of drawings and specifications; probable accuracy of estimate within +/- 10 percent.
5. Detailed estimate (contractor's estimate) based on complete engineering drawings, specifications, and site surveys; probable accuracy of estimates within +/- 5 percent.

Predesign estimates are based mostly on historical data from similar facilities together with utilisation of adjustment factors for cost increase with time, size of the facility and/or composition of the intended equipment. Late estimates are instead largely based on detailed specifications and summations of all the items which contribute to the total cost.

It is important to realise the uncertainties associated with the various stages and possibilities for estimation. Some of them are arbitrary in character as the ones given in the listing above. Others are systematic in character and thereby perhaps more treacherous.

Pitfalls in this context include the following:

- Conceptual error. Performing the "correct" calculation for the wrong process, or for an incomplete one.
- Methodological error. Applying the summation method at too early a stage when only a fraction of all items to be included can be identified.

In the vast majority of cases such systematic errors lead to underestimation of the actual cost.

4.2.2 Early stage cost calculations for decommissioning of nuclear research facilities

In practice, the summation method is frequently being applied at early stages in spite of its inherent tendency to give rise to underestimations of the costs. One important reason for this is that more suitable calculation techniques have not been developed or at least are not generally available.

It is therefore highly desirable to somehow "calibrate" results of early estimates against known costs of already completed projects of similar kind.

An example of such an approach is presented in [8-10], see also Appendix F, and the main features are as follows.

Let the cost for a plant be given by the equation:

$$K^c = \sum_i p_i \qquad (1)$$

Where
K^c = the total calculated cost
p = cost item, and
i = index for cost item

A fit to actual cost K^a for a completed project can be made using the weighing factors wi and a scaling factor s according to the following equation:

$$K^a - K^c = s \sum_i w_i p_i \qquad (2)$$

The weighing factors may be obtained by assessment of which items should have a small, intermediate, large or very large influence on the difference between calculated and actual values. For instance, a weighing factor can be given one of the values 1, 2, 4 or 8. The scaling factor can then be calculated using the equation:

$$s = (K^a - K^c) / \sum_i w_i p_i \qquad (3)$$

For a plant for which a refined cost calculation is to be made, the cost items can be calculated first, and then the total cost according to the equation (1) above.
After that, an adjusted calculated total cost can be calculated using the equation:

$$K^{adjusted} = \sum_i (1 + sw_i) p_i \qquad (4)$$

where s and wi have been derived from a similar reference plant and p_i for the plant for which a refined calculation is to be made.

The application of equation (4) implies an improvement compared to a simple over all scaling since differences in the assessed cost structure influences the result.

The example illustrates how some of the systematic errors might be avoided, or at least turned into errors that are random in character. For projects having a fair size random errors frequently even out. Systematic errors add up, however, and give rise to a total error (figured as percentage) which is just as large as the small ones.

It should be noted that the above approach is just an example and that many schemes might be worked out to the same end. Ideas in this regard might be found e g in Reference [55].

4.2.3 Available methodologies for cost calculations for nuclear research facilities

State of the art on industrial cost calculations can be found in many sources, e g [55-61]. Many of these are general, but some apply to specific types of projects and there is also some literature on cost calculations for environmental remediation and decommissioning of nuclear facilities, see e g [56-57]. Also, there exists data bases for such calculations, at least one of which can be downloaded from the Internet at the price of a thick book.[12] It includes "difficulty factors" which are termed "safety levels" where level "E" has 100 % efficiency and level "A" has 37 % efficiency. Clearly the data base cannot be intended for other than lightly radiologically contaminated sites and facilities. Nonetheless, it illustrates state of the art.

Traditionally, in cost calculations in industry in general, the comparison method is applied at early stages when detailed data are not available. When the detailed design is available for a new facility, then the precise cost estimate can be made. It is crucial for a

[12] *Environmental Remediation Cost Data – Unit Price*, 12th Annual Edition, sold by Azimuth Group, Ltd. (www.echos-online.com).

bidder to know as precisely as possible what it would cost to build the facility in question. If the bid is too high, the contract will go to someone else. If the bid is to low, then it is even worse, since the bidder will get the contract and probably loose money. Thus, before a bid is placed, the bidder will have to know the cost together with the uncertainty of the estimate.

There are a number of prerequisites that must be fulfilled in order for a bid to be precise and the level of precision known, e g:

- A classification scheme for the costs such that each item means the same thing at all occasions.
- A data base with per item cost for each of the items ("unit price")
- A detailed design that can be "converted" by the cost engineer to items and volumes
- Historical data on errors in estimates that can be utilized to assess the error in the current estimate

When managed well, such a "unit price estimating method" might have a precision of ± 5 %. This presupposes that considerable efforts have been spent on actually achieving the basis prerequisites listed above. This is a sizeable task that requires access to many projects.

Big actors may have what is required in house but small ones will have to get together or consult someone who possesses the prerequisites required.

It has been tempting sometimes to apply this method in a straightforward and uncritical manner to the exact volumes of items that can be extracted from drawings and "as built" in nuclear facilities. Formally, the level of detail may even exceed that of a detailed design. However, since the quality factor has such a decisive influence, the precision of such an approach is illusive.

Frequently, owners of nuclear facilities turn to consulting companies specializing in cost estimations. However, word of mouth conveys to the present authors that such estimates on the same facility carried out by different consulting firms may deviate by as much as a factor of two.

Obviously, the knowledge base for cost calculations will have to include radiological conditions and prerequisites in a relatively detailed manner in order for a high precision to be achievable.

If the summation method with unit cost data is to be used, "difficulty factors" will have to be applied with high relevance and precision. Similarly, in the comparison method, the similarities and differences will have to be dealt with in detail. Even so, only a moderate level of precision might be expected until sufficient experience is accumulated. Furthermore, experience is also necessary in order for the error in the estimate to be assessed.

5 Reactor DR1 at Risø National Laboratory in Denmark

5.1 General approach

5.1.1 Prerequisites and method used for cost assessment

The material below is mainly taken from [24] as well as a document with the title "*Decommissioning in Denmark*" (cf Appendix A, see also Appendix B [62]); it is also presently available at the website of Danish Decommissioning (http://www.ddcom.dk).

Risø National Laboratory (RNL) was established in the late 1950'es as a Danish research centre for preparing the introduction of nuclear energy in Denmark. Three research reactors and a number of supporting laboratories were built. However, Denmark has not yet built any nuclear power plants, and in 1985 the Danish Parliament decided that nuclear power should no longer be an option in the national energy planning. The facilities at RNL are thus the only nuclear facilities in Denmark. Subsequent to the Parliament's decision the research at RNL related to nuclear power was reduced and the utilisation of the facilities concentrated on other applications, such as basic materials research, isotope production and silicon transmutation doping. Already in 1975 one of the reactors had been taken out of service for economical reasons and the activities moved to the 10 MW materials test reactor, DR 3. Furthermore, in 1989 the hot cell facility was closed, and over the next four years it was partly decommissioned.

As part of Risø's strategic planning in 2000 it was taken into account that the largest research reactor, DR 3, was approaching the end of its useful life, and that the decommissioning question was becoming relevant. Since most of the other nuclear activities at Risø depended on DR 3 being in operation, it was decided to decommission all nuclear facilities at Risø National Laboratory once the reactor had been closed. Therefore, a project was started with the aim to produce a survey of the technical and economical aspects of the decommissioning of the nuclear facilities. The survey should cover the entire process from termination of operation to the establishment of a "green field"[13], giving an assessment of the manpower and economical resources necessary and an estimate of the amounts of radioactive waste that must be disposed of.

After thorough preparations, including an Environmental Impact Assessment, the Danish parliament in March 2003 gave its approval to funding the decommissioning of all nuclear facilities at Risø National Laboratory to "green field" within a period of time up to 20 years. The decommissioning is to be carried out by a new organisation, Danish Decommissioning (DD), which is independent of Risø National Laboratory, thus avoiding any competition for funding between the decommissioning and the continued research activities at Risø.

[13] In this context "green field" means a situation where facilities and areas are free released to other use without any radiological restrictions. Thus clean buildings and equipment may be re-used for other purposes than nuclear.

As the facilities are (and were) 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[54] (cf Section 4.2.1) was used as a template for specifying the costs of decommissioning operations. 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 it was felt that they had sufficient insight, or with the help of consultants or the PRICE programme, described below. A standard rate of 231 DKK/hour ([24] was published in the year 2001) 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, it was 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.

The approach taken by Danish Decommissioning is to find a sufficient knowledge base so that the summation method (cf Section 4.1) could be applied and justified. This was achieved through a combination of compilation of existing data together with supplementary investigations along the lines described in Section 3. The underlying descriptions together with the actual assessments are documented in [24].

5.1.2 The computations using the computer program 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:

- 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 in Figure 5-1.

5.1.3 Limitations

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 in [24] 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.

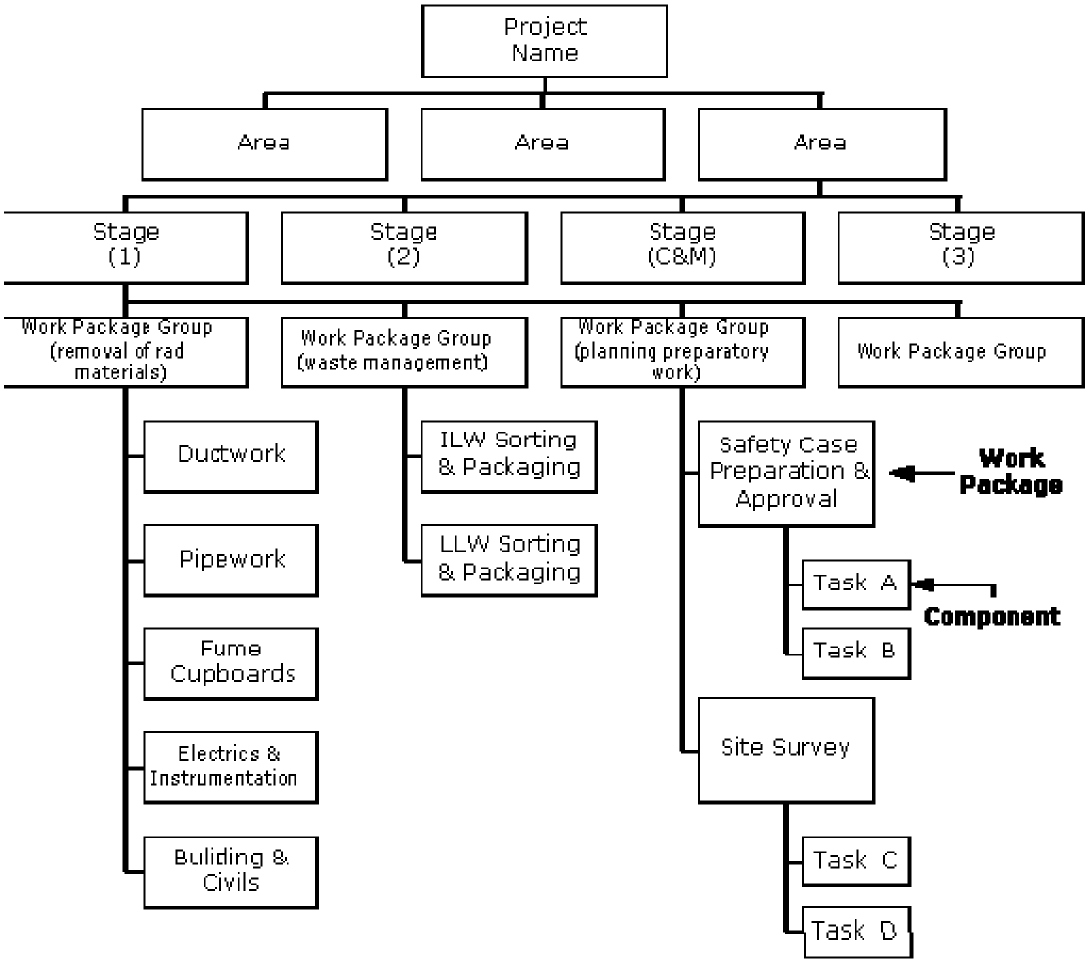


Figure 5-1. The hierarchical structure or Work Breakdown Structure used by the cost calculation programme PRICE.

In awareness of these limitations, Dansk Dekommissionering analyses and assesses the total decommissionning project of the nuclear facilities at the Risø National Laboratory on an iterative basis by means of the "Successive approaching calculations –principle" supported by the Programme "Futura Nova" and facilitated by Lichtenberg&Partners consultants. This principle gives a good possibility to identify and rank the decisive factors of uncertainty.

5.2 Estimated and actual costs for the decommissioning of Reactor DR1

The DR1 research reactor was stopped permanently in year 2001, and it was decided to start immediate planning of the decommissioning.

The reactor was a small "University reactor" with a thermal power of 2 kW, used mainly for basic reactor physics experiments and for educational purposes. It is briefly described in Sections 5.2.1 – 5.2.3.

As a part of the description of the project for the decommissioning of the research reactor DR1 an estimate of the total costs were carried through (cf Section 5.1 and [24]).

The total project was broken down in sub- projects, and the project group discussed the necessary man hours and expenditures related to the sub projects, based on the group members' experience from related work operations in the past. The hourly cost rate was calculated as a weighted rate, taking into account the composition of the necessary work force i.e. technician or engineer, and furthermore took into account an estimate of the distribution between external- and internal workforce hours, as described in the calculation scheme in Table 5-1.

At present (i e December 2005) the main part of the DR1 decommissioning project has been finished - the final radiation survey still remains. (A general description of this decommissioning project can be found in [63].)

Therefore a summing up of the actual costs has been performed as shown in Table 5-2.

Some of the tasks in the actual project were carried out in an order different to that shown in the original plan, and the activities were similarly accounted otherwise. This has been marked with notes a, b, c, etc. in Table 5-2.

Some estimated costs (plastic tent around the biological shield during demolition, demolition of the reactor building, several pieces of health physics radiation measuring equipment, waste registration system) have been omitted or accounted for outside the DR1 project and consequently have been removed from the original cost estimate as they were shown in Table 5-1.

It should be noted that as well the estimated costs, as the actual costs are without overhead.

If the external costs about 2.5 million Dkr are subtracted, the total costs of the project are about 2.9 million Dkr, which primarily comprises internal wages and costs for concrete containers.

If overhead of 112% is added to this amount we get internal costs of 6.1 million Dkr which added to the external costs of 2.5 million Dkr brings the total project costs to 8.6 million Dkr.
As can be seen, the total actual costs at present only sums up to about 5.4 million Dkr, compared to an estimated total cost of 7.3 million Dkr. For the still unfinished tasks the estimated costs have been used in the total summation.

The difference between estimated- and anticipated actual costs thus is about 1.9 million Dkr or 26% lower than the estimated total project costs. A deviation of 26% is within the usual interval of plus and minus 25%-30%, which normally is considered to be the uncertainty of an initial cost estimate of a decommissioning project.

Table 5-1. Costs for decommissioning of DR1 estimated before the start of the project.

1 working week = 5 working days	F1= DKK 247 Ext. Tecnician	F2= DKK 216	F= DKK 224	F = $(1/4\ F_1 + 3/4\ F_2)$
1 working day = 7,4 hours	E1= DKK 380 Ext. Engineer	E2= DKK 322	E= DKK 337	E = $(1/4\ E_1 + 3/4\ E_2)$

Activity	F/E	Number of persons	Working time in man-days	Hourly salary	Calendar time		Manpower expenses	Acquisitions + External assistance	Total cost	Accumulated cost	Remarks
			Days	DKK	Weeks	Days	DKK	DKK	DKK	DKK	
Flushing of fluid level meter (has been completed)	E	1	10	337	2	10	24 901		24 901	24 901	
Plan for removal of fuel solution in DR 1 (J.nr.: RD-2001-412-1-Dok. 3, Rev. C)											
Planning	F	1	70	224	12	60	115 903	120 000	609 418	634 319	4 Containers, 4 lead flasks and 4 carts
	E	1	150	337			373 515				
Removal of fuel solution (has been completed)	F	1	20	224	8	40	33 115		132 719	767 038	
	E	1	40	337			99 604				
Flushing of primary system (has been complete	F	1	20	224	8	40	33 115		82 917	849 955	
	E	1	20	337			49 802				
Determination of Sr-90 content in core solution								20 000	20 000	869 955	NUK
Clearing and removal of sources etc.	F		130	224	6	30	215 248		464 258	1 334 212	
	E		100	337			249 010				
Removal of recombiner	F	3	30	224	3	15	49 673	10 000	139 376	1 473 588	NUK incl. Nonbøl (10000) +
	E	3	30	337			74 703	5 000			Misc. Accessories (5000)
Removal of control- and safety rods	F	3	30	224	3	15	49 673	50 000	149 475	1 623 062	NUK incl. Nonbøl (20000) +
	E	3	20	337			49 802				Flask (30000)
Removal of reflector and core	F	4	160	224	8	40	264 920	45 000	524 128	2 147 190	Graphite analysis by NUK
	E	3	80	337			199 208	15 000			Special tools
Removal of remaining parts of the primary system	F	3	60	224	4	20	99 345		174 048	2 321 238	
	E	2	30	337			74 703				
Removal of cooling system	F	3	30	224	2	10	49 673		99 475	2 420 713	
	E	2	20	337			49 802				
Cleaning of the reactor- and recombiner caves	F	4	80	224	3	15	132 460	20 000	177 361	2 598 074	Cleaning agents + vacuum cleaner (20000)
	E	2	10	337			24 901				
Detailed characterisation of activity in shielding and reflector tank	F	2	50	224	12	60	82 788	20 000	292 194	2 890 267	Bore samples
	E	2	60	337			149 406	40 000			NUK
Detailed planning of demolition of shielding	F	1	15	224	6	30	24 836		99 539	2 969 806	
	E	1	30	337			74 703				

Table 5-1. Costs for decommissioning of DR1 estimated before the start of the project, continued.

Spot tests in reactor building	F	2	60	224	12	60	99 345		161 598	3 131 404	
	E	1	25	337	5	25	62 253				
Concrete containers, 3 pcs								120 000	120 000	3 251 404	
Demolition of shielding	F	3	120	224	8	40	198 690	500 000	998 294	4 249 698	Plastic tent (500000)
	E	1	40	337			99 604	200 000			Demolition and removal of reactor block
Cleaning and control measurements prior to breaking up the floor	F	3	60	224	3	15	99 345	250 000	399 147	4 648 845	Radiation monitor (mobile for floor) (25000
	E	1	20	337	2	10	49 802				
Disconnection of supplies	F	3	30	224	2	10	49 673	50 000	124 574	4 773 418	Transformer, electricity, water, sewer (5000
	E	1	10	337			24 901				
Release measurements of buildings	F	2	30	224	12	60	49 673		111 925	4 885 343	
	E	1	25	337			62 253				
Release measurements of reactor block	F	2	20	224	2	10	33 115		58 016	4 943 359	
	E	1	10	337	2	10	24 901				
Demolition of buildings	F	1	5	224	1	5	8 279	500 000	508 279	5 451 638	Contractor
Survey of areas	F	2	60	224	8	40	99 345	100 000	249 147	5 700 785	Bore samples
	E	1	20	337			49 802				Analysis
Measurement equipment for AHF (Applied Health Physics)								260 000	260 000	5 960 785	2 contamination detectors (100000), hand- and clothes monitor (1600000)
AHF education (for release measurements, 3 weeks)	F	11	88	224	3	15	145 706		220 409	6 181 194	AHF internally
	E	4	30	337			74 703				
Bathing- and changing facilities								300 000	300 000	6 481 194	Container with shower- and changing facilit
Tagging and registration of materials	F	1	100	224	20	100	165 575	300 000	963 595	7 444 789	Registration system (300000)
	E	1	200	337	40	200	498 020				
Transportation	F	1	100	224	45	225	165 575		165 575	7 610 364	Internal transportation
Planning	F	3	175	224	45	225	289 756		1 597 059	9 207 423	
	E	9	525	337	45	225	1 307 303				

Table 5-2. Costs for decommissioning of DR1 summarised after the completion of the project.

The first column shows the corrected estimated costs for the actual sub tasks in the project (Cost est.). The second column shows the accumulated estimated costs (Sum est.). The third column shows the actual costs (Cost act.), and the fourth column shows the accumulated actual costs (Sum act.).

Decommissionning of Reactor DR1 **07.11.05 KI**

Activity	**Cost est.**	**Sum est.**		**Cost act.**	**Sum act.**
Planning and preparing	1597059			1537591	1537591
Flushing of fluid level meter	24901	1621960	a		
Core sol. Flasks and planning	609418	2231378	a		
Remov. of core solution	132719	2364097	a		
Detremination of Sr-90 in core	20000	2384097	a		
Clearing and removal of sources	464258	2848355	a		1537591
Removal of recombiner	139376			100913	1638504
Removal of control and saf. Rods	149475		c		
Removal of reflector and core	524128			321617	1960121
Removal of remaining prim.syst.	174048		d		
Removal of cooling syst.	99475		d		
Cleaning of reator- and recomb. Caves	177361			59257	2019378
Detailed caracterization of react. block	292194			0	2019378
Planning of demolition of shield	99539		e		
Contamination spot meassurements	161598		e		
Demolition of shield	998294			2025379	4044757
Cleaning and contam. survey of floor	399147		b		
Disconnection of supplies	124574		d		
Clearance meassurements of buil ilding ongo	111925			111925	4156682
Clearance meassurements of shield	58016		b		
Clearance of site ongoing	249147			249147	4405829
Health phys. Educ. Fo clearance meassm.	220409			489356	4895185
Active bath and change facilities	300000			38685	4933870
Concrete containers 3 pcs. ongoing	120000			450000	5383870
Transport of materials	165575		e		
TOTAL		**7292636**			**5383870**

Notes:
a: Actual costs included in "Planning and preparing".
b: Included in Health phys.Educ. Fclearance meass
c: Included in "Removal of reflector and core
d: Included in "Reactor- and recomb. Caves"
e: Included in "Demolition of shield"
ongoing: means the activity is not yet finished, the estimated cost has been used as the actual cost,
cost, although the concrete containers has been raised in price due to preliminary bids

5.2.1 Description of the facility and surroundings

DR 1 (Danish Reactor No. 1) was a thermal homogeneous research reactor with an output of 2000 watts. The reactor was supplied by Atomics International in the USA and was commissioned in August 1957. The design of buildings and installations and the set-up of the facility were by Danish companies under the guidance of technicians from Atomics International. The location of DR 1 on the Risø site can be seen from Figure 5-2.

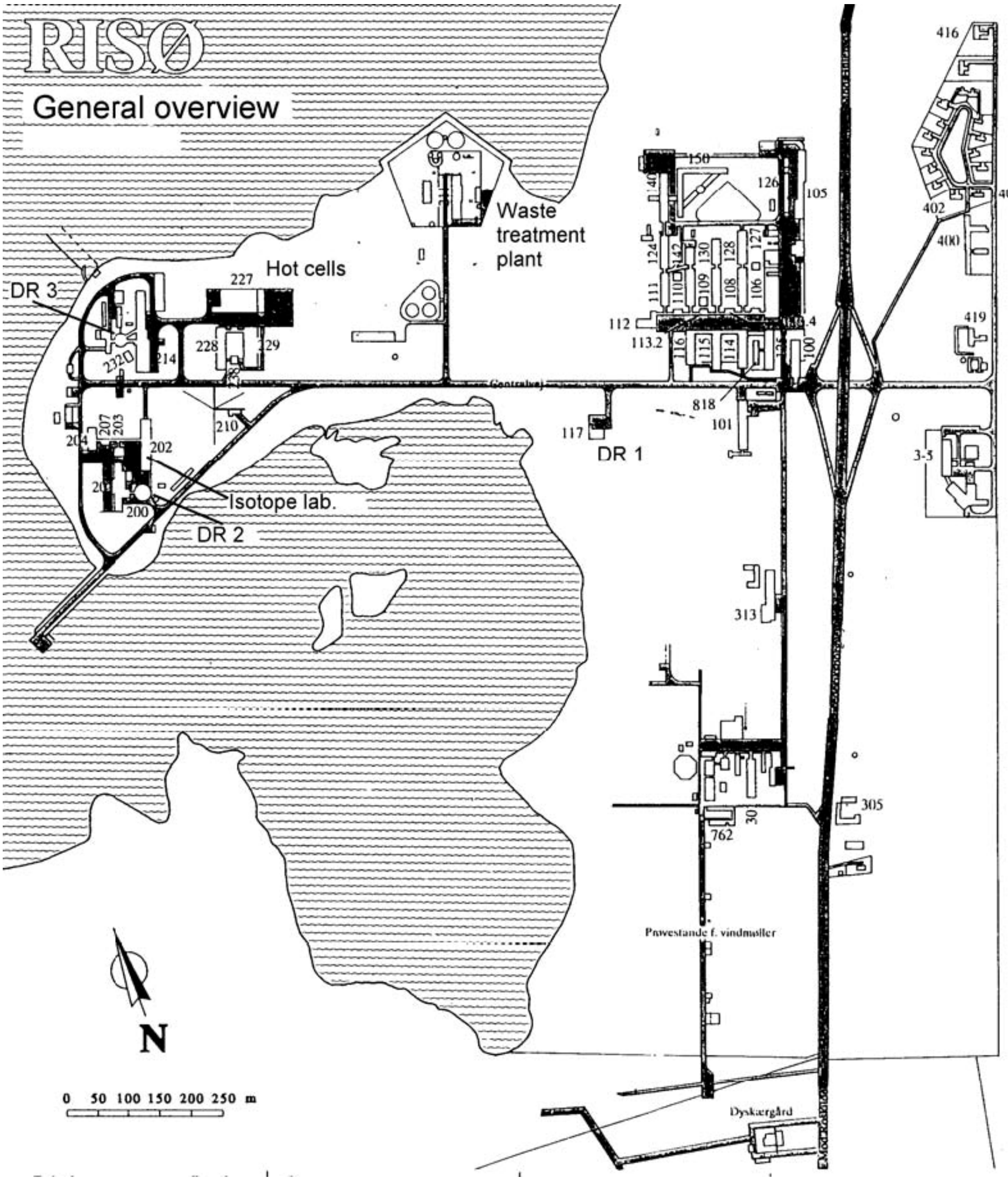


Figure 5-2 Map of Risø.

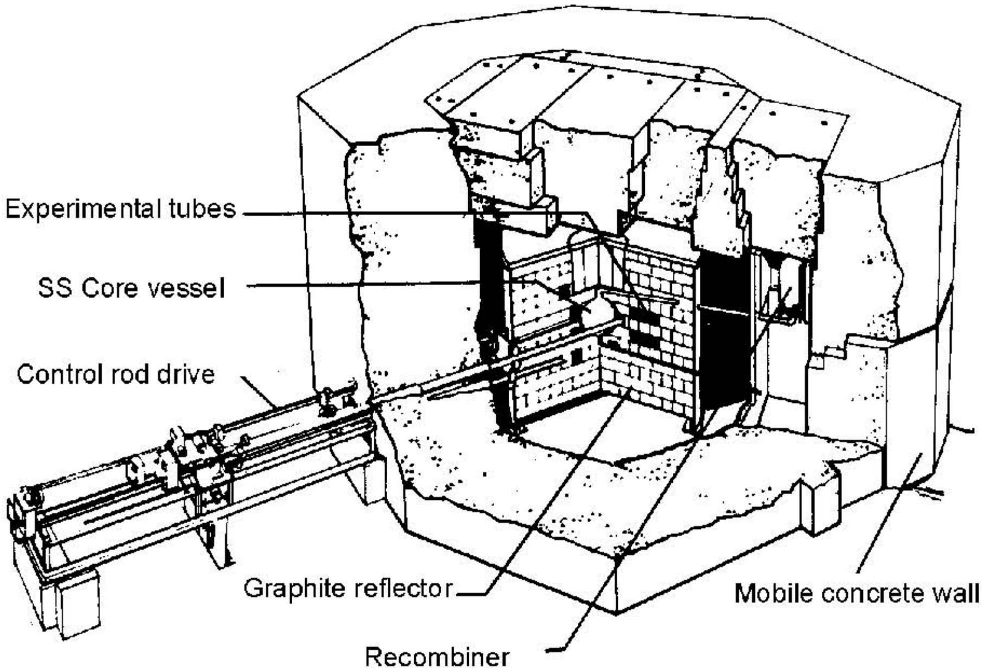


Figure 5-3. Sketch o/the structure o/DR 1.

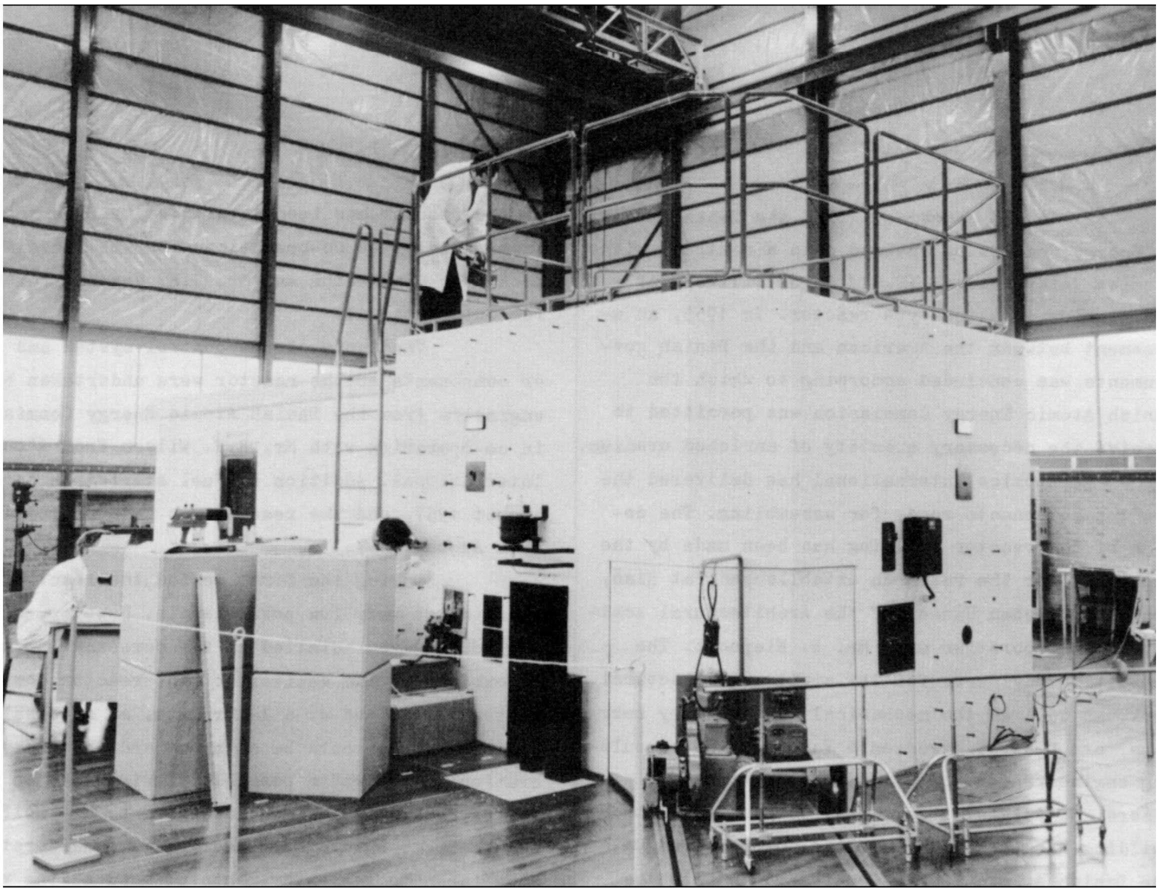


Figure 5-4. Reactor DR 1.

Originally, the reactor was built to generate an output of 5 watts. In the spring of 1959, the output was increased to 2000 watts following the installation of cooling systems and improvement of the shielding and the reactor has been subjected to a test run at 2.3 kW. At an output of 2 kW, the maximum thermal flux in the reactor is approximately 6×10^{10} n/(cm2 · sec). The reactor used 19.9 % enriched uranium as a fuel in the form of uranyl sulphate dissolved in light water.

5.2.2 Reactor build-up

The reactor consists of a ball-shaped stainless steel vessel (the core container) with a diameter of 32 cm (See Figure 5-3.). When the reactor was started, 984 grams of U-235 was added; the solution volume was 15.5 litres. The surplus reactivity of the reactor was less than 1.5 %.

Around the core container is a graphite reactor in a cylindrical steel tank with a diameter of 1.5 m and a height of 1.3 m.

On its sides, the reactor is shielded by a 1.2 m thick heavy concrete wall, while on top the shield consists of 85 cm thick concrete blocks (See Figure 5-3. and Figure 5-4.).

During operation, water from the core solution decomposes into oxygen and hydrogen. A pipe connects the core vessel to a recombiner outside the reflector tank, in which the oxygen and the hydrogen are recombined into water that runs back to the core container. Recombination is effected by means of a platinum catalyst heated to 70-100 ºC. Together, the core container, recombiner and connecting pipe form a closed system kept at a negative pressure (See Figure 5-5.).

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

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

The output of the reactor is governed by two control rods and two safety rods, moved horizontally in the reflector tank just outside the core vessel. The rods consist of a stainless steel jacket containing boron carbide. Each rod governs approx.1.5 % reactivity.

The essential reactor instruments are located in the control room. The most important instruments are the four independent neutron flux channels including a period meter, as well as instruments for recording the temperature of the core vessel and the catalyst in the recombiner, as well as the pressure in the care vessel/recombiner. Furthermore, values are given for the radiation level in the ventilation pipe from the reactor block and in the reactor hall, as well as the temperature of the cooling circuits, etc.

A pipe with a 2.54 cm diameter goes horizontally through the centre of the care vessel. With the reactor running at 2 kW, the max. thermal flux in the pipe is approximately 6×10^{10} n/(cm2 · sec).

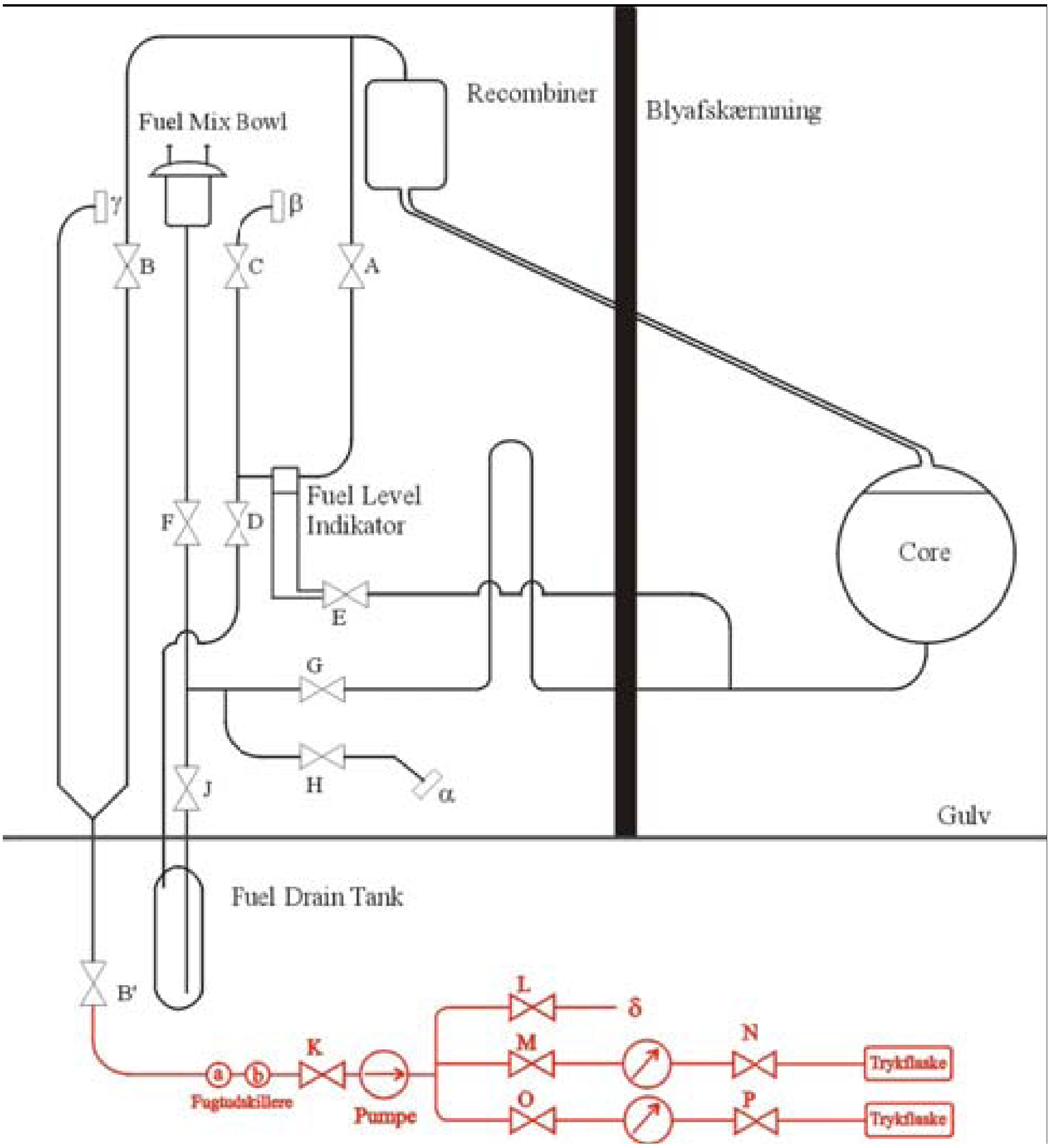


Figure 5-5. Block diagram of the primary core system.

5.2.3 Reactor hall

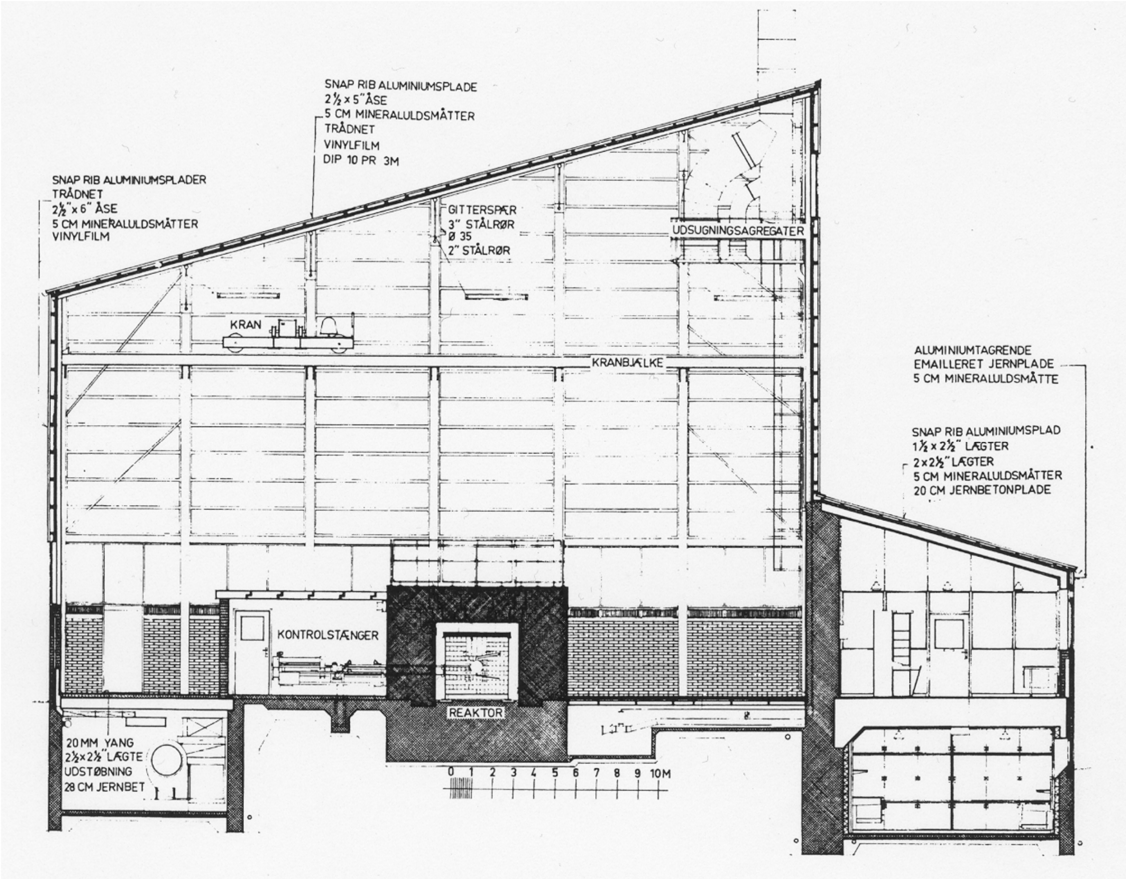


Figure 5-6 Vertical section of building 117.

The reactor building at DR 1 (see Figure 5-6.) consists of a reactor hall, a control room with an office and an entrance section, a counter room in the basement under the control room and an aggregate room for the air-conditioning system under the system end of the reactor hall (See Figure 5-7.).

The air-conditioning system blows warm air through the floor ducts along the facades and from here through ducts in the hollow parapets to injection grates underneath the windows. Under normal conditions, the ventilation was 9000 m^3/h, of which 6000 m^3/h was recirculated, which meant that fresh air intake corresponded to one exchange of air per hour (See Figure 5-6. and Figure 5-8.).

In 1960, the professional engineering journal "Ingenieren" published an interesting article about reactor DR 1 (and the two other reactors) which formed part of the background material for the planning of the decommissioning.

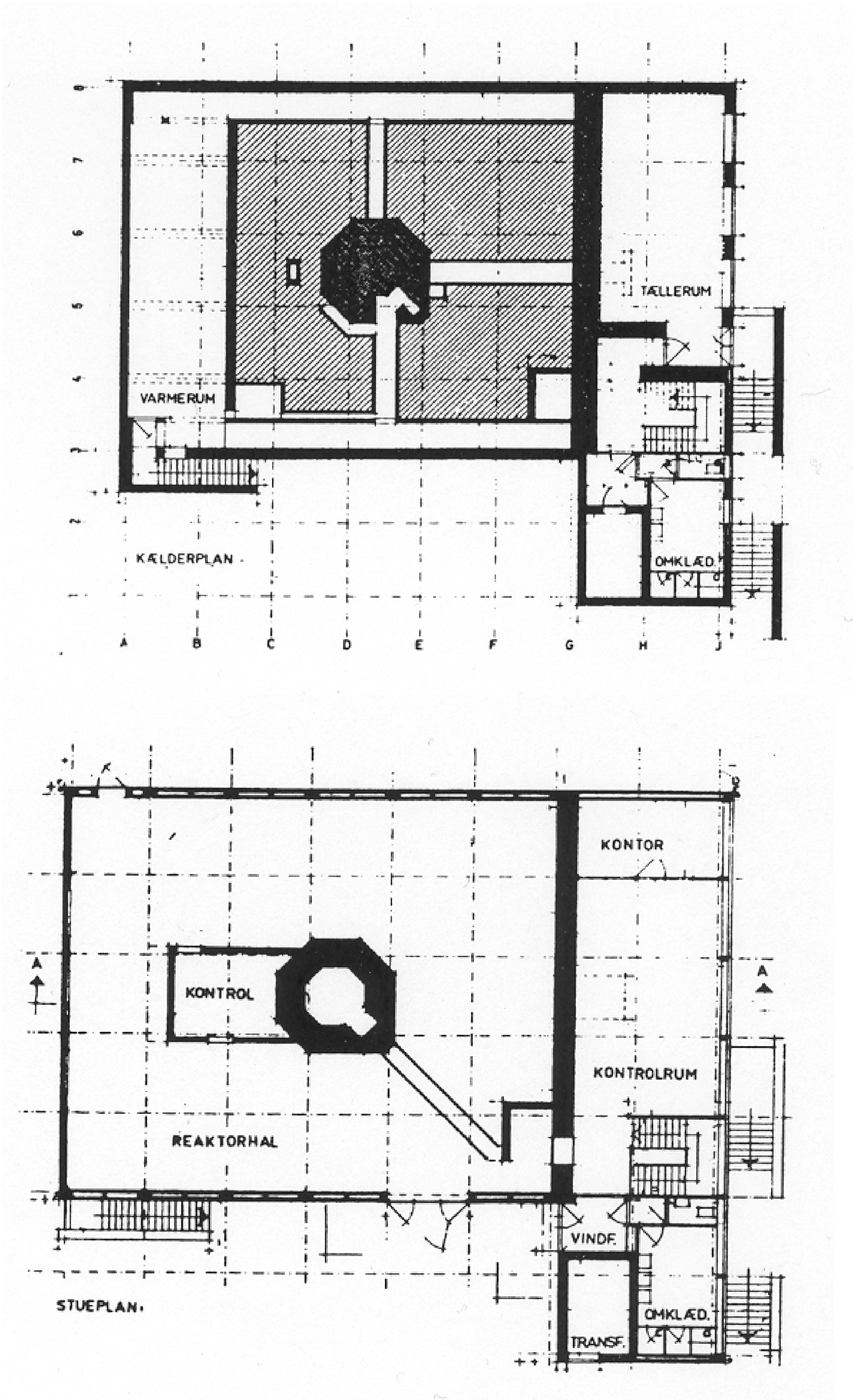


Figure 5-7. Horizontal section of the underground floor (far above drawing) and ground floor (just above drawing) of building 117.

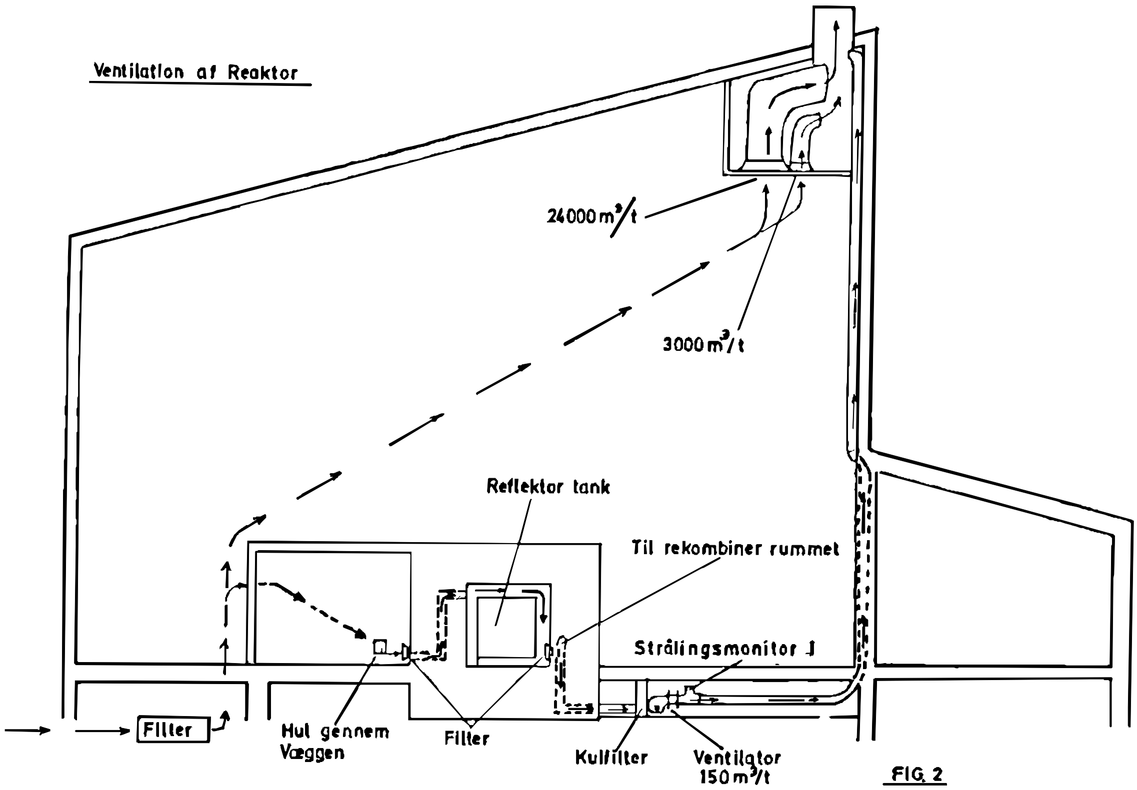


FIG. 2

Figure 5-8. Block diagram of the ventilation system.

6 Overview of the nuclear waste management plan of the TRIGA FiR 1 research in Finland

The FiR 1 –reactor, a 250 kW Triga reactor, see Figure 6-1, has been in operation since 1962. The main purpose to run the reactor is now the Boron Neutron Capture Therapy (BNCT). The BNCT work dominates the current utilization of the reactor. The weekly schedule allows still one or two days for other purposes such as isotope production and neutron activation analysis.

According to the Finnish legislation the research reactor must have a nuclear waste management plan. The plan describes the methods, the schedule and the cost estimate of the whole decommissioning waste and spent fuel management procedure starting from the removal of the spent fuel, the dismantling of the reactor and ending to the final disposal of the nuclear wastes. A new item in the plan will be the implementation of the environmental impact assessment for the decommissioning of the reactor. This will be done in the near future. The cost estimate of the nuclear waste management plan has to be updated annually and every fifth year the plan will be updated completely.

In Finland the producer of nuclear waste is fully responsible for its nuclear waste management. The financial provisions for all nuclear waste management have been arranged through the State Nuclear Waste Management Fund. The cost estimate of the nuclear waste management will be sent annually to the authorities for approval. Based on the approved cost estimate the authorities are able to determine the assessed liability and the fees to be paid to the Fund. The main objective of the system is that at any time there shall be sufficient funds available to take care of the nuclear waste management measures caused by the waste produced up to that time. The system is applied also to the government institutions as FiR 1 research reactor operated by the VTT.

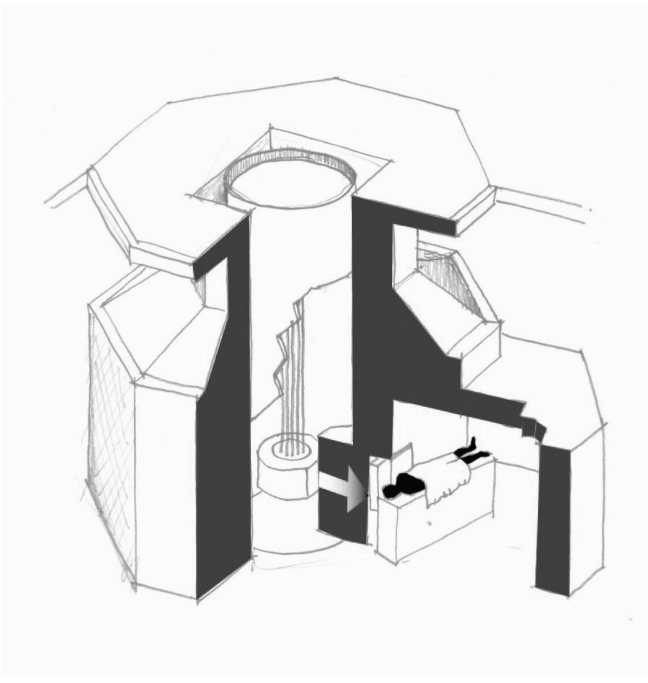


Figure 6-1. The FiR 1 –reactor, a 250 kW Triga reactor, at VTT in Helsinki, Finland.

The Finnish Nuclear Power Companies founded in 1995 a separate company Posiva to develop the technology and carry out safety analysis and site investigations for implementing the spent fuel final disposal. In 1999 Posiva submitted an application for a decision in principle for a final repository to be built at Olkiluoto, on the western coast of Finland. Olkiluoto is also one of the two nuclear power plant sites in Finland. At the same site there is also the new EPR unit under construction. At the end of the year 2000 the Finnish government approved the application and sent it to the parliament for ratification. The ratification took place in May 2001. Separate licenses still will be needed for the construction of the facility, scheduled to start in 2010, and also for the operation, 10 years later. The government alone will grant these licenses and no political aspects are supposed to involve in the licenses.

The back end solution of the research reactor spent fuel has been for years an important problem in many countries, also in Finland. After the recent extension of the USDOE acceptance policy we shall have both of the final disposal options, Posiva option and USDOE option available. USDOE has recently extended the acceptance policy for 10 years until 2016, which means that Posiva and USDOE alternatives are nearly equal for our purposes. The difference between the two options is that the Posiva option has practically no time limit. (The final disposal facility will start in 2020 and will operate for several tens of years.)

In order to further our possibilities to use the Posiva final disposal option we have made safety studies about the long term behaviour of the spent TRIGA fuel in the final disposal surroundings. The main safety aspects, which have to be analyzed and compared to the spent fuel coming from nuclear power plants, are the criticality safety, the solubility of the fuel ($UZrH_x$) in water and the existence of some moving and long-lived radioactive isotopes. The TRIGA fuel is much more reactive compared to the spent fuel coming from nuclear power plants and therefore the TRIGA fuel can not be situated so tightly in the heavy final disposal canister. The Triga containers will be situated in the outer zone of the canister and the inner zone will be left empty. In practice the empty positions will be loaded with dummy assemblies made of cast iron. The criticality safety calculations show, however, that it is possible to load safely all the TRIGA fuel elements in one final disposal canister. This is important, because if the criticality safety would demand the fuel to be divided to two or more canisters, the expenses would also be about twice or more compared to the one canister alternative.

The decommissioning waste is supposed to be disposed of in the repository constructed in the bedrock of the Loviisa nuclear power plant site at the depth of 110 m. Preparatory work has been done to clarify the possible problems of the decommissioning waste of our reactor in the surroundings of the decommissioning wastes of the nuclear power plant. The decommissioning waste studies concentrate mainly on the long term safety of the decommissioning waste disposal. Among others the amount and behaviour of some long-lived radioactive isotopes like C-14 belong to these studies. The main part of the active reactor components will be packed in concrete packages in the waste disposal facility. The volume of the packaged waste is about 100 m^3.

The total costs of the decommissioning of the FiR 1 reactor are assumed to be 5.4 M€. The costs can be divided roughly in four main parts:

- Spent fuel, transport and final disposal 52 %
- Planning of the decommissioning 16 %
- Dismantling of the reactor 16 %
- Final disposal of decommissioning wastes 16 %

Further information on the TRIGA reactor at VTT in Finland is given in Appendix C which is the same as Reference [64]

7 Decommissioning of the uranium reprocessing Pilot Plant in Norway

The text in this section comprises an extract and a compilation of the material in Reference [31].

7.1 Summary

A pilot plant for reprocessing of irradiated fuel was in operation at Institute for Atomic Energy (IFA) (now Institute for Energy technology (IFE)), Kjeller/Norway, during the years 1961 - 1968. In this period about 1200 kg of uranium were processed and plutonium and fission products separated by means of liquid-liquid extraction. The plant comprised of a tube system of more than 6000 meters and a total of 50 tanks, evaporators and extraction columns.

The plant was shut down and partly decontaminated in 1968, but decommissioning proper was not carried out until 1982, and then again during the period 1989 - 1993. The experience from decontamination and dismantling of the plant is reported by the team that performed the decommissioning work. A reprocessing plant is contaminated by radioactive solutions, but due to the absence of neutrons there is no activation of the construction material. The purpose of the decommissioning was to remove radioactive and contaminated material so that the building could be reused for treatment of low and intermediate level radioactive waste. This requires decommissioning to "stage 3" and "stage 2" according to IAEA nomenclature.

The main part of the radioactive deposits inside the process equipment was removed by use of chemical solutions during three consecutive decontamination steps after shutdown of the plant. Remaining liquid in the tubing was a source of contamination during dismantling operations. This was dealt with by means of special tools. The next step was dismantling of process equipment. Before starting safety procedures were issued and alternative strategies for handling waste were conceived. For the dismantling phase many tools needed to be specially adapted to the difficult cutting operations that must take place in narrow cells.

The collective dose to the decommissioning crew was kept as low as 50 mSv and the highest dose received by one person was 10 mSv. It was a major concern to prevent the intake and inhalation of radioactive deposits, especially alpha contami¬nants.
It is important to generate a smallest possible waste volume in the decommission¬ing process. For the major part of the installation it turned out that less waste was ob¬tained by avoiding decontamination methods using liquids, since considerable secondary waste volumes would then be generated. Several factors such as labor cost and the cost of intermediate and final waste disposal must be considered before deciding how far decontamination should be pursued or whether direct packing of partly decontaminated equipment would be preferable. Melting of metal parts for recycling could have been an interesting alternative, but this was not pursued since no such installation is available in Norway.

In general, reuse of decontaminated metal turned out not to be profitable, keeping in mind the low scrap value of the metal and the complications encountered in obtaining clearance from radiological control. The volume of solid waste could be kept low by careful planning, reasonable cutting and tight packing. Using boxes instead of drums for storage of the waste further reduced the volume. On the other hand, lead-shielding blocks can be decontaminated for reuse as shielding material by means of mechanical milling of the contaminated surfaces.

One lesson learned was that conservation of all essential written information and drawings is an obligation that must be recognized by plant management during the operational phase and that strict control of this material is essential when decommissioning is delayed for a longer period.

Figure 7-1. The pilot reprocessing plant.

7.2 Introduction

The Norwegian-Dutch reprocessing pilot plant (Figure 7-1) at Kjeller, Norway, went into operation in 1961. The emphasis was on experimental processing of natural uranium fuel elements from the research reactor JEEP I and testing of the "Purex" process, equipment, instrumentation and various flow sheets, especially for Eurochemic in Mol, Belgium. Another objective was to obtain operational experience and know-how for the design of a full-scale plant. The Swedish "AB Atomenergi" completed an additional facility in 1964 with the intention to study a separation process using a silica gel column. The Norwegian-Dutch "Purex" part and the Swedish "Silex" part were connected in 1964 to increase the purification capacity. The plant is described in [65-66].

The plant was shut down and partly decontaminated in 1968. The dismantling was delayed due to economic reasons and re-started in 1982 for a one-year period [67]. The decommissioning work was resumed in 1988/1989 [68-69] and continued in 1994/1995. Some pipes and tanks in the basement are still not dismantled.

The need to collect the experience from the decommissioning of the Kjeller reprocessing pilot plant was recognized in the Nordic research program NKS 1990-1993. Here the project KAN-1.2 was carried out with the objective to document the decontamination and decommissioning experience and to draw conclusions on preferable decontamination practices. In order to achieve this, decommissioning operations and the related research program were adjusted to each other.

7.3 A brief description of the plant and equipment

The pilot reprocessing plant is shown in Figure 7-1 as it was in 1968 after shutdown. The left cubical part was the Pond building where the aluminum canning was me¬chanically removed from the irradiated uranium elements. To the right of the pond was the hot-cells arrangement with extraction-, evaporation-, and purification cells on the ground floor. The basement contained the dissolver cell, pumps to extraction columns and purification cells. The right hand part of the building contained analytical laboratories, wardrobes, offices and auxiliary equipment. The lead and concrete shielded cells were located in a two-storied building directly connected to the waste treatment plant. The layout of the basement and the ground floor are shown in Figure 7-2 and Figure 7-3, and a photograph of the process system is shown in Figure 7-4. .

Equipment for dissolution of irradiated uranium elements and for subsequent extraction and evaporation was located in six concrete cells. There were a total of 47 vessels and evaporators, 9 extraction columns, and in addition phase separators and various sampling stations, filters etc. Lead blocks were used to provide additional shielding. There were 6000 meters of process piping of stainless steel with an average diameter of 19 mm. All cells were enclosed in metal sheet housings and provided with "drip trays" of stainless steel in order to collect possible leakages and to protect the concrete from contamination.

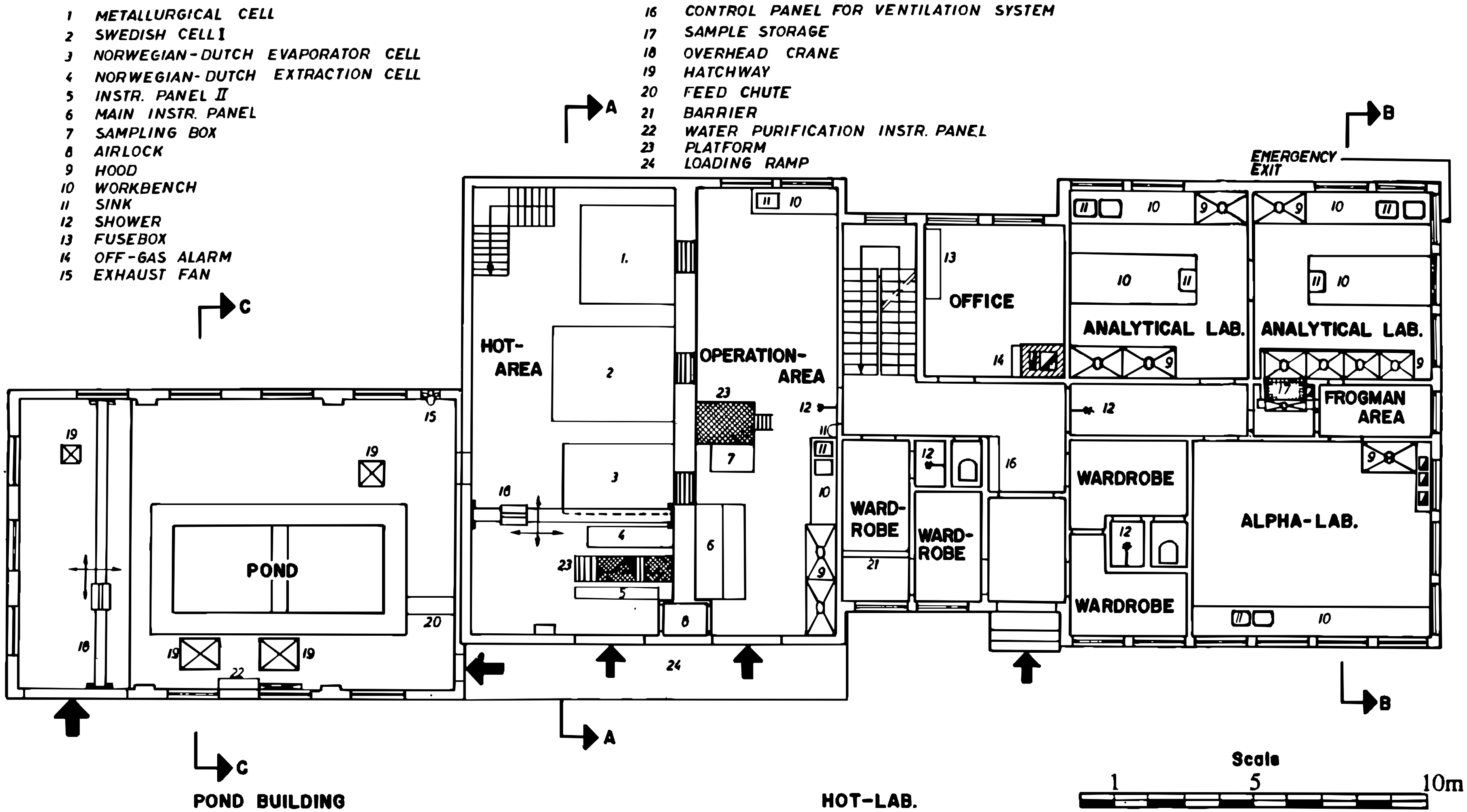


Figure 7-2. The reprocessing plant. Ground-floor plan.

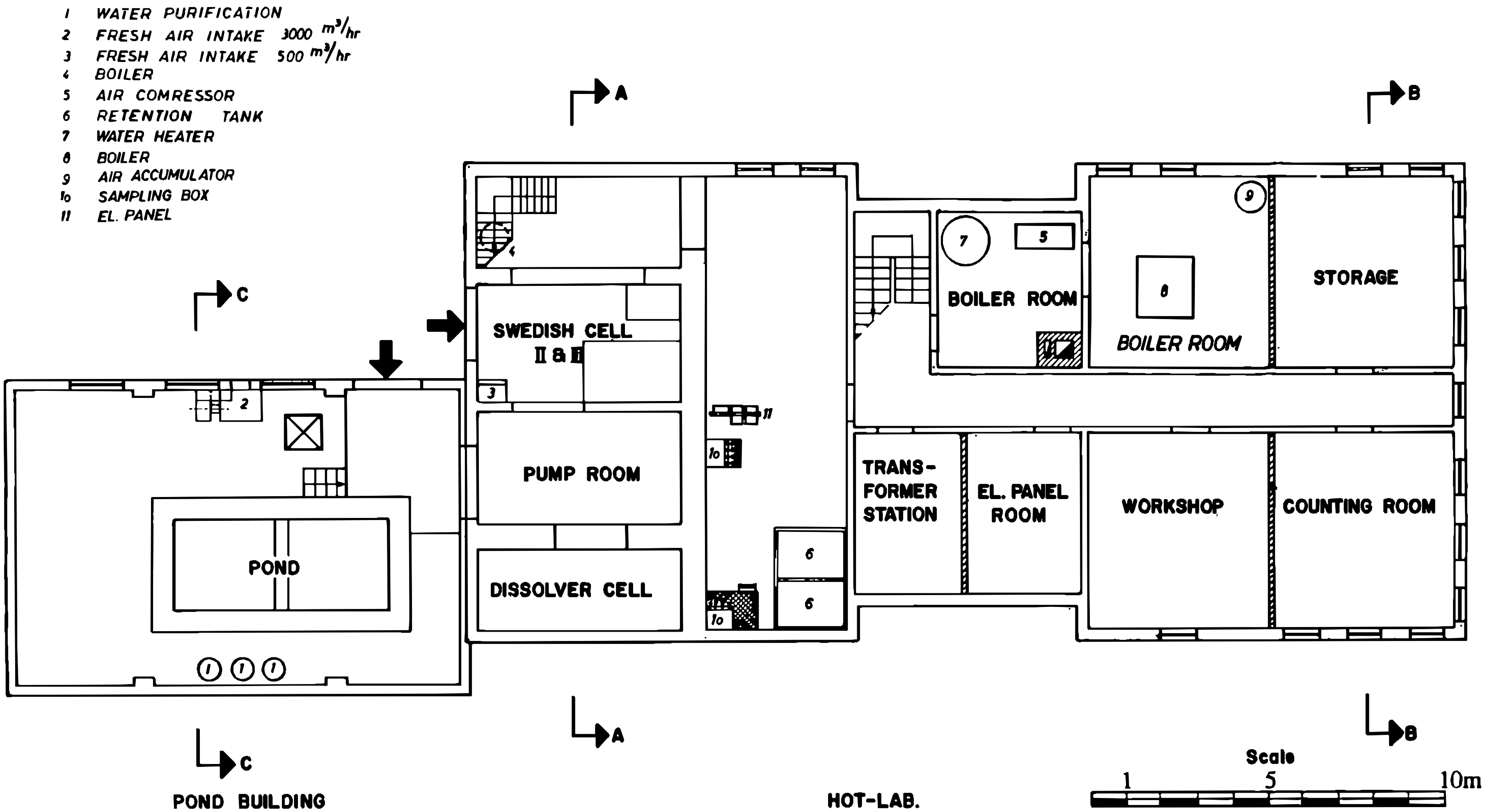


Figure 7-3. The reprocessing plant. Basement plan.

Figure 7-4. An exposed cell for reprocessing.

7.3.1 Radiation levels and preliminary decontamination work

After the shutdown in 1968 the plant had been drained for all radioactive solutions and then internally decontaminated using warm nitric acid and sodium hydroxide. From an average radiation level of about 20 mSv/h, the dose rate was brought down to 1-2 mSv/h or lower measured at close distance from most of the process equipment at the "outer edges" of the plant [70]. A thorough survey of the radiation and contamination levels of all areas and components of the plant was then performed. The survey was made after the removal of all non-radioactive parts in order to obtain better access to the areas that contained the most radioactive components. A further decontamination with oxalic acid/tartrate reduced the levels to 0.15 - 0.3 mSv/h.

In 1982 it was decided to start complete dismantling of the plant. The main objective was to remove the essential part of remaining radioactive materials and equipment so as to permit reuse of the building. The intended use of the building was for radwaste work.

The dose rates measured were not significantly different from what had been found in 1968, but better access to active parts revealed dose rates up to about 10 mSv/h at some "hot spots".

The contamination levels were generally below detection limits on the floor, but on "drip trays" below process equipment, beta levels of up to 5000 Bq/cm2 and alpha levels of up to 500 Bq/cm2 were detected. The average activity was a factor of twenty lower.

7.4 Planning of decommissioning, safety measures and radiation doses

7.4.1 Planning

The purpose of the decommissioning was to bring various parts of the pilot plant (cfr. Figures 7-2 and 7-3) to stages 2 and 3, respectively according to the IAEA definition [71]. Some areas were to be used for waste treatment operations, while others were to be brought to stage 3 for radioactive laboratory work and for non-radioactive work.

Dismantling of the plant was a new challenge to IFE although the Institute had gained some experience with the dismantling of research reactors (JEEP I and NORA). Visits to the Eurochemic plant in Mol and the Kernforschungszentrum in Karlsruhe gave information of great value in the planning of the work.

From a radiological point of view, the most important aspect in preparing the dismantling work was to prevent the intake of radioactive materials, to keep the exposure as low as possible and to avoid the spread of contamination to clean areas. To achieve this goal in practice, detailed working instructions were issued. These included access to the building and plant areas, use of special wardrobes, change of clothing, wearing mandatory protection equipment, routine radiation protection monitoring and specific procedures for the work including handling and dismantling of components.

The general ventilation of the dismantling areas was secured by using the original, ventilation system of the plant. In addition, spot ventilation was used where the risk of inhalation of airborne contaminants was assumed to be high.

The process cells and adjacent areas were relatively narrow and crowded by equip¬ment. In order to reduce contamination of the nearby areas it was decided to complete the disassembly of one cell or area before proceeding to the next.

Normally two persons at a time were to deal with the dismantling. No one was allowed to work alone.

7.4.2 Radiation protection monitoring

The radiation protection of the dismantling workers aimed at:

a) reducing the exposure to external radiation to the lowest practical level,

b) preventing the intake of radioactive materials in the body.

In addition to standard film dosimeters, TL-dosimeters were used for finger monitoring, but have shown to be of minor importance. Also direct reading dosimeters were worn for "personal use" during the work. Portable monitors for beta and plutonium activity were used to for supervising air contamination. Traces of activity have been recorded during the first dismantling period only.

7.4.3 Radiation doses

External exposure

The dose rate levels in the plant were generally low and did not present any special problems. As the work was proceeding there was close communication between the dismantling workers and the radiation protection staff. Practicing on inactive components prior to dismantling has contributed positively to the actual low exposure records.

The recorded collective effective doses from film dosimeters and the expected dose from remaining work were:

Decommissioning work (including waste treatment)	46	manmSv
Remaining work	4	manmSv
Total	50	manmSv

It was difficult to correlate the received doses to specific work situations since they were not measured for that particular purpose. A total of ten persons was involved in the actual decommissioning work for shorter or longer periods. The highest total dose received by one operator was 10 mSv.

Internal exposure

Regarding intake of radioactivity inhalation risk was of greatest concern but also the ingestion risk was taken into account. Ingestion was avoided by instructions for change of protective clothing, washing, and monitoring of skin (hand) contamination when leaving the work area.

In most operations the general ventilation system of the building was supplied with spot ventilation, and this was regarded as sufficient to control the inhalation risk. But in operations where the dust risk was assessed to be high, dust masks or air stream helmets were used.

Intake of activity was routinely monitored every second month by whole body monitoring and radiochemical analyses of urine samples. The whole body monitoring revealed only small amounts of caesium-137. The effected dose was estimated to be below 0.1 mSv. The urine analyses have primarily focused on plutonium, but no samples showed concentrations above detection level ($< 1 \times 10^{-5}$ Bq/litre).

7.5 Decommissioning procedures

7.5.1 General

The first step in a decommissioning operation was acceptance of all safety and protection procedures. Thereafter planning of the practical approach could take place. Several procedures were specified and some examples are given below.

Organization, responsibility and management must be clearly defined. A decommissioning procedure is a wholly practical and technical operation and successful completion depends upon the presence of a qualified and well-motivated work staff. In the early phase of the process the complete planning must be outlined. However, it is more or less impossible to foresee in detail the problems that will arise and therefore the operation must allow improvising with respect to the original plans.

Wardrobes, protective clothing and control- and safety procedures must be ready beforehand, and the selection of tools is important. Management of material, active or inactive, must be prepared for. The type of waste containers to be used will depend on packing methods and the possibility of cutting metallic components into smaller parts. If severely contaminated metals are to be decontaminated, alternative methods should be investigated since such surfaces may turn out to be difficult to clean. For dismantled parts that cannot be packed in situ, temporary storage areas must be arranged.

A convenient way to start decommissioning work is first to dismantle abundant inactive components, and thereafter approach areas/components with increasing radiation levels. Due to the risk of contamination it is of importance that dismantled parts are removed from the working zone as soon as possible. If more hot cells are to be dismantled, they should be completed one at a time.

A decommissioning operation must be performed as teamwork. In order to solve the various practical and technical problems encountered it is useful to engage the crew in group discussions. Hereby innovative suggestions are often brought forward. In the decommissioning project described here the knowledge of the operation crew has been of great importance, and the presence of members of the original plant staff was of great help. Without their assistance, the work would have been much more costly and troublesome. If decommissioning is delayed for a longer period, the availability of all written information, drawings etc. must be assured [72].

Dismantling of process equipment and of shielded cells is a physically tough job. A good health condition of the operators is a necessity since lifting heavy components and difficult working positions occur frequently.

It is recommended that:

- that instrumentation and ventilation equipment be kept in operation as long as possible
- that cutting, shearing and packing of active components be done inside the decommissioning area
- that spot ventilation be used to prevent the spread of dust
- that there is a strict definition of the working zones with respect to the degree of contamination possibilities to avoid cross-contamination
- to wash and clear the working area daily and to pack and remove the produced waste for further treatment.

7.5.2 Complications

Some construction features turned out to complicate decommissioning. Joints in extraction and evaporation cells had been connected as screwed and flanged systems. When dismantled, they showed that leakages of radioactive liquids had occurred. These had caused contamination outside of the process equipment so that it had to be handled carefully to avoid inhalation of alpha contaminants (uranium and plutonium). The majority of the joints in the "Silex" part were welded, and here only a few leakages were observed. Screwed or flanged joints are thus not suitable for components exposed to thermal gradients if they cannot be inspected daily.

Piston pumps used for pulsation of the extraction columns had caused high contamination in parts connected to the pumps and in the pump enclosure. In other parts of the plant all pumps for radioactive liquids were of the double membrane type and had no flanged joints except near the active dosage heads. This part was placed inside fitted boxes equipped with leakage alarms, and this construction caused much less contamination problems.

Explosive drilling clamps were available to drain U-pipe constructions. Unexpectedly, other parts of the pipelines could also contain liquids, so that cutting would result in a spurt. Additional awareness of this phenomenon is mandatory.

7.6 Special tools

Whenever practical, standard commercially available equipment and tools can be used for decommissioning. Thus, where industrial safety aspects are of less concern, conventional tools are used, such as jigsaw, for cutting sheet steel and pipe lines nibbler, for cutting sheet steel bolt cutters, for cutting electric cables, pipelines, etc.

Work with standard tools within temporary containment systems can be difficult and potentially hazardous. Also, modifications may have to be made to enable tools to be operated safely and efficiently by workers wearing protective clothing [73]. A range of tools can be adjusted including [74] cutting tools where sections of the cutting part normally would be unprotected ("hands-off" tools), dismantling tools, lifting aids, etc.

7.7 Choice of decontamination methods

7.7.1 Introduction

In the course of a complicated decommissioning sequence, decisions have to be taken at several stages, depending on the outcome of the previous step. Optimization of subsequent steps is thus a recurrent task. Several aspects have to be considered.

The 50 vessels, evaporators, and extraction columns had several inlet and outlet lines in addition to instrumentation lines. Reuse of the equipment for other purposes was not regarded as possible. Due to the very compact construction, the piping system had to be cut into short pieces. No practical reuse of the pipe ends was possible.

The plant had been decontaminated by chemicals and the radiation level lowered by a factor of 100, to an average of 0.2 mSv/h. Many of the short-lived nuclides had decayed. Under these circumstances, further decontamination of the process equipment had to be evaluated in order to generate as small waste volumes as possible. Only the scrap value of the equipment was interesting for comparison with the cost of decontamination and waste handling.

Melting of metal parts for recycling was not considered as an economical alternative. This is due to the relatively small size of the plant and the fact that no melting plant for radioactive material is available in Norway. Also, the complication with obtaining a license for recycling must be taken into account.

7.7.2 Decontamination or disposal as waste

In this section various calculations are presented that indicate whether it would be cost efficient to decontaminate piping systems, vessels, and shielding blocks. The scrap value was compared to the cost of decontamination.

Unprofitable decontamination: decontamination compared with cutting and packing

Piping systems

The 6000 meters of contaminated pipelines had an average of 16.5 mm in outer diameter and a wall thickness of 2.5 mm. The weight is approximately 1 kg/m, that means 6000 kg totally. The scrap value for stainless steel was NOK 3,-/kg, giving a total of NOK 18 000,-. If the pipes are molten and sold to a steel factory, the most realistic price is the scrap value. On the other hand, stainless steel as a raw material resource represents a somewhat higher value.

The economic comparisons indicate that, since decontamination would require considerable efforts and costs, direct packing as radioactive waste could be achieved for one quarter of that cost. The total decontamination cost including chemicals, operational cost, waste handling and disposal are calculated to NOK 500 000,-.

This is an example of unprofitable decommissioning, where high packing tightness of steel components can be obtained, and where secondary waste volumes would have exceeded the volume of the item to be cleaned.

Summary:

Scrap value	NOK	18 000,-
Decontamination	NOK	500 000,-
Direct packing	NOK	150 000,-

Vessels

40 vessels might have been decontaminated, but due to the complicated inlet and outlet connections, it was not possible to reuse them either for radioactive or non-radioactive purposes.

With an average volume of 200 1, wall thickness 4 mm and a weight of 80 kg, the scrap value of all the vessels was NOK 10 000,-.

Two vessels were decontaminated manually. The cost is listed in the summary below. Decontamination by high pressure water flushing was a possible alternative, but the investment cost would have been high. In addition, treatment, solidification, and final disposal cost would arise. It turned out that placing vessels inside other vessels was the most economic method, about half the cost of manual cleaning. Space between the vessels could be filled with other waste items. This packing system results in eight packages of 1 m3 each.

Summary:

Scrap value	NOK	10 000,-
Manual cleaning	NOK	150 000,-
High pressure flushing	NOK	320 000,-
Vessel inside vessel	NOK	70 000,-

Profitable decontamination: shielding blocks

Lead shielding blocks

Lead shielding blocks that are contaminated represent a considerable value and could be reused after mechanical surface treatment (milling). A typical lead block has the size 10 x 10 x 5 cm and a value of NOK 300,-. 1000 blocks (out of a total of 3000) were contaminated, and their value was NOK 300 000,-.

An average thickness of 0.25 mm of the surface was removed, and when the lead chips are molten, they give a volume of 12 litres waste. The investment cost for the installation was NOK 40 000,- including a milling machine. The net reuse value for the 1000 lead blocks was thus NOK 165 000,-. They could be reused for shielding purposes.

Concrete shielding blocks

The cells were shielded with concrete blocks. Some 550 blocks were somewhat contaminated. 350 of them were reused for a shielding wall (25 m^2) in the waste treatment plant. For this purpose the permitted contamination level was 2 Bq/cm^2 α-activity and 20 Bq/cm^2 β/γ-activity. Maximum radiation level was set to 10 µSv/h. The blocks were coated with two layers of paint. The saved waste disposal costs are calculated to NOK 50 000,-.

Radioactive material had normally penetrated the outer 1-2 mm (up to 5 mm), and two methods were used for removal: chiseling, and chemical attack by 3 molar nitric acid + 2% hydrofluoric acid. The acid makes the surface "boil", and after 10 minutes the contamination can be wiped off with absorbing paper. Treatment of 200 blocks (1800 liters) resulted in 100 litres of waste. The working hours and waste treatment cost was NOK 15 000,-, and the reuse value of the blocks was estimated to NOK 50,- pr. block, i.e. that a total of NOK 10 000,- was saved. The cost for packing and storage of 200 blocks (1800 liters) as radwaste was estimated to NOK 50 000,-.

7.7.3 Chemical decontamination

Surfaces

All structural work of steel, located close to the hot cells such as framework, steel sheets and steel sections must be handled as potentially contaminated. Likewise service- and auxiliary equipment and steel constructions such as sampling stations, staircases, landings and staircase railings in the operational area may have to be treated as contaminated, even if the aim is to obtain clearance from being radioactive waste.

Both stainless steel and carbon steel, often painted with a two component hardening paint, can be decontaminated by:

- pickling with corrosive agents (unpainted surfaces)

- paint removers
- water abrasive blasting (using glass beads)

or a combination of these methods.

The success of such treatments will vary from case to case, since the handling is very individual, depending on the degree of adhesion of radioactive material to the components.

Parts of the aluminium structural work was decontaminated to a level, sufficient for clearance of radiological control, by using 2 % sodium hydroxide for one minute at 60°C.

In most cases it was possible to decontaminate metal parts to the exemption level by using different chemicals and time consuming operations, but the costs, the secondary waste volume produced and the final treatment should be born in mind. If more secondary waste than the volume of the original part is produced then decontamination is unprofitable. One example: A square meter steel sheet with a thickness of 2 mm has the volume of 2 litres. It is not possible to decontaminate this to exempt level without exceeding a final waste volume of 2 litres.

If there is a risk of radioactive dust, the surface of components to be removed should be moistened and wiped before dismantling. Spraying with water, detergents or nitric acid (for stainless steel constructions) is useful. Smooth and accessible stainless steel surfaces were normally decontaminated by manual wash with 3 molar nitric acid, with or without 2- 4 % hydrofluoric acid, depending on the degree of contamination. The addition of hydrofluoric acid is very effective. A drawback was, however, that the waste has to be neutralized to the corrosive action.

Process equipment

In situ decontamination to an exemption level of the process equipment was not possible in practice and due to the crowded construction the pipelines had to be cut into short pieces (on average 1 meter). As shown above, decontamination of pipelines and vessels turned out to be unprofitable. This is not necessarily true in other cases with less complicated and more similarly shaped plant arrangements, and where contamination levels are different.

As an experiment smaller pieces were treated for 6 hours at 80^0C with forced circulation, using 3 molar nitric acid. The inside remaining activity level was 2.6 Bq/cm^2 β/γ-activity and a factor of ten lower for α-activity. Addition of 3 % hydrofluoric acid to 3 molar nitric acid brought the inside activity level down to 0.6 Bq/cm^2 β/γ-activity (background level) but due to the corrosive effect, problems with concentration of larger volumes of liquid waste before the solidification step must be taken into account. In practice, decontamination of piping must be done in another way with longer pieces and by forced flow of the decontamination solution.

7.8 Measurement of surface contamination

7.8.1 General

The goal of the decontamination operations was to remove radioactive deposits from surfaces so as to either allow clearance from regulatory control or to optimise waste management. The operations are followed closely by measurement of the remaining surface contamination.

In the case dealt with here, a portable contamination monitor could be used for measurements and for exempt control of surfaces. The measuring area is 50 cm2 and the efficiency is 10% for alpha as for beta/gamma activity.

For unrestrictive reuse the metal scrap shall be free from any radioactivity, and for restrictive reuse inside active working areas the limitation was < 0.4 Bq/cm^2 for α-activity and < 4 Bq/cm^2 for β/γ-activity [75]. The general levels at IFE to allow clearance for reuse of decontaminated materials have been lowered by a factor of two in order to account for the uncertainties in measuring the activity levels of surfaces.

The Radiation Protection Section at IFE, Kjeller, performed the radiation protection control during the decommissioning. This section had a long experience in measuring technique, also while the plant was in operation. They were well acquainted with the history of the actual parts to be measured. This was helpful when it comes to decide whether these items should be handled as radioactive waste, or whether they can be subject to exemption from radiological control.

7.8.2 Internal control of narrow steel pipes

The control measurements inside steel piping were performed using a micro-probe GM-detector, mounted to a 1 m long insulated steel rod and connected to a detector at the other end. With this equipment, measurements can be made inside pipelines with a diameter down to 16 mm and lengths up to two meters by inserting the probe from both sides. The goal of constructing this instrument was to make possible an inside control of narrow steel pipes for possible clearance as non-restrictive material. With the equipment developed it was possible to detect contamination levels that approach the natural background [76].

7.9 Waste treatment

7.9.1 General

Waste management is a major cost factor in decommissioning. Once a decommissioning strategy has been decided, e.g. whether to arrive at stage 1, stage 2 or stage 3, a main objective is to arrive at the lowest possible overall cost, and this mostly implies keeping the waste volumes as small as possible. This can be achieved by detailed planning of the dismantling process. The choice of either decontamination with the view of clearance or direct packing can be difficult because the secondary waste may very quickly exceed the volume of the dismantled parts.

Even where pipelines or other process equipment can be decontaminated (cf. section 6), the most reasonable solution often turns out to be direct packing. This is especially the case if tight packing can be achieved with components placed inside each other.

7.9.2 Methods

Absorbent paper, plastics, textiles, rags etc. used for outside cleaning of low-con¬taminated equipment were shredded and thereafter compressed inside concrete shielded drums. They can also be incinerated. Floor-covering and mixed items can be handled in the same manner. Spot contaminated floor plates, framework and sheet steel were cleaned by manual rubbing or wet blasting, using glass beads. If this is unsuccessful, the spots can be removed using a cutting blowpipe. The metal scrap is cast into drums or stainless steel boxes.

Remaining liquids from process equipment were transferred to 200 1 drums lined with polyethylene, and thereafter solidified by the addition of cement and additives. For compacted waste the original volume was reduced to 15 %, and for burnable waste to 2 - 3 %. For liquid waste that had to be solidified without any concentration, the volume increased by a factor of 1.6.

7.9.3 Choice of containers

In the early stages of the decommissioning described here, nearly all of the waste was collected in shielded drums. Thus in the beginning, standard 210 1 drums were used as the outer container for most of the waste. At that time only smaller parts had been dismantled. It soon turned out that the 210 1 mild steel drums were not satisfactory as the only container. A better performance was expected of a 110 1 drum placed inside a 210 1 drum with 5 cm of concrete in between. This type is mostly used for compression of paper, plastic, rags, brushes, protective clothes etc. In some cases corrosive liquids were observed inside dismantled pipelines and therefore the use of a 210 1 stainless steel drum as a waste container was also evaluated. The 210/110 1 drums were later replaced by stainless steel boxes, with the following dimensions: length x depth x width = 120 x 80 x 80 cm.

The choice of containers does not only depend on their cost, but also on the cost of operation and on intermediate storage and final disposal. In the project reported here, the temporary storage cost at IFE is NOK 2 300,- per drum and the cost of final disposal (rock depository) [77] is assumed to be NOK 3 000,-/m^3 (NOK 2000,-/m^3 for rock repository and NOK 1 000,-/m^3 for transporting). One working hour is arbitrarily set at NOK 300,- (internal cost). It was also necessary to compare how much pipelines and how many smaller items can be packed into the different drums and containers available. It was possible to pack 245 m of pipelines (172 kg) and smaller items of an average dimension of 16.5 x 2.5 mm inside a 110 1 drum (se Figure 7-5). A box can be filled with 2200 m of pipelines and smaller items. The utilization of the volume inside the container can be expressed in two ways, either using the net volume of the pipe material, or using the outer volume actually occupied by the pipes.

Calculations show that the 860 1 stainless steel boxes have the lowest storage space requirement for a given amount of solid waste. The same amount of solid material filled into 210 1 drums would require 1.7 times this storage volume, and if using 220/110 1 drums, 3.3 times the volume would be needed. The use of larger containers also means less cutting work. The average cutting lengths for the three types of containers are 400, 500 and 1190 m, respectively.

Figure 7-5. Drum packed with steel pipes and other metallic objects.

7.10 Discussion of costs

The total decommissioning cost was approximately NOK 6 million, including investment in tools of NOK 0.6 million and a similar amount for waste treatment. The cost of final disposal was calculated to NOK 0.1 million [77]. Rehabilitation of the building was not included. The value of reusable components was estimated to NOK 1.5 - 2 million.

The investment cost of the plant was approximately NOK 42 million (1992 kroner) and the operating cost approximately NOK 80 million including waste treatment and decommissioning. The total cost of the project was thus NOK 122 million.

The decommissioning cost amounts to 14 % of the investment and 5 % of the total project costs. The ratio between labor cost, tools cost, and waste treatment during decommissioning was 8:1:1. The value of components, if reused, amounts to 1.4 % of the project cost.

The total labour involved adds up to 10-12 person years. 70-80 % of the decommissioning time was used for the practical dismantling work, 10-15 % for treatment of waste, and 10-15 % for safety and management.

7.11 Conclusions and recommendations

From the decommissioning of a small reprocessing plant, some general conclusions can be drawn.

The role of plant management is decisive in creating the right basis for a successful decommissioning operation. Firstly, sufficient funds should be put aside from the outset of any project of this kind. Secondly, management must attach importance to the work by showing positive encouragement of the crew. The decommissioning operation itself may otherwise be regarded as a negative task that may discourage operators. Motivation of the crew is thus an issue for management.

High flexibility of operating teams should be strived for by combining different skills, including experience from work in active areas and mechanical abilities.

In the course of the decommissioning work, it is important frequently to re-examine the planned steps. In discussions with the team, genuine solutions generated by its members can be obtained.

Strict housekeeping and permanent radiological surveillance help to maintain a high working moral. Transportable and automatic radiation monitors should be available to control radiation fields where operations take place. Daily washing of the working area will contribute to avoid spread of contamination. Dismantled parts should be removed from working areas for further treatment, either for decontamination or for direct packing as radioactive waste. Tools to be used for dismantling can be adapted so as to facilitate operations in narrow spaces.

Prior to each decontamination step the generation of secondary waste and the total cost should be estimated in a realistic way. The value of metallic scrap for reuse in commercial products may be low compared to the operating cost and the secondary waste treatment costs.

In many cases it may turn out that secondary waste volumes will exceed the volume of the original object to be decontaminated. In the present case, comparison of direct packaging with decontamination and the use of boxes instead of drums reduced the waste volume to half the original estimate.

Delayed reprocessing can be a preferred option, when advantage can be taken from the decay of short-lived nuclides. Delayed decommissioning requires availability and updating of all relevant information about the plant, including drawings, operating instructions, etc. On the other hand, availability of members of the original operating crew will facilitate decommissioning to a great extent. Thus, the reduced doses to the operating crew that can be obtained by delaying operations must be weighed against the disadvantages.

After termination of operations of the plant it is important to maintain a staff witch is given responsibility to take care of safety, inventory, the building and first of all the archives. During the period up to the start of decontamination and dismantling it is important to avoid uncontrolled situations with respect to removal of equipment and the risk of contamination.

8 The decommissioning of Sweden's nuclear research reactor R1

8.1 Conclusion

Sweden's first nuclear research plant, R1, was located near the Royal Institute of Technology in Stockholm in an underground closed chamber. Further information on the reactor and its operation can be found in [21], see also Appendix E. The research plant started operations in July 1954. After being used for research purposes and isotope production for 16 years and an operating time of 65000 hours, the reactor was finally closed down in June 1970.

In 1979, Studsvik suggested complete decommissioning of R1 and radiological decontamination. A radiological survey was started in May 1979, after which Studsvik made a detailed investigation for the demolition of the reactor. The preparation work began on site in April 1981 with an overview and continued until the end of October 1981.

During the planning of the decommissioning project the requirements of SSI (the Swedish Radiation Protection Authority) were the line of aim. Those requirements were meant for complete decommissioning of the plant and a "greenfield" level for the rock chamber. As a first step in the decommissioning project all the equipment in the plant hall and the areas nearby were surveyed before being handled as exempted material. As the project went on, bit by bit of the exempted material was screened and measured and then placed in Berglöv boxes, a special kind of box constructed for the purpose of transporting active waste, or placed inside the plant to be handled later on. In preparation for the exemption project the localities were divided into classified areas and non-classified areas. Thus all parts of the facility were searched and screened. Different methods were used to control the individual doses to which the staff was exposed.

A reactor plant has very little conventional equipment and structures and so the work had to be planned using special arrangements, specially constructed equipment and machines as well as different kinds of protective shields. The project experienced both drawbacks and progress. Some of the surprises that caused some problems were:

- The water pipes and lamps were in poor condition due to corrosion and the electrical equipment had to be replaced.
- When SSI changed the limits for decontaminated goods, the new limit was considerably lower than proposed from the project team so the amount of material classified as active waste increased a lot.
- Misjudgements were made about the activity in the graphite from the thermal column.
- There were problems in obtaining spare parts for the old equipment. Luckily some old spare parts had been left at the R1.

- There were problems in dealing with the decommissioning of equipment from areas that had not been classified before, which caused a lot of delay since discussions on this problem had to take place with the authorities at the end of the project.

There were also some circumstances and inventions that were important for the project and its progress:

- Much of the human capital was still accessible, from the time when the plant was in operation, and their knowledge was very valuable to the project.
- One of the advantages of the decommissioning project was that there had been no serious incidents in the course of the plant's sixteen years of operation.
- A lot of preparation work had been done when the plant was closed down.
- Transportation using the Berglöv boxes went almost twice as fast as expected.
- There were no accidents during the whole project due to the discipline among the whole staff.
- The special smear test equipment, Berthold LB2711 was a valuable help in measuring all the smear tests in the clearance project.
- The major success in this project was a newly invented machine called "MiniMax PH 250". This machine could handle the special kind of concrete in the biological shield.

The time schedule was followed and the radiological doses to the staff were under control and kept low. Studsvik was granted a total of MSEK 25 for the demolition of R1, starting in the second quarter of 1981, the demolition work lasting until May 1983. The expenses were underestimated and the budget was MSEK 21.7 instead of MSEK 25. The decommissioning project for Sweden's first nuclear research plant R1 proved to be a success financially and technically, and even the time schedule was followed almost as planned.

8.2 Introduction

It has proved to be very important to gather significant experience from different types of nuclear decommissioning projects in order to simplify the cost calculation for this kind of project. The most difficult part of these projects is estimating the cost, which is closely associated with the radiological and technical issues. Thus it is most valuable to gather experience and data from other projects on radiological, technical and financial aspects. It is most important for future decommissioning projects to have access to these different kinds of experience. Only a few nuclear facilities have been decommissioned completely. The decommissioning project for Sweden's first nuclear research plant R1 proved to be a success financially and technically and even the time schedule was followed almost as planned. The experience from this project may be very valuable for other projects in the future even if the conditions vary from one facility to another.

8.2.1 Background

Sweden's first nuclear research plant, R1, was located near the Royal Institute of Technology in Stockholm in an underground closed chamber. The research plant came into operation in July 1954. It had a rated effect of 1 MW. After being used for research purposes and isotope production for 16 years and an operating time of 65000 hours the reactor was finally closed down in June 1970. The reactor was cooled and moderated by heavy water. Metallic, natural uranium was used as fuel. After closedown, the fuel, heavy water and ion exchange system were transported to Studsvik. In the second half of 1970 work started on sealing the plant. A final radiological scan had been made before the plant was closed in 1971. The plant was sealed for eleven years before the R1 decommissioning project started in May 1979.

In 1979, Studsvik suggested complete decommissioning of R1 and radiological decontamination. A radiological survey was started in May 1979 and after this Studsvik started a detailed investigation for the demolition of the reactor. The preparation work began on site in April 1981 with an overview of the electrical installations and continued to the end of October 1981.

Studsvik was granted a total of MSEK 25 for the R1 decommissioning project. The project started in the second quarter of 1981 and the demolition work lasted until May 1983. The expense was underestimated and the budget was SEK 21.7 instead of MSEK 25 . The time schedule was followed and the radiological doses to the staff were under control and kept low. This report deals with the decommissioning of Sweden's first nuclear research plant R1 from three different points of view: radiological survey, technical planning and cost calculation.

8.2.2 Purpose

This report is for the purpose of analysing three main issues associated with the R1 decommissioning project. The three issues are: radiological survey, technical planning and cost calculation. There are some questions concerning these issues: What were the experiences during the project as regards those three main issues? Were there any positive surprises and/or negative surprises? What were the main criteria for this project to become a success?

8.2.3 Critical treatment of sources

During the major R1 decommissioning project at least 30 different kinds of reports were written. In this report mainly one of them is used as a reference, the Studsvik Report : *Rivning av forskningsreaktorn RI Stockholm* [78]. The report is actually a summary of the whole R1 decommissioning project .

This report is intended as an outline and the material and facts are gathered from the report mentioned above unless another source is referred to.

To provide the reader with the right background a conclusion in English of the Studvik Report Studsvik/NW-84/627:*Rivning av forskningsreaktorn RI Stockholm* [78] is attached to this report as a supplement.(Supplement 1).

8.3 Radiological survey

Initially a radiological survey was carried out of the whole site in 1979. (Supplement 2).
The whole project was planned in three different phases:

- Radiological survey of the whole facility.
- Planning for the segmentation of components and estimate of the individual doses.
- Planning how to handle the waste [79].

Later on a thorough schedule was made for the decommissioning work. This was done by Studsvik to obtain an estimation of the decommissioning project.
The preparatory work started off in April 1981. A special heavy-duty filter had to be installed for the air conditioning. Monitoring equipment to control the personal doses had to be built and protective equipment for the staff had to be arranged.

The individual doses were expected to be as low as 4 manrem (40 mmanSv) because the activity in general in the facility was estimated as being low [80].

A lot of equipment to measure the activity on the localities and in various pieces of material also had to be set up. All areas including the reactor plant had to be divided into classified and non-classified areas. Special packaging for all the active waste and transportation of this material to Studsvik had to be prepared.

8.3.1 Preparation

Before the decommissioning work started all the localities were classified areas. Arrangements for stepover limits, dressing rooms, monitors for individual surveys and showers were made.

A study was made of how to measure the activity inside the Berglöv boxes. A Berglöv box was a special package for radiological waste.

Some concrete was extracted from the biological shield by drilling and tests showed that there was induced activity only 25-30 cm from the inner part of the biological shield, as well as in some of the channels inside the shield. (Supplement 3).

Mapping

The preparatory project survey included the following:

- The rate of activity was measured on the spot by dosimeter etc.
- Measuring of the activity inside the graphite, the graphite that had been taken from the plant's thermal column.

- Measuring of the boron guts taken from the plant's thermal column.
- An estimate was also made for the active/contaminated waste as follows [79]:

Carbon steel/ aluminium / lead	c. 110 tonnes
Cadmium sheet	c. 5 "
Graphite	c. 68 "
Concrete	c. 75 m³

The activity in the construction was roughly estimated to:

- 1TBq CO-60
- 0.2 GBq Cs-B4
- 25GBq Eu-152
- 5GBq Eu-154

There was no evaluation of the amount of C-14 inside the plant at the first examination. The so called "Wigner-effect" was studied but there was no risk of spontaneous generation of heat from this special phenomenon.

Method

During the planning of the decommissioning SSI's (Swedish Radiation Protection Authority) requirements were the line of aim. Those requirements were intended for complete decommissioning of the plant and a "greenfield" level for the rock chamber. Clearance levels were also proposed for the material and fromthis an estimate could be made of the amount of material needed to be stored in Studsvik and how much could be placed on a landfill.

The clearance levels fixed by the SSI were:
- < 5 kBq/kg material
- < 1 MBq/m³ liquid

The levels fixed by the SSI were lower than expected and therefore the amount of active material to deal with was higher than first estimated by Studsvik.

8.3.2 Radiological ongoing work

When all the localities including the reactor plant had been divided into classified and non- classified areas, the decommissioning project could be started.

Radiological survey during decommissioning

As a first step in the decommissioning project all the equipment in the plant hall and the areas nearby was surveyed before it was handled as exempted material. As the project went on bit by bit, material from the plant was screened and measured before it was placed in Berglöv boxes or placed inside the plant to be handled later on.

The work on cutting up the reactor vessel was done down in the uranium container, where the reactor vessel was placed. (Supplement 4).

After demolishing the top of the plant the surface of the inside of the reactor vessel was screened.

After the whole decommissioning work on the reactor vessel a radiological survey was made inside the graphite reflector. The same procedure was followed with the thermal column although the lead door had been replaced first. The dose rate was found to be considerably higher than expected and therefore the procedure for handling the graphite material had to be reorganized. This was all due to the fact that the test material taken from the graphite column did not represent the total activity properties.

The whole work of decommissioning the graphite reflector gave a collective dose of 49 millimanSv divided between 8 persons. (Supplement 5).

After dismantling the graphite reflector, another radiological survey was made inside the biological shield. The maximum dose-rate was 15 mSv/h of the surface. Thereafter the cadmium sheet metal was decommissioned and together with the mechanical components in the biological shield, this element of the project caused the greatest collective dose, a total of 56 millimanSv divided among 10 persons.

The concrete from the biological shield was tested during the preparation work, which made it easier because now the clean concrete could be torn down to prevent cross contamination.

The demolishing of the biological shield caused a collective dose of 16 millimanSv divided among 15 persons.

Demolition of the engine room equipment and cooling tower as well as the equipment of the laboratory areas went on without any negative surprises. All material and areas were thoroughly screened for contamination. In particular the cavity bellow the biological shield was measured and it was found that even more concrete had to be removed to get rid of all the activity.

Cleaning of the facility for clearance

SSI (Swedish Radiation Protection Authority) had fixed a limit for clearance of 8kBq/m^2. At this point no more cleaning was necessary.

Before the clearance project for the localities, all of them were divided into classified areas and non-classified areas. Thus all localities in the facility were searched and screened.

A vacuum cleaner was used in the reactor hall and nearby localities, and the surfaces were wiped. Radiological screening was again carried out first roughly and then more thoroughly, following a special schedule.

All surfaces in classified localities were checked into squares measuring one m² each. All the squares were numbered and smear tests were taken from all of them. They were evaluated for general beta as well as for tritium. There were also screenings of the surfaces from a portable instrument.

The non-classified localities were screened with a portable instrument and two smear tests had been taken from each.

All results had been recorded, archived and stored at Studsvik. The SSI has been given copies of all these records.

A lot of concrete was found to be still contaminated and had to be taken away. After another round of cleaning all surfaces they were washed with cleaning agent. The work went on like this in the whole facility and on the 10th June 1983 the classification into activity zones was no longer necessary.

An estimation of the activity still present was made from the smear tests taken from all the localities. Those turned out to be very low.

Those tests were batched and classified by gamma spectrometric instrument. All localities were controlled by smear tests and by measuring total beta. Some of these tests were batched and examined for type of nuclide and activity with regard to gamma nuclides. The batch tests showed very low values and thus that surface contamination was very low.

Because of the decommissioning of R1, SSI had required samples and analyses of the soil from the surrounding area. Samples were taken 50-1000 metres from R1. No high levels of activity were found.

Equipment and methods for individual dose control

Different methods were used for the control of the individual doses to the staff.

Individual- dosimeters, TDL (Thermoluminescence), were used for the staff at R1. The dosimeters were read once a month. For special tasks working-dosimeters were used to survey these special operations. This was requested by SSI. The background radiation could be distinguished using special dosimeters.

Whole-body radioactive contamination monitor: A monitor called Herfurth type 1361EC.

Whole-body counter: A special whole-body counter at Studsvik called HUGO. This was done when the staff were occupied with critical tasks. After some critical tasks an internal *contamination count* was made on the staff. A total of 66 whole-body readings were made at Studsvik in the HUGO.

Blood tests were performed on the staff before and after the decommissioning of the Cd-walls in the biological shield.

The total dose to the staff participating in the decommissioning of R1 was 142 millimanSv, divided among 25 persons as follows:

1 person	28 mSv
6 persons	10-20 mSv
8 "	1-10 mSv
10 "	0.1-1 mSv

In the course of some parts of the project the staff used dosimeters to obtain information about the dose on that particular occasion, and the results are shown in Table 6.1.

Tabel 6-1. Compilation of the data from the dosimeters on the project staff.

Elements of work	**Collective dose (mmanSv)**	**Persons**
Lifting the tank and cutting the flange	4.4	7
Cutting apart the tank	6.4	5
Demolishing the graphite reflector	49	8
Demolishing mechanical equipment inside the biological shield	56	10
Demolishing the biological shield	16	15
Radiological survey	4.4	4
Transport	3.5	16
Lifting the tank and cutting the flange	4.4	7
Sum	**140**	

Radiological classification

Air

During the whole demolition project a HEPA-filter was installed in the air channel. This channel was connected with the chimney. The filter was changed every week and the dose rate of the surface of the HEPA-filter (special kind of fresh air filter) was controlled and analysed in a gamma spectrometer frequently. The total outflow of activity was 0.4 kBq/day during the 40 week period. Other arrangements made to prevent activity into the air were:

- Special equipment (Counting Ratemeter, RM-51M) was placed to control the air in the room next to the hall where the plant was situated.
- Breathing mask filters were controlled.
- Special kinds of portable equipment monitored the air.

During critical phases special tents made of plastic material were put together and used as protection from any spread of activity. There were special arrangements with extra filters and special arrangement that took care of air in special spots plus special filters.

Water

The water from the classified areas was separated from the water from showers and lavatory-basins. Thanks to this method the last mentioned category of water never exceeded an activity level of >50 kBq/m³, 20 points below allowed limits. The total discharge of this kind of water was approx. 150 m³. There was no discharge of water with an activity level of >50kBq/m³. The upper limit allowed was <1MBq/m³ if the water was to be channelled to the wastewater vent. Water with >1MBq/m³ was transported to Studsvik.

Waste control

All sorts of waste and recycling material produced from R1 were measured and registered.

A special kind of steel boxes, called Berglöv boxes, 600 litres; were used for storing the waste. Each box was registered, numbered and the activity and nuclides were measured. The equipment used for this purpose was a gamma spectrometer, Canberra S-85 and as detector GeMac-detektor (GeMac=Germanium Multi Attitude Cryostat). The calculations of the activity were made by a HP-97 calculator. The Berglöv boxes were measured from two different directions, thus making it possible to make a calculation of the gamma radiation in the material.

All the packages were controlled by smear tests and screening instruments before they were taken from the site. The smear tests were measured in equipment with beta detection. The detection limit was 2kBq/m². If the surface contamination exceeded 8 kBq/m² after it had been wiped off, the packages were sent to Studsvik.

Recycling material was carefully screened and no material was cleared if the detection limit 2 kBq/m² had been exceeded.

The measurable limit was fixed to 5kBq/kg, the same limit fixed for low active concrete at Studsvik's landfill. First there were a lot of problems due to the high sensitivity of the radiation detector and its equipment to vibrations and noise. The equipment was replaced and it was all sorted out. The equipment was also protected from the background radiation by 5 cm lead.

In order to gauge the average activity inside the concrete a small piece was taken for control.

All waste with an average of < 5kBq/kg was placed at the landfill at Studsvik and covered by at least 1 meter of soil.

Waste with an average of >5kBq/kg was placed at a separate location at Studsvik.
All the documentation dealing with waste management was stored and the waste placed in Studvik has been registered in special files.

Studsvik got permission from the Swedish Nuclear Power Inspectorate, SKI, to transport the material from the site of demolition of R1 to Studsvik. Active waste transports followed the European Agreement on the International Carriage of Dangerous Goods by Road (ADR). When the limits of the ADR were exceeded special permission for transportation from R1 had to be granted by SSI.

A form for external radiological transport had to be issued for all transportation.

Screening equipment

A Nuclear Enterprise PCM5 with double scintillation detector for both alfa and beta was used for local controls of contamination. The sensitivity was 4 kBq/m².

A radiation monitor specially designed for floors was used for large floor areas, FH 545. Sensitivity 30 cpm kBq/m². A PCM5 was used for small areas. Sensitivity 0.67 cps kBq/m².

Smear tests were also used as a supplement at the clearing and the Berthol LB 2711 was most successful. Its sensitivity was 2 kBq/m². The GM-detector was also used for the smear tests. This equipment had a sensitivity of 10/100 seconds at each kBq/m²(Eu-152).

A gamma spectrometer, including 17% HPGE detector and 4000 channel Canberra 85 analysis equipment were used for screening surfaces.

The activity and nuclides were measured for each Berglöv box. The equipment used for this purpose was a gamma spectrometer, Canberra S-85 and as detector GeMac-detektor (GeMac=Germanium Multi Attitude Cryostat). The calculations of the activity were made by a HP-97 calculator.

Tritium- and gamma spectrometric classifications were made at the laboratory at Studsvik.

8.4 Technical Planning

A reactor plant has very little conventional equipment and structures. The biological shield inside R1 was made of a special kind of concrete. The work had to be planned with special arrangements and protective shields. There also had to be plans for working at a distance, due to the radioactivity. Therefore, a lot of special machines and arrangements had to be made. Some of the dismantled parts and equipment from the plant were extremely heavy and required special transportation.

8.4.1 Planning

Some technical preparations were made when the vessel was closed down. The vessel was drained of heavy water, which was transported to Studsvik. The heavy rods were placed in the uranium well for cooling. The plant was lined to prevent activity from leaking out. There was also a flood alarm system installed.

Before the practical decommissioning work started all those involved worked thoroughly on a project plan. This plan was carefully worked out and turned out to be very helpful throughout the project. It was a comprehensive plan dealing with the radiological work as well as the demolition work, transportation, organisation, time schedule, costs and environmental security.

8.4.2 Preparation

To prepare for the heavy transports the floor had to be reinforced in some places and a special wagon had to be constructed for the heavy components.

A special saw that could be manoeuvred remotely was constructed for cutting apart hot radiological components.

Equipment that could handle graphite blocks remotely, as well as equipment that could remotely demolish parts of the flanges was constructed.

There were a lot of tools designed solely for some special elements of the project. After a thorough examination of the electricity system it was found that several parts had to be replaced as well as some associated electrical equipment.

8.4.3 Technique for the demolition part of the project

All loose equipment was screened and transported out of the reactor hall and localities nearby. Engines and gears from the reactor construction were dismantled. The reactor hall was emptied. The uranium well was opened and its heavy lid was placed in a corner of the reactor hall. The uranium well was then filled and prepared for the reactor tank later on. This part of the project ended in cleaning and painting the floor of the reactor hall to prevent decontamination later on.

Work on the reactor tank

A major part of the work regarded the reactor tank. The two big lead doors from the thermal column were taken away and using a specially constructed wagon, four blocks of 1 cm thick concrete, the previous radiation shield inside the thermal column, could be taken out. Using the same wagon a thick graphite pin inside the graphite column could also be taken away. All of the six lids were placed behind a wall of concrete blocks in a corner of the reactor hall and the first and most contaminated lid was placed underneath and covered by the others.

Working from a distance with a special saw the seven hot flanges could be cut loose from the reactor tank. The remote saw was placed behind a lead shield on the specially constructed wagon mentioned before. After cutting loose the hot flanges it was possible to lift the reactor tank out of the biological shield by means of an overhead crane. There had to be four more types of technical action before the reactor tank at last could be placed inside the uranium well. In this position a lid from the biological shield covered the tank and this lid could also act as a platform during the work on dismantling the tank.

Some air conditioning arrangements had to be made before the major work of cutting apart the reactor tank could start. The reactor lid and five stainless flanges were dismantled and the lid could be taken up to the hall and placed on the wagon for transportation of heavy goods. From there it was cut apart by the remote saw, which was placed behind a radiation shield, and placed in Berglöv boxes for further transport. The flanges from the tank could be handled in the same way when they had been cut from the tank by a plasma cutting tool. (Supplement 4).

The plasma cutting tool was also used to cut apart the upper part of the tank. The lower part of the tank had to be cut with a saw.

Dismantling equipment inside the biological shield

The work on dismantling equipment from the biological shield went on well apart from a minor piece of equipment that was stuck as a result formation of corrosion products. In order to be more effective and reduce handling time a conveyer belt and some more equipment were obtained. By the thermal column the graphite reflector was dismantled and the conveyer belt could be placed in the centre of the biological shield. The graphite block could be placed in Berglöv boxes directly. For a short time one person had to loosen the blocks by hand.

The cadmium-and aluminium plate was dismantled by hand. They were folded together with the cadmium plate, covered by the aluminium plate and thereafter placed in Berglöv boxes. Other mechanical components inside the biological shield, such as the inner lead door, were cut and/or hatched down. Flanges were left behind for the moment but covered with lead blocks and lead carpets.

Dismantling of the biological shield

A newly invented machine called "MiniMax PH 250" was used to demolish the biological shield (Supplement 6). This was good both as regards radiological matter and the economical aspect. This machine could handle the special kind of iron ore concrete from the biological shield that was both soft and tough (leathery). The MiniMax was driven by electricity and manoeuvred remotely by just one person. It billed the concrete with a hydraulic hatch hammer and the concrete fell right down into a Berglöv box. The hydraulic bill hammer was also equipped with a jet and could spray water mist over the concrete dust and fix it on the spot. The reinforcing iron and beams, however, were cut off by means of a fusing burner. The machine was small enough to fit into the elevator. The technique of minimising contaminated concrete was to take away the clean concrete first so as to start from the outside to prevent cross contamination.

At special critical phases of the job with the biological shield it was necessary to erect a tent to protect against dust from concrete with high contamination. Some parts of the biological shield had surface dose rates as high as 20 mSv/ so had to be cut by a fusing burner operated from a distance. Special air conditioning and extra evacuation inside a tent were arranged in this case.

Dismantling of equipment inside the engine house, cooling tower, laboratory areas etc

Special air conditioning and extra evacuation were set up when working in the laboratory areas. The MiniMax machine was also useful in all other parts of the construction where walls and other parts were made out of ordinary concrete and cast iron. They were pulled down and by the MiniMax and the parts were placed directly into Berglöv boxes and after measuring the activity they could be transported to Studsvik.

The big stainless waste reservoirs from the laboratory areas were closed and sent to Studsvik intact. In other areas the equipment was dismantled without any further problems.

The sewers in the facility were dismantled. Just one sink was left behind to take care of washing water. The waste water reservoir for showers and washing was also left undemolished.

8.5 Financial risk identification

The initial examination was comprehensive and dealt with estimates for the radiological work, transportation, organisation, time schedule, costs, collective doses and environmental safety. This examination was thorough and there was also a thorough lay-out dealing with the dismantling work that meant a lot to the progress of the decommissioning project and prevented drawbacks.

8.5.1 Progress

Since only eleven years passed from when the reactor was finally closed down in June 1970 until the start of the decommissioning project in May 1981 much of the human capital was still accessible, from the time when the plant was in operation, and their knowledge was very valuable to the project.

One of the advantages for the decommissioning project was that there had not been any serious incidents during the sixteen years the plant had been in operation.

A lot of work had been done when the plant was closed down, which was an advantage for the decommissioning project. For example the budget for the decommissioning project did not have to deal with the costs for drainage foil of the heavy water since this had already been transported to Studsvik. The fuel rods had been placed in the uranium well for cooling and thereafter they had been transported to Studsvik. The biological shield had been sealed in order to prevent any radiation leakage. In 1971 a radiological survey was made of all the localities.

During the whole project everything was noted in a journal which has been of great use. A reference group with members from the authorities and power industry was involved in order to exchange experience.

Transportation using the Berglöv boxes went almost twice as fast as expected. With a special kind of lifting equipment five lorries could be loaded instead of two or three. For that reason the demolition could go on with just half the number of stops expected. Thus there was less need for the hired crane lorry and the lifting truck.

There were no accidents during the whole project due to the discipline among the whole staff. Nobody accepted carelessness about protective equipment or protective measurement. The staff had good knowledge about the instructions concerning transportation and handling of heavy goods. They had also experience of radiological work or other difficult environmental work.

It was very helpful for the management of the project that the managers had been involved in the planning of the project and the radiological survey. Their sound knowledge of the facility and the condition of the plant meant that there were no discussions as to how and when different parts of the project were to be carried out.

8.5.2 Drawbacks

In the course of deliberations with the authorities, SSI (Swedish Radiation Protection Authority) and SKI (Swedish Nuclear Power Inspectorate) the limits for decontaminated goods were fixed to the level of <5 kBq/kg for the concrete waste for it to be allowed to be placed at Studsvik's landfill. The limit for discharging liquid waste to the municipal outflow was < 1 MBq/m³. This was considerably lower than proposed by the project team. Thus the amount of material classified as active waste increased a lot.

Some misjudgements were made about the activity in the graphite from the thermal column. It was first thought that these tests could be representative for all activity in the inventory but it was discovered, when radiological screening was carried out inside the graphic reflector and the thermal column, that the activity was higher than expected. Then new tests were taken from the graphite from the reflector and these tests showed a level of activity 60 points higher than assumed. This caused a delay of four weeks.

The plant had been sealed for 11 years and during this time the temperature had been 10 degrees Celsius in a very damp environment. For this reason some of the equipment was corroded and the associated dismantling involved some problems and delays. The water pipes and lamps were in poor condition and for example the electrical equipment had to be replaced.

There were problems in obtaining spare parts for the old equipment. Luckily there were some old spare parts left at R1.

During the initial examination part of the project, the project team had not discussed with SSI the problems of equipment from areas not classified before. This caused a lot of delay because the discussions had to take place at the end of the project. There were also some delays in dealing with the evaluation of some of the measurements.

8.5.3 Technical equipment successful for the project

The special smear test equipment, Berthold LB2711, together with scaffolding equipment LB1026 and a printer used in the second half of the project, helped in measuring all the smear tests in the clearance project.

The major success in this project was a newly invented machine called "MiniMax PH 250". As mentioned before this machine could handle the special kind of concrete that the biological shield was made of. The MiniMax machine was also useful for all parts of the construction made of concrete. MiniMax saved more than MSEK 1.5 as less staff had to be involved than a conventional dismantling would have required. Although the start of the dismantling of the biological shield was delayed by four weeks because of the high radioactivity and 5 m^3 more concrete had to be dismantled, the time schedule could be followed thanks to the MiniMax.

8.5.4 Cost calculation

How to deal with the problems of depositing radioactive waste in geological formations had never before been addressed by the authorities in Sweden. For this reason there were no routines for the actions of the authorities. In December 1982 some guidelines were given and in July 1983 decisions were made concerning permission under the special nuclear law. Therefore, the cost of the practical part of handling the low activity waste was planned in another project. This was because there was not enough time for the authorities to process the permits during the R1 decommissioning project.

Low and medium level waste was stored at Studsvik. This was an interim arrangement while the facilities for final storage of this kind of waste were under construction. This is obviously not included in the budget.

The documentation process was not as complicated as nowadays. For example there was no law about environmental impact assessment, security accounts and proposed decommissioning plans. There was no need for the budget to include these kinds of documents

Budget

The total cost for the decommissioning of the R1-plant was MSEK 21. 7 (with a granted budget MSEK 25). The overall allocation of the budgeted costs are shown in Table 6-2, and a more detailed itemisation is shown in Table 6-3.

Table 6-2. The allocation of the budgeted costs for the decommissioning of the R1 reactor.

Cost item	Cost MSEK
Preparatory study, radiological mapping, preparation work on site	2.9
Project management, staff management, communication with the authorities, reports etc.	4.5
Service and costs for running the facilities during the demolition period	0.7
Mechanical demolition, including taxes	9.5
Transportation, packing, waste treatment	2.1
Radiation protection and decontamination	2.0
Total	**21.7**

Table 6-3. The allocation of the incurred costs for the decommissioning of the R1 reactor.

Cost item	Cost MSEK
Radiological survey and pre-examination 1979-1980	1.2
Preparation on site until 1981, special equipment developed, initial experimental tests and measurement.	1.7
Communication with authorities, project management, management, reports and visits	4.5
Service and management at R1 1981-1984	0.7
Dismantling of the parts of the reactor and equipment inside the reactor hall apart from the concrete	1.7
Mechanical dismantling in engine room, laboratory etc.	4.0
Dismantling of the biological shield and the concrete in laboratory locations	3.1
Radiation protection including radiological measurements	1.3
Transportation including costs for packing (Berglöv boxes)	1.1
Waste disposal at Studsvik AB	1.0
Measurement for decommissioning, cleaning etc.	0.7
Tax	0.7
Total	**21.7**

8.6 Supplements

8.6.1 Supplement 1. Conclusions from the Studsvik summary report on decommissioning of the R1 reactor

This supplement is a translation from Swedish to English of the conclusions of the Studsvik summary report [78] on the decommissioning of the R1 research reactor.

The reactor was located near the Royal Institute of Technology in Stockholm in an underground closed chamber. The research plant started operations in July 1954. After being used for research purposes and isotope production for 16 years and an operating time of 65 000 hours the reactor was finally closed down in June 1970. The reactor was cooled and moderated by heavy water. Metallic, natural uranium was used as fuel. After closing, the fuel, heavy water and the ion exchange system were transported to Studsvik. The rest of the plant was sealed.

In 1979, Studsvik suggested complete decommissioning of R1 and radiological decontamination. A radiological survey was accordingly started in May 1979. Based on this, Studsvik started a detailed investigation for the demolition of the reactor.

Studsvik was granted a total of MSEK 25 for closing down R1, starting in the second quarter of 1981, and the demolition work lasted until May 1983.

The preparation work began on site in April 1981 with an overview of the electrical installations and continued until the end of October 1981. The R1 plant was then divided into different zones from a radiological and ventilation point of view. New in- and out ventilation filters were installed, lifting- and transportation routes were examined, sanitation equipment was completed, monitoring equipment for staff, waste, ventilation were installed and tested.

The actual demolition work started at the end of October with the scanning and removal of all movable equipment in the main hall and adjoining sectors. The so-called uranium container underneath the reactor floor was opened and prepared to be able to contain the reactor tank for its dismantling.

After the opening of the biological shield and the removal of the seven radiological flanges, the tank was lifted and put in the uranium container where it was cut up and prepared for the journey to Studsvik. The work was done with both plasma cutting as well as with more traditional mechanical cutting.

The graphite reflector, consisting of chunks of graphite weighing up to 60 kg, was removed from the biological shield via the thermal column with the help of a conveyor belt. The graphite was then packaged in steel crates and shipped to Studsvik. After the removal of the graphite from the biological shield, the mechanical components were removed.

We had now reached the stage where the disassembly of the exceptionally reenforced biological shield could begin. For the execution of this task we chose a company that had developed a special machine “MiniMax PH 250”. The machine is electro-hydraulic, can be manaeuvered from a distance and demands little manpower. Its jack-hammer was provided with four jets that sprayed water mist over the site in order to reduce the concrete dust. It did not take more water than was absorbed by the concrete waste. The demolition went according to plan. We first tore down the outer (approximately 1.5 m) concrete layer of the biological shield which according to earlier radiological mapping consisted of pure concrete. This material with an activity content of 5 kBq/kg could be deposited for disposal at Studsvik according to a decision by SSI (The Swedish Radiation Protection Authority). The remaining concrete with a higher activity content was treated like the rest of the radiological waste from R1.

Together with the demolition of the biological shield, dismantling of equipment in engine rooms, cooling towers and in laboratories was carried out. MiniMax was also used to demolish concrete containers and radiation shields in these areas.

All the waste and recyclables produced in R1 were documented and nuclide-specific measurements were made with a gamma spectrometer. Almost all the waste was transported to Studsvik. The exception was electrical engines, ladders, stairs and such items from non radiological spaces that after scanning and control by SSI were shipped off as non-radioactive waste. Solid waste, except for big lead doors and steel lids from the biological shield, was segmented and packaged in 600 litre steel crates for transportation. This also included concrete to the landfill at Studsvik. For transportation of the radiological hot stainless flanges a special lead shielded bottle was used.

The total amount of waste transported from R1: 750 tonnes of concrete to a regular landfill, 340 tonnes of concrete to a special radiological waste repository, 116 tonnes of metallic waste etc, 6 tonnes of liquid waste and 52 tonnes of graphite. The total activity content was approximately 800 GBq.

All the work in controlled areas was done with the staff dressed in special clothes. Protective measures for the staff were safety helmets, breathing mask, hearing protectors, special protection overalls covering the whole body with an air supply etcetera in accordance with normal workplace regulations and radiological considerations

The total collective dose to the staff participating in the project was 142 milliman Sv divided between 25 men. The dominating dose was from the demolition of the graphite reflector that gave 49 milliman Sv divided between 8 persons. Demolition of the mechanical equipment in the biological shield gave 56 milliman Sv divided between 10 persons. The staff was also controlled by whole body count in the HUGO-facility at Studsvik. No internal contamination was found.

On no occasion was measurable activity released to the environment during the demolition.

The technical demolition was finished in May 1983, about a month later than planned. At this point the radiological measurement for the clearance of the R1 localities started. All the surfaces in the rooms and spaces that had earlier been classified as radiological were now divided into 1 m^2 squares and were scanned with properly portable scanning equipment and were smear tested.

The surfaces in the non classified localities were scanned with the scanning equipment in the same way as the classified ones, but smear test were used to a smaller extent.

The limit for taking further decontamination was fixed at 8kBq/m^2 by SSI. None of the test results exceeded this limit and were in fact normally considerably lower. Collection of samples (batch measurement with the smear test) from the classified localities showed surface contamination on an average of 80Bq/m^2 and for unclassified localities 20Bq/m^2.

The measurements for clearance were terminated in October 1983. Application to SSI for clearance was made in February 1984. Thereafter SSI performed control measurements and in the beginning of February 1985 SSI concluded that no more restrictions from a radiological point of view were needed for the further use of the localities.

The total cost for the decommissioning of the R1-plant was MSEK 21.7 (with a granted budget of MSEK 25) and was allocated as follows:

Cost item	**Cost MSEK**
Preparatory study, radiological mapping, preparation work on site	2.9
Project management, staff management, communication with the authorities, reports etc.	4.5
Service and costs for running the facilities during the demolition period	0.7
Mechanical demolition, including taxes	9.5
Transportation, packing, garbage treatment	2.1
Radiation protection and decontamination	2.0
Total	**21.7**

8.6.2 Supplement 2. Survey over the reactor construction

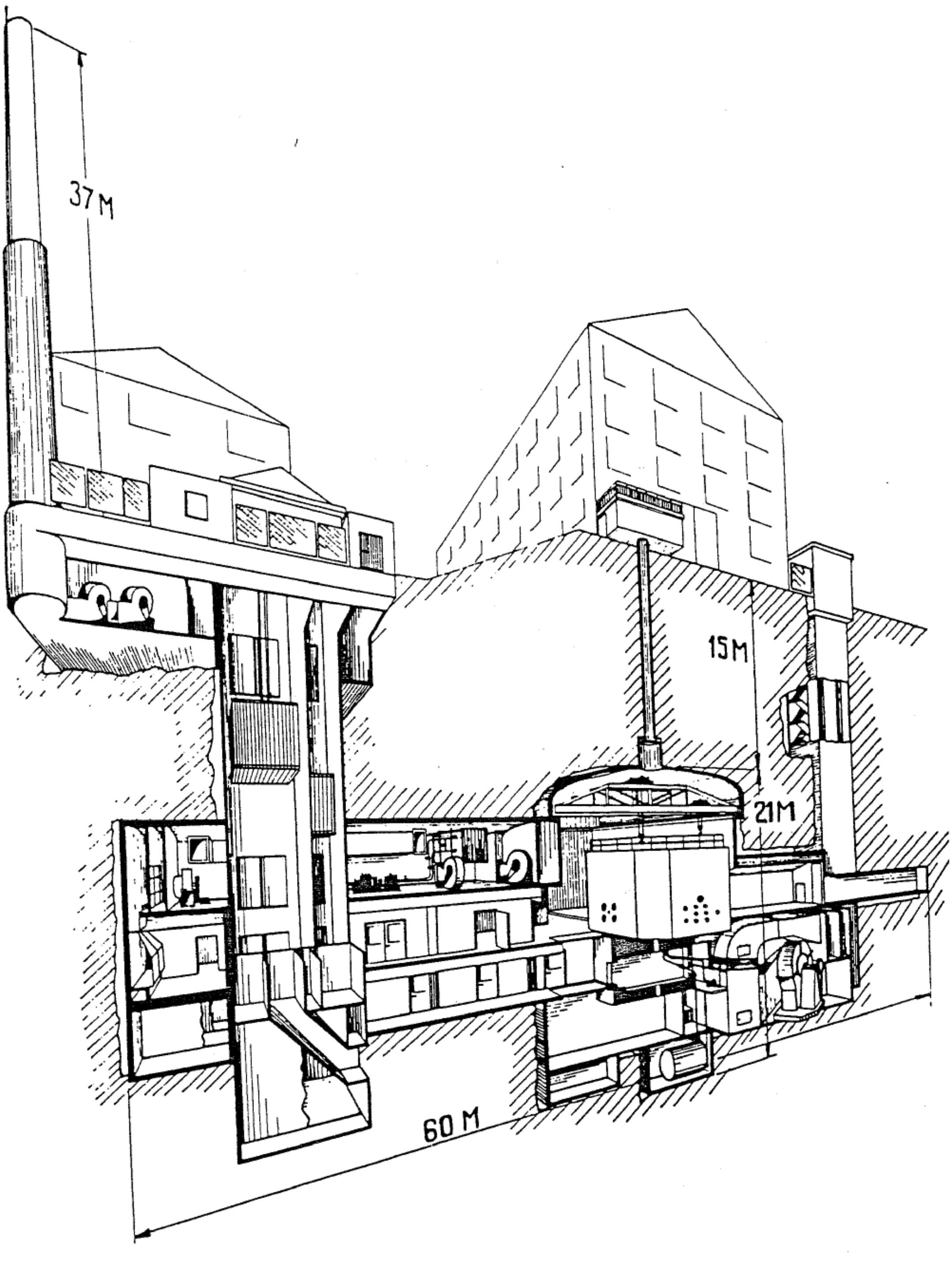


Fig 3-1 Reaktoranläggningen

8.6.3 Supplement 3. Inside the biological shield, the reactor vessel and the graphite reflector.

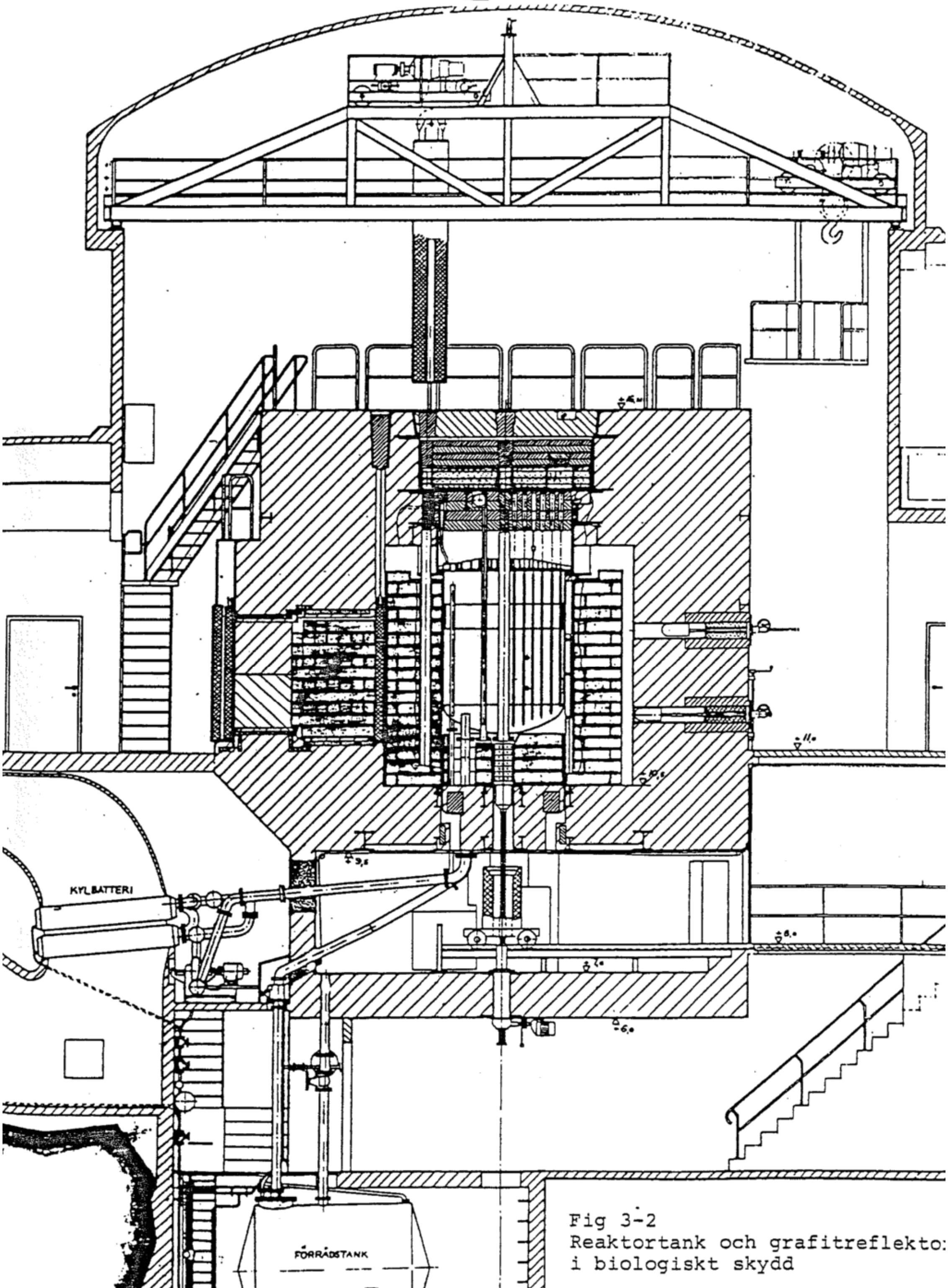
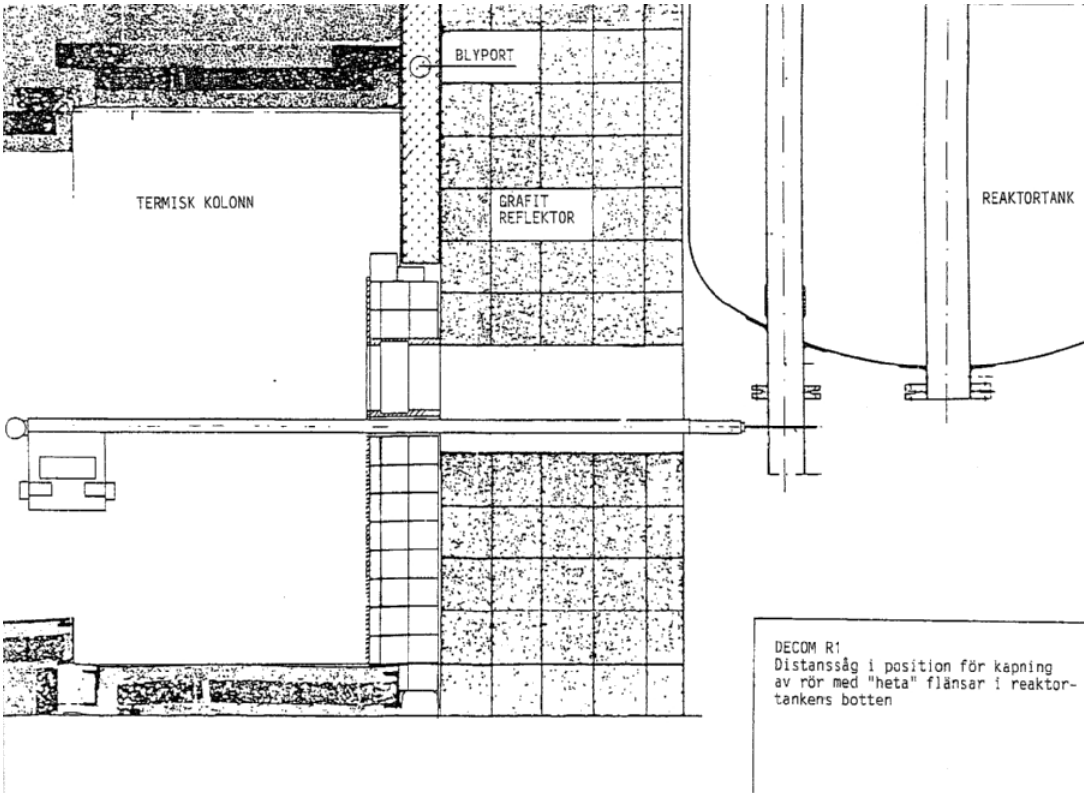
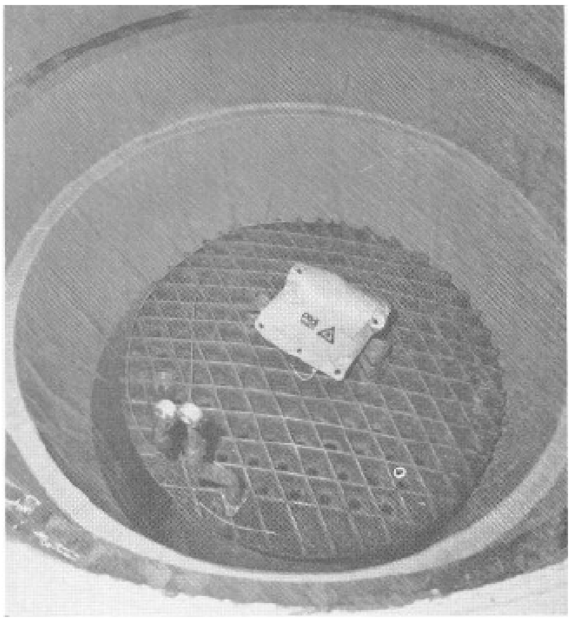


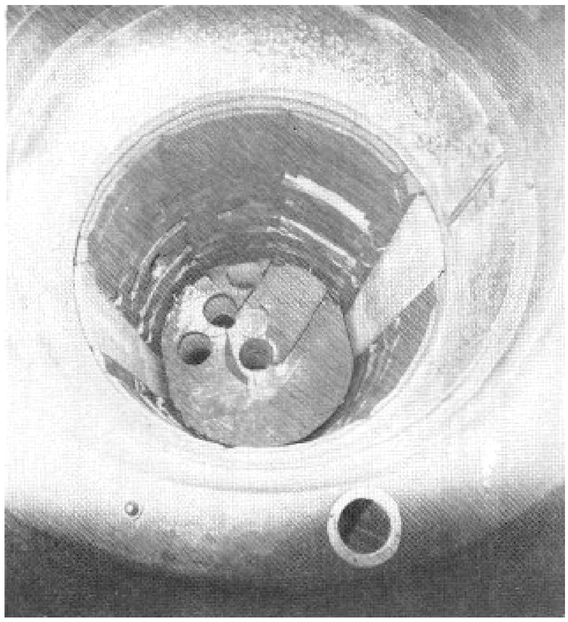
Fig 3-2
Reaktortank och grafitreflekto:
i biologiskt skydd

8.6.4 Supplement 4. 8-1. The distance working saw, the reactor vessel placed in the uranium well and the inside of the graphite reflector .



The distance working saw





The reactor vessel placed in the uranium well.

The inside of the graphite reflector .

8.6.5 Supplement 5. 8-4. Dose rate inside the biological shield without the reactor vessel.

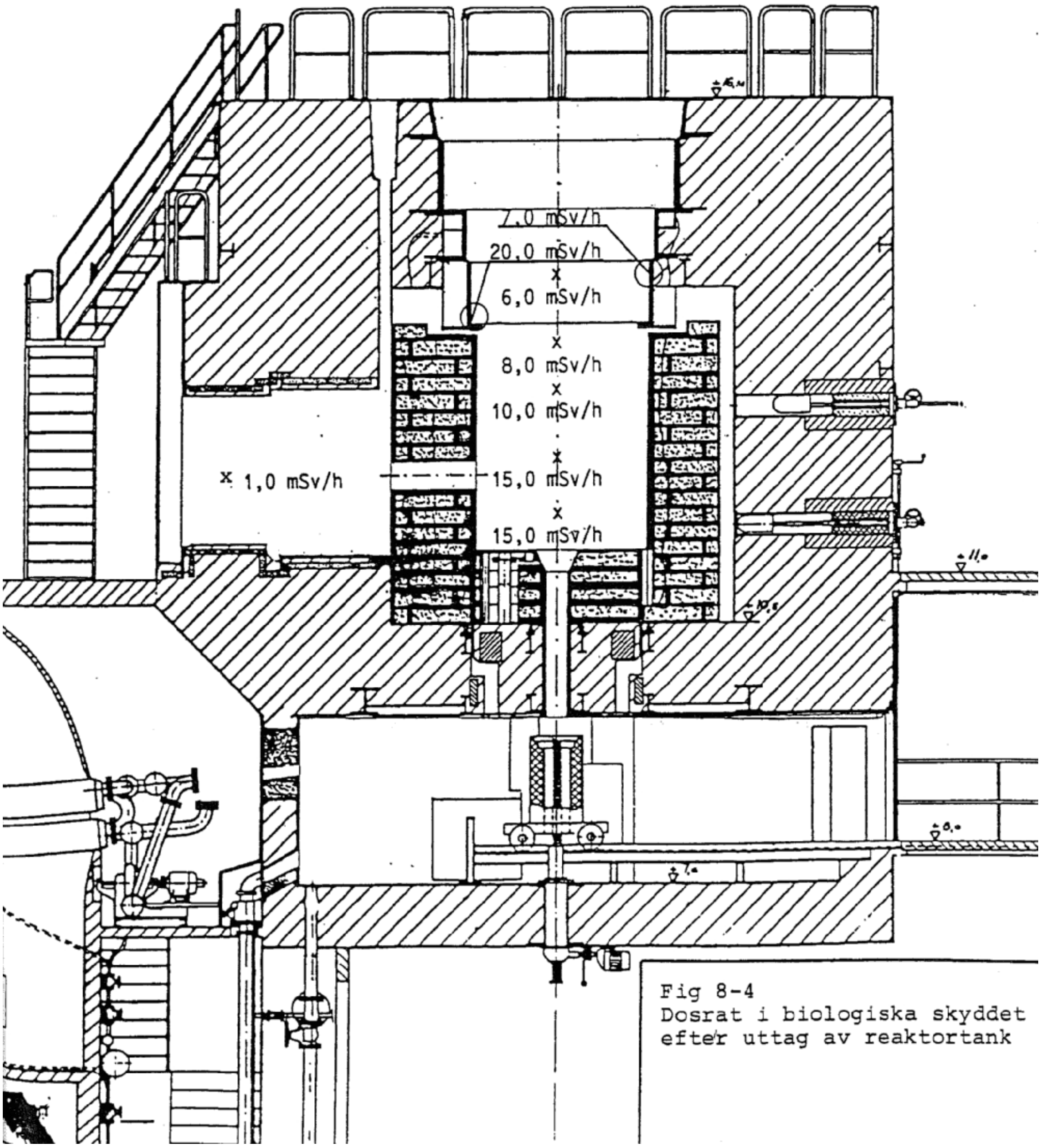
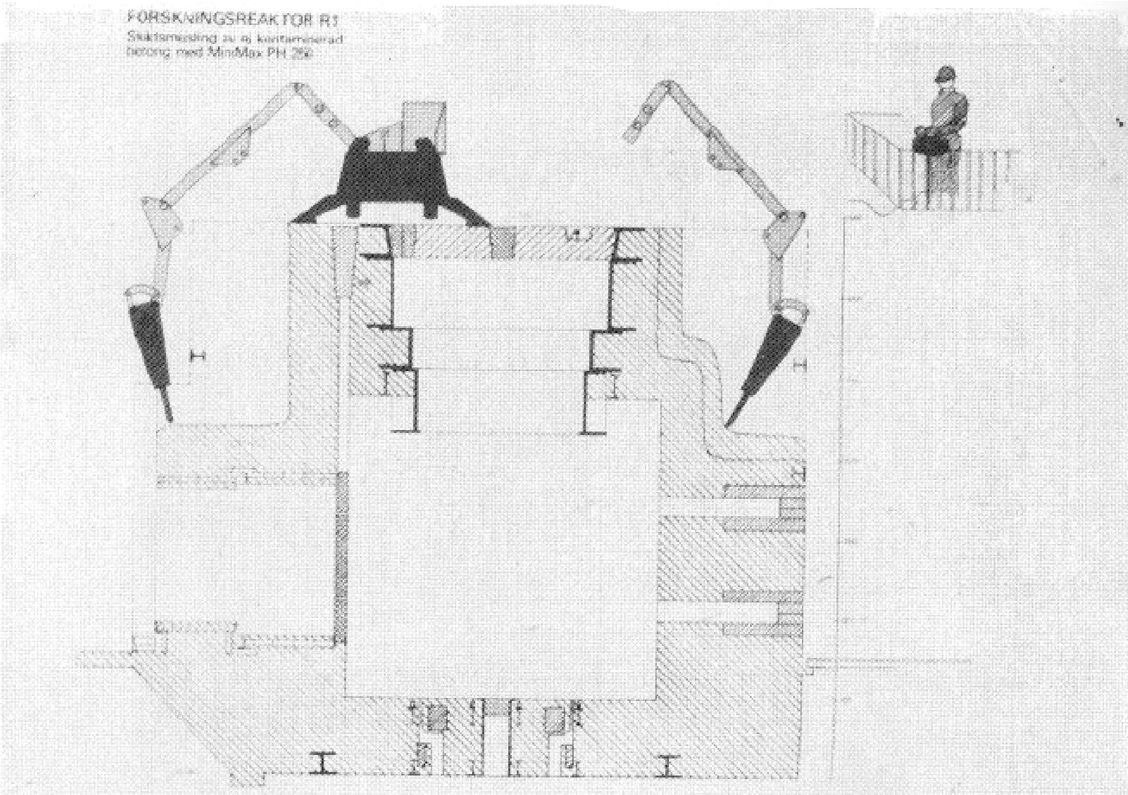
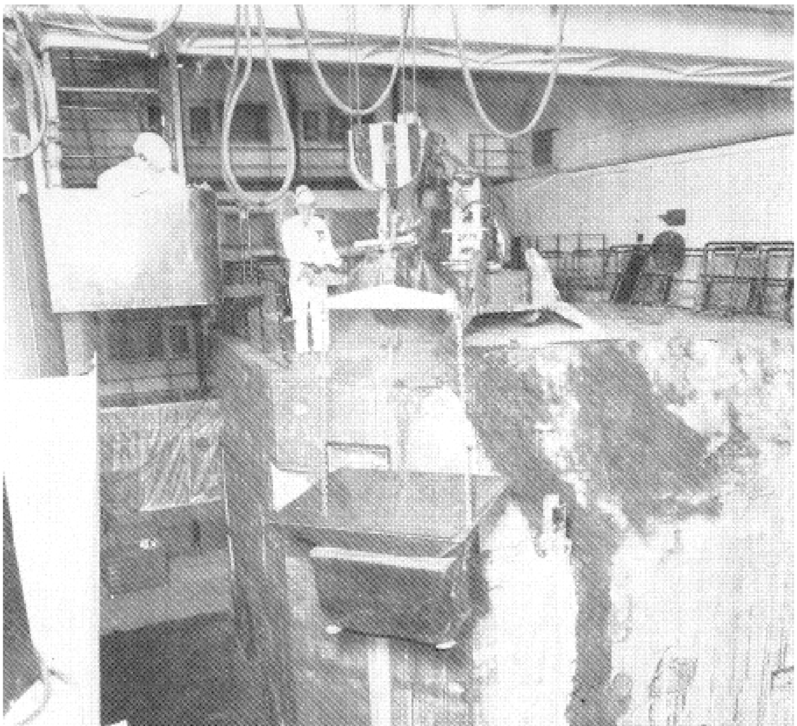


Fig 8-4
Dosrat i biologiska skyddet
efter uttag av reaktortank

8.6.6 Supplement. 8-8. MiniMax tears down the biological shield (upper image). 8-9.MiniMax tears down the concrete into a Berglöv box (lower image).





9 Concluding remarks

In the present report, the different sections illuminate the appropriateness of a historic and operations perspective, and the pertinence of suitable strategies with regard to radiological characterization, methodology selection, and financial risk analysis. Examples are provided from the four participating countries, Denmark, Finland, Norway and Sweden.

The project work has also included a number of plant and facility visits, exchange of information and networking. Such activities do not usually give rise to elaborate reports, and are not described here, but are essential for the decommissioning planning and the cost estimations.

The project work has confirmed the validity of the basic idea behind the project. Decommissioning planning, including financial prediction, is very complicated and especially small actors need co-operation and information exchange.

The character of the benefits may be somewhat different for different facilities. There are many owners of TRIGA reactors internationally and they operate a network of communication in a number of areas, including decommissioning. Thus decommissioning experience, including financial information, is being shared, see e g [81]. This is very helpful as compared to those who have more unique facilities. There do exist common design features in the old plants, but this does not generally include decommissioning aspects which do not seem to be even mentioned in the contemporary literature, see e g Reference [82].

The R1 decommissioning represents very early work on decommission and even more so on cost estimation. The present authors are very pleased that this project is now openly published for the first time. The agreement between prognosis and outcome can be compared with that of the DR1 and may actually be better than that of the recent decommissioning of the Active Central Laboratories at Studsvik.[83-84] Obviously, the careful preparation and planning had a great and positive impact on the precision.

The DR1 example represents a successful application of state of the art, and illustrates how a small organization in a small country can manage well by a combination of in house competence and some external information and support.

The uranium reprocessing plant represents a relatively facility. Its decommissioning illustrates the great value of access to the competence of the staff that operated the facility. It also illustrates the synergy between on one hand development work and documentation and on the other execution of the decommissioning work.

The work during the third year includes an international workshop during which the conclusions of the first two years – and as presented in the present report – will be discussed and challenged. The reason is not that the present authors feel insecure about the conclusions, but that the content of a document describing what is claimed to be “good practice” must be properly quality assured. As will be apparent from upcoming documentation, the findings in the present report have withstood this trial.

The same can be said about a presentation of the findings of the project to the ICEM’07.[85]

Most of the work during the third year comprises plant visits as well as information exchange, including compilation of documents into a searchable data base.

10 References

1 Decommissioning of nuclear power plants and research reactors. Safety Guide. IAEA safety standard series No WS-G.2.1.

2 Decommissioning of Nuclear fuel cycle facilities. Safety Guide. IAEA safety standard series No WS-G.2.4.

3 Decommissioning of medical, industrial and research facilities. Safety Guide. IAEA safety standard series No WS-G.2.2.

4 Predisposal management of radioactive waste including decommissioning. Safety Requirements. IAEA safety standard series No GS-R-2.

5 *Financial aspects of decommissioning. Report by an expert group.* IAEA-TECDOC-1476. November 2005.

6 Decommissioning Nuclear Power Plants: Policies, Strategies and Costs. Nuclear Energy Agency, Organisation for Economic Co-operation and development, OECD / NEA, 2003.

7 Lindskog et al. *Metodutveckling avseende kostnadskalkylering för programmet för avveckling och rivning, Rapport över analys i maj - augusti 2003 av framtida myndighetskostnader*. SKI Rapport 2003:31, september 2003.

8 Sjöblom R and Lindskog S. *An applied study of the storage for old intermediate level waste at the Studsvik site.* SKI Report 2004:11. The Swedish Nuclear Power Inspectorate, February 2004.

9 Sjöblom R, Sjöö C, Lindskog S and Cato A. Cost studies on decontamination and dismantling of nuclear facilities (in Swedish). The Swedish Nuclear Power Inspectorate, in print.

10 Sjöblom R, Sjöö C, Lindskog S and Cato A. *Early stage cost calculations for determination and decommissioning of nuclear research facilities.* The 10th International Conference on Environmental Remediation and Radioactive Waste Management. Glasgow, Scotland, 4-8 September, 2005.

11 Varley G and Rush C. *An applied study on the decontamination and decommissioning of the map tube facility 317 area Argonne National Laboratory, Chikago.* SKI Report 2005:34. The Swedish Nuclear Power Inspectorate, January 2005.

12 Statens strålskyddsinstituts föreskrifter (SSI FS 2002:4) om planering inför och under avveckling av kärntekniska anläggningar. (In Swedish; available at the SSI webb site in the form of an unofficial translation having the title *The Swedish Radiation Protection Authority's Regulations on Planning for and during Decommissioning of Nuclear Facilities*).

13 Efraimsson H och Lund I. *Föreskrifter om planering inför och under avveckling av kärntekniska anläggningar.* SSI Rapport 2003:15. September 2003. (In Swedish).

14 Lindell B. *Strålningens, radioaktivitetens och strålskyddets historia. Pandoras ask. Del 1. Tiden före andra världskriget.* Bokförlaget Atlantis AB, 1996. ISBN 91-7486-347-9. (In Swedish).

15 Lindell B. *Strålningens, radioaktivitetens och strålskyddets historia. Damokles svärd. Del 2. 1940-talet.* Bokförlaget Atlantis AB, 1999. ISBN 91-7486-779-2. (In Swedish).

16 Lindell B. *Strålningens, radioaktivitetens och strålskyddets historia. Herkules storverk. Del 3. Åren 1950-1966.* Bokförlaget Atlantis AB, 2003. ISBN 91-7486-744-x. (In Swedish).

17 Triay I. *The U.S. Department of Energy Environmental Management (EM) Program.* The 10th International Conference on Environmental Remediation and Radioactive Waste Management. Glasgow, Scotland, 4-8 September, 2005.

18 *Thorium fuel cycle — Potential benefits and challenges.* IAEA-TECDOC-1450. IAEA, May 2005.

19 Njølstad O. *Strålende forskning. Institutt for energiteknikk 1948-1998*. PDC Tangen, 1999. ISBN 82-518-3736-7. (In Norwegian).

20 Marcus F R. *Half a century of Nordic nuclear co-operation. An insider's recollections.* NKS secretariat, November 1997. ISBN 87-7893-018-9.

21 Larsson K-E. *Kärnreaktorn R1 – ett stycke högteknologisk pionjärhistoria.* Daedalus. Tekniska museets årsbok, årgång 50, 1981.

22 *Kärnkraftens historia.* Information hämtad från Statens Kärnkraftinspektions hemsida. (In Swedish). http://www.ski.se/dynamaster/file_archive/011017/8911207943/karnkraftenshistoria.pdf.

23 *Decommissionering in Denmark.* Danish Decommissioning. http://www.ddcom.dk/

24 Lauridser K et al. *Decommissioning of the nuclear facilities at Risø National Laboratory. Descriptions and cost assessment.* Risø National Laboratory, Report Risø-R-1250(EN), February, 2001.

25 VÄÄTÄINEN A E, MANNINEN J. *Funding of future dismantling and decommissioning costs in the Finnish State Nuclear Waste Management Fund. Safe Decommissioning for Nuclear Activities.* Proceedings of an International Conference Berlin, 14–18 October 2002, IAEA, Vienna (2003)

26 Kärnenergilag 990/1987, Kärnenergiförordning 161/1988 (www.finlex.fi)

27 Lundby J E. Private communication.

28 *Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities*. Technical Reports Series 351. IAEA, March 1993.

29 *First Annual Report 1951/1952*. Joint Establishment for Nuclear Energy Research JENER. Kjeller pr. Lillestrøm, Norway.

30 *Survey of activities 1948-1960, Kjeller and Halden – Norway*. Institutt for Atomenergi, Norway.

31 Lundby J E, *Decommissioning of a Uranium Reprocessing Pilot Plant – practical experiences*. Final Report of the Nordic Nuclear Research Project KAN-1.2. May 1994, TemaNord 1994:594.

32 *Safety Analysis report, Halden Boiling Water reactor. Part I Description, SAR I*, April 15. 2006. Institutt for Energiteknikk, Norway.

33 Nordlinger S och Hallberg B. *Radiologisk verksamhet i ACL*. Studsvik RadWaste, Rapport (ES-01/42), 2001. In Swedish.

34 Gustavsson L. *Avecklingsplan, R2 respektive R2-0 reaktorn med tillhörande byggnader och annex.* Studsvik Nuclear AB, Rapport N(R)-95/022, 1996. In Swedish.

35 Gustavsson L och Söderbäck B. *Alternativa avvecklingsstrategier för R-2anläggningen och Isotopcentralen.* Studsvik Nuclear AB, Rapport (N-05/35), 2005. In Swedish.

36 Ekenborg F och Tarkpea P. *Preliminär avvecklingsplan för Mellanlager för låg- och medelaktivt avfall-AM, Lagret för fast medelaktivt avfall-AT, Hall för aktivt avfall-AU, samt f.d. Uranförrådet-UF.* Studsvik Nuclear AB. Rapport Studsvik/S-05/12, 2005. In Swedish.

37 Gräslund C. *Preliminär avvecklingsplan för Förvaringsanläggningen för Bränsleelement i Studsvik.* Studsvik Nuclear AB, Rapport Studsvik/S-05/09, 2005. In Swedish.

38 Gräslund C. *Preliminär avvecklingsplan för Hot cell laboratoriet".* Studsvik Nuclear AB. Report Studsvik/N-05/284, 2005. In Swedish.

39 Hansson M och Tarkpea P. *Preliminär avvecklingsplan för Smältanläggningen-SMA.* Studsvik Nuclear AB, Rapport Studsvik/N-05/291, 2005.

40 Lund I. *Decommissioning of nuclear power plants; policies, strategies and costs.* Strategy Selection for the Decommissioning of Nuclear Facilities Seminar Proceedings. Tarragona, Spain, 1-4 September 2003. OECD/NEA 2004, No. 5300.

41 Safe, efficient, and cost-effective decommissioning. Working Party on Decommissioning and Dismantling (WPDD) Workshop in Rome, Italy, 6-10 September, 2004. OECD/NEA NEA/RWM/WPDD(2005)6.

42 LaGuardia T S. *Well-Founded Cost Estimation Validated by Experience.* Working Party on Decommissioning and Dismantling (WPDD) Workshop in Rome, Italy, 6-10 September, 2004. Proceedings on CD:s available at the OECD/NEA.

43 Pfeifer W, Gordelier S and Drake V. Cost Estimation Tools in Germany and the UK. Working Party on Decommissioning and Dismantling (WPDD) Workshop in Rome, Italy, 6-10 September, 2004. Proceedings on CD:s available at the OECD/NEA.

44 Walkden P and Taylor P. *Decommissioning in a deregulated market.* Nuclear Energy, volume 41, No 3, June 2002, 219-222.

45 Laraia M. *Technological maturity in nuclear decommissioning; where are we today?* The 10th International Conference on Environmental Remediation and Radioactive Waste Management. Glasgow, Scotland, 4-8 September, 2005.

46 Saaty T L and Vargas L G. *Decision making in economic, Political, Social and Technological environments with the analytic hierarchy process.* RWS Publications 1994. ISBN 0-9620317-7-1.

47 Saaty T L. *Decision making with dependence and feedback; the analytic network process.* RWS Publications 2001. ISBN 0-9620317-9-8.

48 Choppin G, Liljenzin J-O and Rydberg J. *Radiochemistry and nuclear chemistry*. Butterworth-Heinemann, 2002. ISBN 0-7506-7463-6.

49 *Guidelines for process safety in outsourced manufacturing operations.* Center for chemical process safety of the American institute of chemical engineers, 2000. ISBN 0-8169-0812-5.

50 Petersen D. *Analysing Safety System Effectiveness*, 3rd edition. Van Nostrand Reinhold, 1996. ISBN 0-442-02180-1.

51 *Guidelines for hazard evaluation procedures, second edition with worked examples.* Center for chemical process safety of the American institute of chemical engineers, 345 East 47th Street, New York. Published in 1992. ISBN 0-8169-0497-X.

52 Andersson J, Carlsson T, Eng T, Kautsky F, Söderman E and Wingefors S. *The joint SKI/SKB scenario development project.* SKI TR 89:14. Swedish Nuclear Power Inspectorate, 1989. Also published as SKB TR 89-35 (TR = Technical Report, SKB = Swedish Nuclear Fuel and Waste Management Company)).

53 SKI SITE-94. *Deep repository performance assessment project.* SKI TR 96:36. Swedish Nuclear Power Inspectorate, 1996.

54 *A proposed standardised list of items for costing purposes in the decommissioning of nuclear installations*. Interim Technical Document. OECD/NEA, 1999.

55 Peters M S and Timmerhaus K D. *Plant design and economics for chemical engineers, 4th edition.*[14] McGraw-Hill, Inc., New York 1991. ISBN 0-07-049613-7.

56 Rast R R. *Environmental remediation estimating methods*, 2nd edition. Reed construction Data, 2003. ISBN 0-87629-615-0.

57 Taboas A L, Moghissi A A and LaGuardia T S. *The decommissioning handbook*. ASME Press, 2004. ISBN 0-7918-0224-8.

58 Clark F D and Lorenzoni A B. *Applied cost engineering,* 3rd edition. Marcel Dekker, 1997. ISBN 0-8247-9800-7.

59 Navarrette P F and Cole W C. *Planning, estimating, and control of chemical construction projects*, 2nd edition. Marcel Dekker, 2001. ISBN 0-8247-0516-5.

60 Waier P R and Chiang J H, editors. *Unit price estimating methods*, 3rd edition. Reed construction Data, 2002. ISBN 0-87629-662-2.

61 Balboni B, Mewis R W, Nightingale P, Plotner S C and Waier P. *Means estimating handbook*, 2nd edition. Reed construction Data, 2003. ISBN 0-87629-699-1.

[14] A newer edition is: Peters M S, Timmerhous K D and West R E. Plant design and economics for chemical engineers, 5th edition. Mc Graw Hill, 2004. ISBN 007-124044-6.

62 Bagger Hansen M and Larsen K. *Decommissioning of the nuclear facilities at Risö National Laboratory in Denmark.* Safe decommissioning for nuclear activities. Proceedings of an international conference on safe decommissioning for nuclear activities organized by the International Atomic Energy Agency and hosted by the Government of Germany through Bundesamt für Stralenschutz and held in Berlin, 14-18 October 2002. International Atomic Energy Agency, Vienna, 2002.

63 *Decommissioning of DR 1. Final report.* Report DD-18(EN) rev.1. Danish Decommissioning, Roskilde, Denmark. January 2006.

64 Salmenhaara S E J. *Nuclear Waste Management Plan of the Finnish TRIGA Reactor.* Paper presented at the 2nd World Triga Users Conference 15-18 September, 2004, Vienna.

65 *Hazards study of the Chemical Reprocessing Plant at Kjeller.* The Chemistry Division, Institute of Atomic Energy, Kjeller, June 1960

66 Gaudernack B, Lindland K P and Joseph C J. *Operational experience from the Kjeller Reprocessing Pilot Plant.* Paper presented on the Third United Nations International Conference on the Peaceful Uses of Atomic Energy. Geneva, Switzerland, 31 August - 9 September 1964. A/Conf. 28/P/761.

67 Lundby J E, Frydenlund K. *The uranium reprocessing plant at Kjeller. History and first dismantling period.* 01.09.89. In Norwegian.

68 Lundby J E. *Experiences from dismantling of URA. Preliminary project report, part 1.* In Norwegian. The Nordic Research programme on Nuclear Safety 1990-1993 Project KAN-1.2. November 1991.

69 Lundby J E, Frydenlund K. *The uranium reprocessing plant at Kjeller. Second dismantling period 1989.* In Norwegian. IFE/I-90/014. December 1990.

70 Stedje E, Ulimoen K, Frydenlund K, Lundby J E. *Dismantling of the Uranium Reprocessing Pilot Plant at Institutt for energiteknikk, Kjeller. Radiation Protection Aspects.* Paper presented at the Nordic Society for Radiation Protection 9th regular meeting, Ronneby (Sweden), 29 - 31 August 1990.

71 *Safety in Decommissioning of Research Reactors.* IAEA safety series No. 74, IAEA, Vienna, 1986.

72 Lundby J E. *Conservation of information. The Uranium Reprocessing Plant at Kjeller.* In Norwegian. The Nordic Research programme on Nuclear Safety 1990 - 1993, Project KAN-1.3(91)4.

73 Whyte L L. *Decommissioning - 1988.* Proceedings First International Conference on Decommissioning Offshore, Onshore and Nuclear Works, held 22 - 24 March, 1988. University of Manchester Institute of Science and Technology. Page 285.

74 *Decommissioning of Radioactive Facilities.* Paper presented at a seminar organized by the Nuclear Energy Committee of the Power Industries Division of the Institution of Mechanical Engineers on 7. The Institution of Mechanical Engineers, London. November 1984. Page 27.

75 *Rules concerning requirements for use of open radioactive sources.* In Norwegian. Norwegian Radiation Protection Authority 1981.

76 Hannestad G. *Calibration of a small GM-detector for inside measuring of radioactivity in narrow steel pipes.* In Norwegian. Project KAN-1.2(93)3, Kjeller 28.07.1993.

77 *A depository for low- and intermediate level radioactive waste.* In Norwegian. A consequense evaluation.Statens bygg- og eiendomsdirektorat (now Statsbygg), 16 november 1992.

78 Andersson, Dahlgren L T, Devell L, Ervik S, Kärker S, Sundgren O, Widell C O. *Rivning av forskningsreaktorn R1 i Stockholm.* Studsvik Report, Studsvik/NW-84/627, 1985. In Swedish.

79 Ervik S, Kärker S, Menon S, Sundgren O, Widell C O. *Utredning om R1´s avveckling.* Studsvik, Arbetsrapport, K1-79/10, 1979. In Swedish.

80 Menon S, et al.1981. *Avveckling av R1-reaktorn. Förprojekt.* Studsvik Report. Studsvik/K1-80/01, 1980. In Swedish.

81 Meyer F and Steger F. Decommissioning of the ASTRA Re search Re actor – Dismantling of the biological shield. Nuclear Technology & Radiation Protection – 2/2006.

82 Munce J F. *The architect in the nuclear age. Design of buildings to house radioactivity.* Iliffe Books ltd, 1964.

83 Hedvall H R, Stridsman K H, Berg S R and Johnsson B. *Project evaluation of the decommissioning of a laboratory plant in Studsvik.* WM'06 Conference. February 26-March 2, 2006. Tucson, AZ, USA.

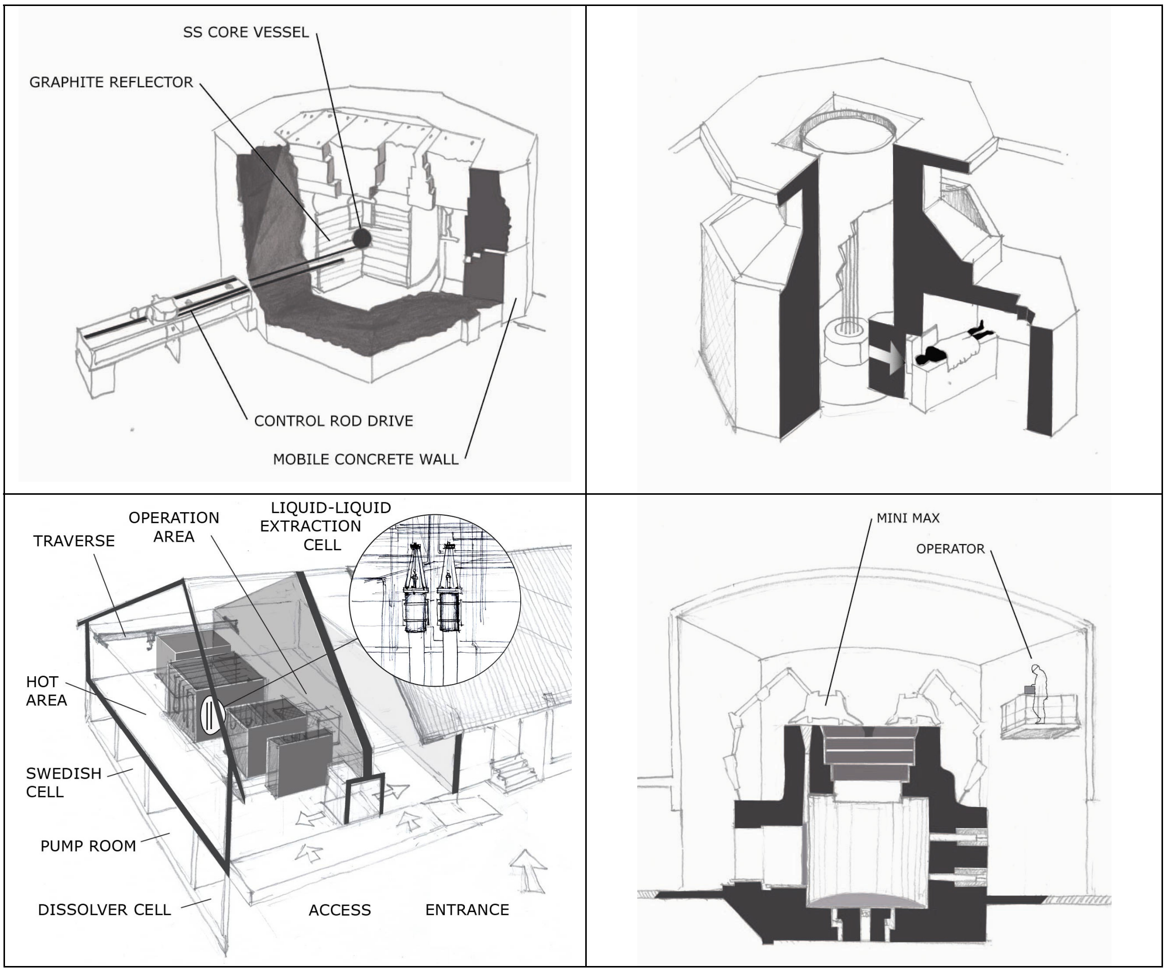
84 Johnsson B, Bergström L and Lindberg G M. *Decommissioning of the ACL and ACF plants in Studsvik, Sweden*. WM'04 Conference. February 29-March 4, 2004. Tucson, AZ, USA.

85 Iversen K, Salmenhaara S, Backe S, Cato A and Lindskog S, Callander C, Efraimsson H, Andersson I, Sjöblom R. *Cost calculations at early stages of nuclear facilities in the Nordic Countries.* The 11th International Conference on Environmental Remediation and Radioactive Waste Management. September 2-6, Bruges (Brugge), Belgium.

COST CALCULATIONS FOR DECOMMISSIONING AND DISMANTLING OF NUCLEAR RESEARCH FACILITIES

Volume II, Appendices

Inga Andersson, Studsvik Nuclear AB
Steinar Backe, Institute for Energy Technology
Anna Cato, Swedish Nuclear Power Inspectorate
Henrik Efraimsson, Swedish Radiation Protection Authority
Klaus Iversen, Danish Decommissioning
Staffan Lindskog, Swedish Nuclear Power Inspectorate
Seppo Salmenhaara, VTT Technical Research Centre of Finland
Rolf Sjöblom, Tekedo AB



May 2008

Content

Volume I

Volume II

APPENDICES

Appendix A

DECOMMISSIONING IN DENMARK

From
Danish Decommissioning

http://www.ddcom.dk/

:

Decommissioning in Denmark

1 Introduction

Risø National Laboratory (RNL) was established in the late 1950'es as a Danish research centre for preparing the introduction of nuclear energy in Denmark. Three research reactors and a number of supporting laboratories were built. However, Denmark has not yet built any nuclear power plants, and in 1980 the Danish Parliament decided that nuclear power should no longer be an option in the national energy planning. The facilities at RNL thus are the only nuclear facilities in Denmark. Subsequent to the Parliament's decision the research at RNL related to nuclear power was reduced and the utilisation of the facilities concentrated on other applications, such as basic materials research, isotope production and silicon transmutation doting. Already in 1975 one of the reactors had been taken out of service for economical reasons and the activities moved to the 10 MW materials test reactor, DR 3. Furthermore, in 1989 the hot cell facility was closed, and over the next four years it was partly decommissioned.

The Laboratory is located about 6 km north of the city of Roskilde (about 40 km west of Copenhagen) at the shore of Roskilde Fjord as shown in Figure 1.

Figure 1 Aerial photograph of Risø National Laboratory. Reactor DR 2 can be seen in the foreground. DR 3 is situated at the left hand side of the peninsula. DR 1 is hidden to the far right in the picture.

2 Planning for decommissioning

As part of Risø's strategic planning in 2000 it was taken into account that the largest research reactor, DR 3, was approaching the end of its useful life, and that the decommissioning question was becoming relevant. Since most of the other nuclear activities at Risø depended on DR 3 being in operation, it was decided to decommission all nuclear facilities at Risø National Laboratory once the reactor had been closed. Therefore, a project was started with the aim to produce a survey of the technical and economical aspects of the decommissioning of the nuclear facilities. The survey should cover the entire process from termination of operation to the establishment of a "green field"[1], giving an assessment of the manpower and economical resources necessary and an estimate of the amounts of radioactive waste that must be disposed of. The planning and cost assessment for a final repository for radioactive waste was not part of the project. Such a repository is considered a national question, because it will have to accommodate waste from other applications of radioactive isotopes, e.g. medical or industrial.

In September 2000 Risø's Board of governors decided that DR 3 should not be restarted after an extended outage. The outage was caused by the suspicion of a leak in the primary system of the reactor, and followed after the successful repair of a leak in a drainpipe earlier in the year. Extensive inspection of the reactor tank and primary system during the outage showed that there was not any leak, but at the same time some corrosion was revealed in the aluminium tank. According to the inspection consultant the corrosion called for a more frequent inspection of the tank. Therefore, the management judged that the costs of bringing the reactor back in operation and running it would outweigh the benefits from continued operation in the remaining few years of its expected lifetime.

The closure of DR 3, of course, accentuated the need for decommissioning planning and for the results of the above-mentioned project. By the end of February 2001 the project report [1] was published. The study was followed by other studies in order to prepare a proposal for legislative action by the parliament to provide funding for the decommissioning. Among other aspects, possible decommissioning strategies were evaluated. Two overall strategies were considered, (1) an irreversible entombment where the nuclear facility is covered by concrete and thereby transformed into a final repository for low- and medium level waste, and (2) decommissioning to 'green field' where all buildings, equipment and materials that cannot be decontaminated below established clearance levels are removed. The entombment option was rejected rather quickly as not being acceptable, among others for ethical reasons ("each generation should take care of its own waste"). Instead, three different decommissioning scenarios were considered with 'green field' as the end point, but with different durations, viz. 20, 35 and 50 years, respectively.

After thorough preparations, including an Environmental Impact Assessment, the Danish parliament in March 2003 gave its approval to funding the decommissioning of all nuclear facilities at Risø National Laboratory to "green field" within a period of time up to 20 years. The decommissioning is to be carried out by a new organisation, Danish Decommissioning (DD), which is independent of Risø National Laboratory, thus avoiding any competition for funding between the decommissioning and the continued research activities at Risø.

[1] In this context "green field" means a situation where facilities and areas are free released to other use without any radiological restrictions. Thus clean buildings and equipment may be re-used for other purposes than nuclear.

3 Description of the nuclear facilities

The nuclear facilities include three research reactors (DR 1, DR 2 and DR 3), a Hot Cell Facility and a Waste Management Plant with storage facilities. The activity content in each of the nuclear facilities has been estimated from both measurements and calculations and the results are shown in Table I with reference to the year 2000.

Table 1 Estimated activity in the nuclear facilities at Risø National Laboratory in 2000 [1].

Nuclear facility	β-/γ-activity [GBq]	α-activity [GBq]
Storage facility for high-radiation waste	700,000	30,000
Storage hall for waste drums	4,800	-
Waste Management Plant	8,500	10
Research reactor DR 3 (excl. fuel)	200,000	-
Hot Cell plant	3,000	100
Research reactor DR 1 (incl. fuel)	100	5
Research reactor DR 2	60	-
Cellar DR 2 (tritium in heavy water from DR 3)	3,000,000	-

Tritium in the heavy water from reactor DR 3 constitutes the largest single activity at the nuclear facilities as can be seen in Table 1, but it is, however, a very low-toxic radionuclide. The major potential radiological risks would arise during the decommissioning of reactor DR 3 and the Hot Cells, although the potentially largest doses could arise from exposure to waste in the storage facility for high-radiation waste. However, this waste is safely contained in steel drums and the probability for being exposed is, therefore, rather low.

The major characteristics of each of the nuclear facilities at Risø are briefly presented in the following paragraphs. A more detailed description of these facilities can be found in [1].

Research reactor DR 1

DR 1, shown in Figure 2, was a 2 kW thermal homogeneous solution type reactor, which used 20% enriched uranium fuel and light water as moderator. The first criticality was obtained in August 1957. During the first 10 years of operation the reactor was used for neutron experiments and thereafter mainly for educational purposes. In the autumn of 2000 it was decided to close the operation of the reactor, subsequent to the closure of DR 3.

The reactor core consists of a spherical steel vessel containing about 15 litres of uranyl sulphate dissolved in light water, which has now been drained. Around the core there is a graphite reflector contained in a steel tank and a biological shield made of heavy concrete. The reactor has various irradiation facilities. Two stainless steel control rods containing boron carbide controlled the reactor. In addition to these major reactor components there are connecting pipes, recombiner, lead shield, cooling coil etc.

The main part of the activity is concentrated in the fuel solution. During 43 years of operation it has consumed less than 1 gram of ^{235}U out of a total amount of 984 grams. The recombiner, the connecting pipes and the core tank are the most active components due to mainly ^{137}Cs deposited on the inner surfaces (and small amounts of actinides). Small amounts of long-lived activation products such as ^{14}C, ^{60}Co, ^{63}Ni, ^{133}Ba, ^{152}Eu and ^{154}Eu are left in the different construction parts, mainly in the core tank, the reflector tank and the concrete shield surrounding the graphite reflector.

Figure 2 DR 1

Research reactor DR 2

DR 2 was a pool-type, light water moderated and -cooled reactor with a thermal power level of 5 MW. The reactor went critical for the first time in December 1958. It was used mainly for isotope production and neutron scattering experiments. It was closed in October 1975 and partially decommissioned. After the final shut down, the spent fuel elements were shipped back to the US. The reactor block and the cooling system was sealed and the reactor hall was used for other purposes until 1997, when a pre-decommission study was commenced. During its 5905 days of operation the integrated thermal power of DR 2 was 7938 MWd. Figure 3 shows a cut-away drawing of DR 2.

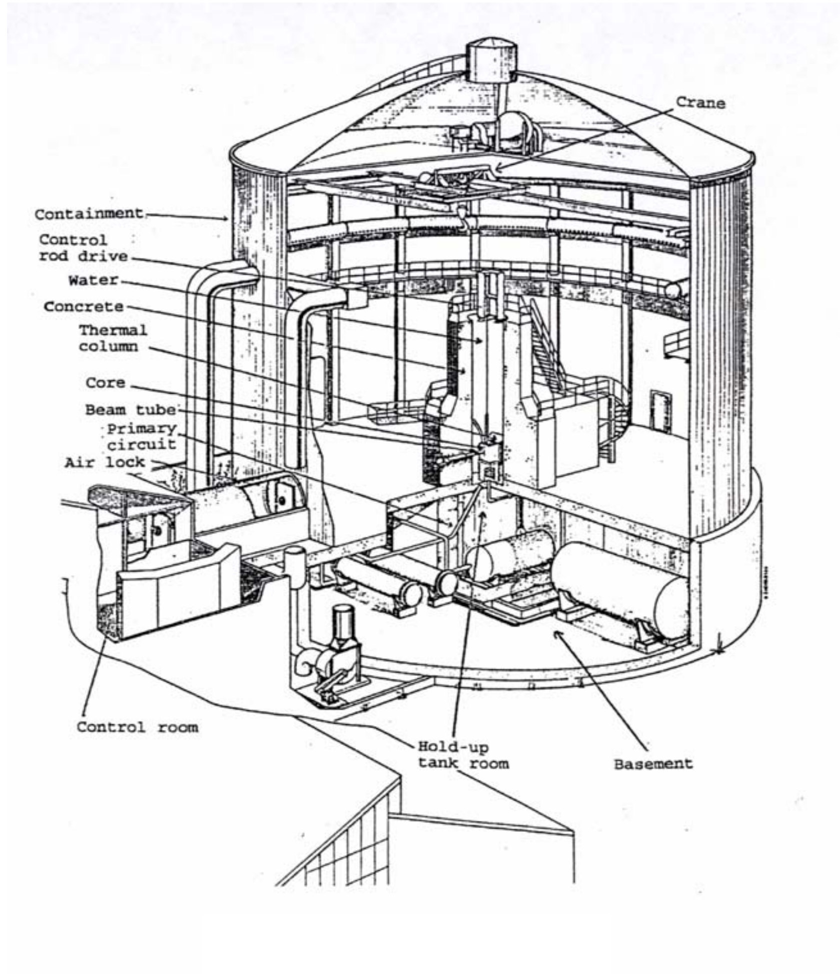


Figure 3 Cut-away drawing of DR 2

The reactor block is made of ordinary and heavy concrete and contains the reactor tank made of aluminium and a lead shield surrounding the core position. A graphite thermal column is situated next to the core position. The reactor tank is 8 metres in height and 2 metres in diameter. The primary cooling system including the heat exchangers is made of aluminium.

The major part of the residual activity in the reactor components is located in the stainless steel components and to some extent in the beam plugs and heavy concrete shield. The radionuclide activity is found in the following parts of the reactor system: reactor tank (^{60}Co), heavy concrete shield (^{133}Ba, $^{152+154}$Eu), beryllium reflector elements (^{10}Be), thermal column graphite ($^{152+154}$Eu, ^{14}C), beam plugs (^{60}Co), guide tubes and S-tubes (^{60}Co), and the primary cooling system (^{60}Co, ^{137}Cs).

Research reactor DR 3

DR 3 was a 10 MW tank type reactor with heavy water as moderator and coolant and a graphite reflector. It is of the DIDO/PLUTO family designed in the UK. The reactor went critical for the first time in January 1960 and since then was operated in a 4-week-cycle with 23 days of continuous operation and 5 days of shut down. It was finally shut down in September 2000 and had its last operation in April 2000. After the final shut down the fuel elements have been removed and shipped to the US and the heavy water has been stored in stainless steel drums (about 15,000 litres).

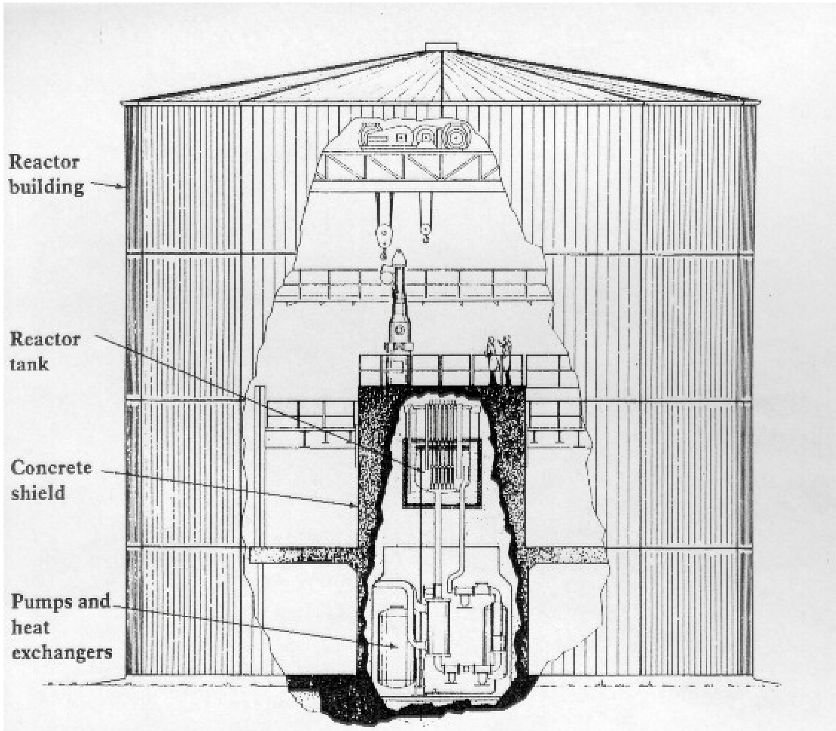


Figure 4 Cut-away drawing of DR 3

The reactor was used for materials testing, beam experiments, isotope production and silicon transmutation doting. The main reactor components are: reactor aluminium tank, primary cooling system (steel), graphite reflector, steel reflector tank, lead shield and biological shield (heavy concrete). The coarse control arms (cadmium contained in stainless steel) are stored outside the reactor in a storage facility for high radiation waste. The auxiliary systems are still in place, but are presently undergoing modification or being removed.

The major activity will be found in the following reactor components: reactor aluminium tank, graphite reflector, reactor steel tank, top shield, lead shield, biological shield, coarse control arms, irradiation rigs, and experimental facilities. The main components have a total weight of about 1000 tons and nearly all the residual activity will be found here, approximately 200 TBq of semi long-lived and long-lived radionuclides (year 2000). The tritium activity in the heavy water is about 3,000 TBq.

Hot Cell facility

The Hot Cell facility was commissioned in 1964 and operated until 1989. The six concrete cells have been used for post-irradiation examination of irradiated fuel of various kinds, including plutonium-enriched fuel pins. All kinds of non-destructive and destructive physical and chemical examinations have been performed. In addition, various sources for radiotherapy - mainly ^{60}Co - have been produced from pellets irradiated in DR 3. Following a partial decommissioning of the Hot Cell facility from 1990 to 1994 only a row of six concrete cells remains as a sarcophagus inside the building. The remaining part of the building has been released and is now being used for other purposes.

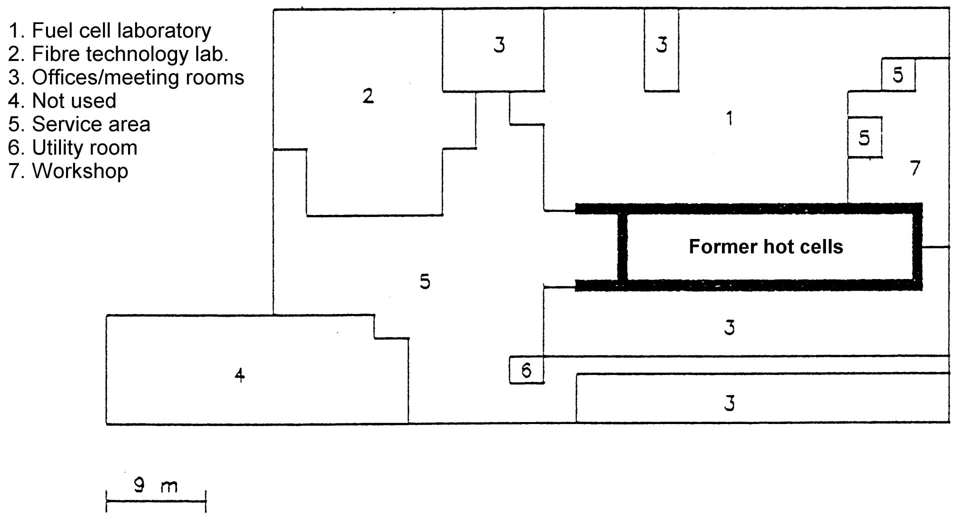


Figure 5 Sketch of the partly decommissioned Hot Cell facility

The dimensions of the interior of the six cells are: 39 metres in length, 4 metre in width and 5 meters in height. The cells are shielded by approximately 2 metres of concrete walls with lead glass windows. The cells are lined inside with steel plates and a conveyor belt, and parts of the ventilation systems still remain. Only long-lived fission products and actinides remain in the cells together with some small activated Co-pellets. Alpha- and gamma-spectrometric analyses of smear samples and dose rate measurements have shown that the major part of the activity, i.e. more than 90% is found in the concrete cells No 1 - 3. The total activity in the cells (1993) is about 3,000 GBq β-/γ-activity (mainly ^{137}Cs and ^{90}Sr) and about 100 GBq of actinides.

Waste Management Plant with storage facilities

The waste management plant is responsible for the collection, conditioning and storage of radioactive waste from the laboratories and the nuclear facilities at Risø as well as from other Danish users of radioactive materials. No final disposal of radioactive waste has taken place in Denmark and all waste units produced since 1960 are presently stored in three interim storage facilities at the Risø site.

The decommissioning of the Waste Management Plant will not take place until the decommissioning of the other nuclear facilities has been completed and a suitable substitute for the plant has been provided. After decommissioning of the nuclear facilities there will still be a need for a system for treatment of radioactive waste in Denmark, because radioactive isotopes will still be used in medicine, industry and research.

4 Considerations about re-use

Although the nuclear facilities are being decommissioned, Risø National Laboratory will continue to exist and carry out research within other areas of natural science [2]. It has, therefore, from the outset been the plan that those buildings and facilities that remain after decommissioning will be re-used by Risø National Laboratory itself. No concrete plans about the future use have yet been made. At this point in time it is not known for many buildings whether it is worthwhile to keep them or whether they should rather be demolished. For the DR 1 building, however, it is known that Risø wishes to keep it, because it offers good facilities for a "semi large" laboratory, e.g. a gantry crane. Another incentive to maintain the building is that it is situated in a location where it would be very difficult to get a permission to build a new building.

Some re-use has already taken place for the DR 2 building and the Hot Cell building, as described below.

DR 2 buildings re-use

After the closure and partial decommissioning of DR 2 the reactor building was clean enough to be used for other purposes. Thus, in the period 1979-95 it served for large scale chemical engineering experiments, in the first hand with a view to developing methods to extract uranium from Greenlandic ore. These experiments left the building in a less than clean condition; but in order to prepare for a characterisation project that started in 1997 the building was cleaned once again and surfaces were painted so that they are easy to decontaminate.

The control room, workshops and office building belonging to DR 2 have been used for other purposes all the time since the closure of the reactor in 1975.

Hot Cell building re-use

After the Hot Cell facility had been partly decommissioned in 1994 the radioactive parts remained as "a block of concrete" in the middle of the building. The remaining part of the building was refurbished and now serves as offices and laboratories for Risø's Materials Research Department, in particular for research in fuel cells and fibre reinforced materials.

The decision not to decommission the Hot Cells completely in 1990-94 was based partly on the philosophy to wait with producing large amounts of waste until the other facilities at Risø were to be decommissioned – and possibly hoping that a final repository for radioactive waste had been established by then. The latter is not yet the case, but the process for establishing a repository has been started. However, the fact that the Hot Cells are going to be decommissioned within a few years from now inevitably will present inconvenience to the activities that have been established in the building. Possibly the laboratories and offices will have to be evacuated during decommissioning, unless – as is aimed at – the cells can be decontaminated completely without demolishing the concrete walls.

5 References

1. Lauridsen, K. (Editor), Decommissioning of the nuclear facilities at Risø National Laboratory. Descriptions and cost assessment. Risø-R-1250(EN). ISBN 87-550-2844-6. Risø National Laboratory, February 2001. Available as a PDF-file at the Internet address: http://www.risoe.dk/rispubl/SYS/ris-r-1250.htm
2. http://www.risoe.dk

Appendix B

DECOMMISSIONING OF THE NUCLEAR FACILITIES AT THE RISØ NATIONAL LABORATORY IN DENMARK

by
M. BAGGER HANSEN
Risø Decommissioning, Risø, Roskilde
and
K. LARSEN
Danish Decommissioning, Roskilde

SAFE DECOMMISSIONING FOR NUCLEAR ACTIVITIES
PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
ON SAFE DECOMMISSIONING FOR NUCLEAR ACTIVITIES
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HOSTED BY THE GOVERNMENT OF GERMANY
THROUGH THE BUNDESAMT FÜR STRAHLENSCHUTZ
AND HELD IN
BERLIN, 14–18 OCTOBER 2002
INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2003

DECOMMISSIONING OF THE NUCLEAR FACILITIES AT THE RISØ NATIONAL LABORATORY IN DENMARK

M. BAGGER HANSEN
Risø Decommissioning,
Risø, Roskilde
E-mail: mogens.bagger.Hansen@risoe.dk

K. LARSEN
Danish Decommissioning,
Roskilde

Denmark

Abstract

After 40 years of nuclear research, Denmark has decided to close down all nuclear facilities except the Waste Management Plant at the Risø National Laboratory, namely the DR 1, DR 2 and DR 3 research reactors and the Hot Cells. At a later stage it will be decided to decommission these facilities of which the Waste Management Plant will be decommissioned last. The DR 2 reactor was closed in 1975, the Hot Cells in 1993 and the DR 1 and DR 3 reactors in 2000. The selection of an optimum decommissioning strategy depends on many factors, e.g. the national policy, the characteristics of the facilities, environmental protection, radioactive waste management, future use of the site, and the cost and availability of funds for decommissioning. Two overall strategies have been considered: (1) an irreversible entombment, where the nuclear facility is entombed in concrete and thereby transformed into a final repository for low and medium level waste, and (2) decommissioning to 'green field' condition, where all buildings, equipment and materials that cannot be decontaminated below established clearance levels are removed. Entombment has been rejected and three different decommissioning scenarios with green fields as the end point are being considered. The total duration of the scenarios is 20, 35 and 50 years, respectively. The paper describes the national policy on decommissioning and the organization responsible for the decommissioning is presented. The decommissioning scenarios are described with special emphasis on safety implications and costs. Management of the decommissioning waste and its characterization in terms of activity content are presented, including the construction of standard concrete containers and temporary storage facilities at the site. A large amount of inactive or very low active waste will be created during decommissioning, and clearance of this waste from regulatory control is discussed with regard to both methodology and clearance criteria. Finally, the impact of the decommissioning on the environment is briefly addressed.

1. INTRODUCTION

The Risø National Laboratory (Risø) was the creation of the famous Danish physicist Niels Bohr. He took intellectual responsibility for the introduction of experimental nuclear physics in Denmark and was the driving force in convincing the relevant Danish politicians to plan for the peaceful use of nuclear power as an important part of Danish energy production.

The aim of Risø when the first Danish reactor (DR 1) went critical in 1957 was to prepare — in the long term and through experimental work — a Danish nuclear power programme. That was the motive of Niels Bohr and of Danish Governments. The DR 1 research reactor was followed by the DR 2 (1958) and DR 3 (1960) research reactors and the Hot Cell plant (1964). Around these research facilities a national laboratory was constructed and developed. In the beginning, applications of nuclear technology created a joint strategic basis for all departments at Risø. In 1985, the nuclear option was removed from Danish energy planning. Risø was at that time by far the largest research facility in the country.

After the decision to close the nuclear facilities was made in 2000, energy production and distribution remained a general research theme at Risø, with wind energy as a good example. However, the research palette of today has plenty more colours than before, and the 'new Risø' no longer depends on the old nuclear facilities. A new strategy for future research has already been implemented.

In the light of this overall development Risø wants to dissociate itself from the past. Therefore — and in accordance with this desire — the Danish Government has decided to create a new State company, independent of Risø, with the plan to transfer the task to execute the decommissioning of all the nuclear facilities from Risø to this new company.

This paper is the first international presentation of the decommissioning strategy elaborated by the new company Danish Decommissioning.

2. DESCRIPTION OF THE NUCLEAR FACILITIES AT THE RISØ NATIONAL LABORATORY

The Risø National Laboratory is located about 6 km north of the city of Roskilde. At the site the nuclear facilities are situated close to Roskilde Fjord. The nuclear facilities include three research reactors (DR 1, DR 2 and DR 3), the Hot Cell facility and the Waste Management Plant with storage facilities. Their locations are indicated in Fig. 1. The DR 2 and DR 3 research reactors and the interior of the Hot Cell plant during the early days of its operation are

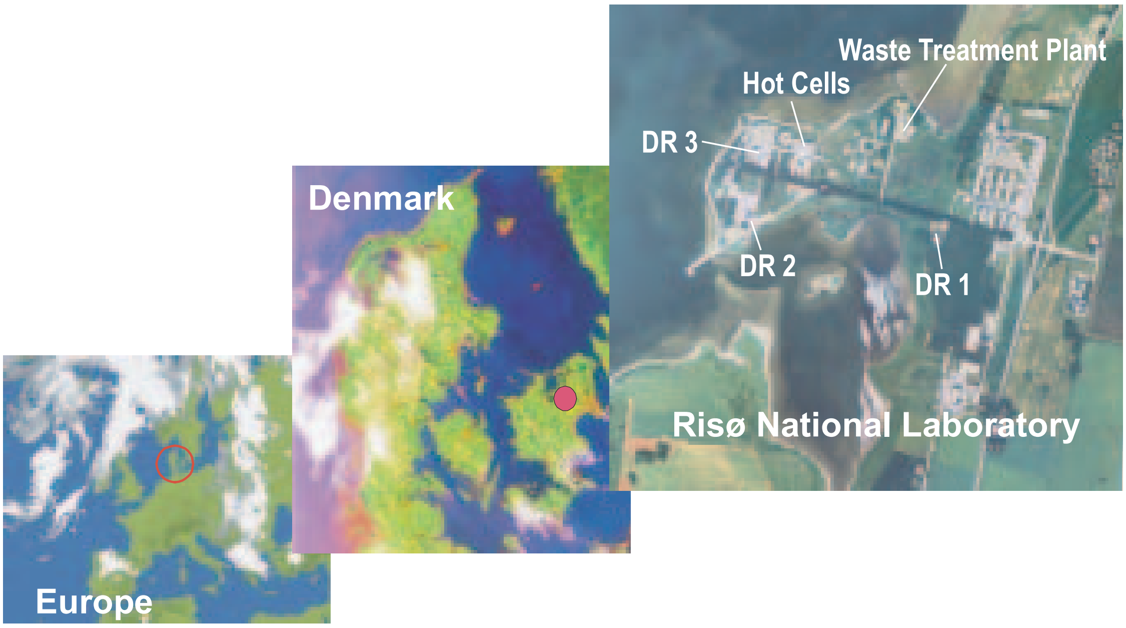


FIG. 1. Location of the Risø National Laboratory close to the city of Roskilde, some 40 km west of Copenhagen, and the location of the nuclear facilities on the Risø peninsula.

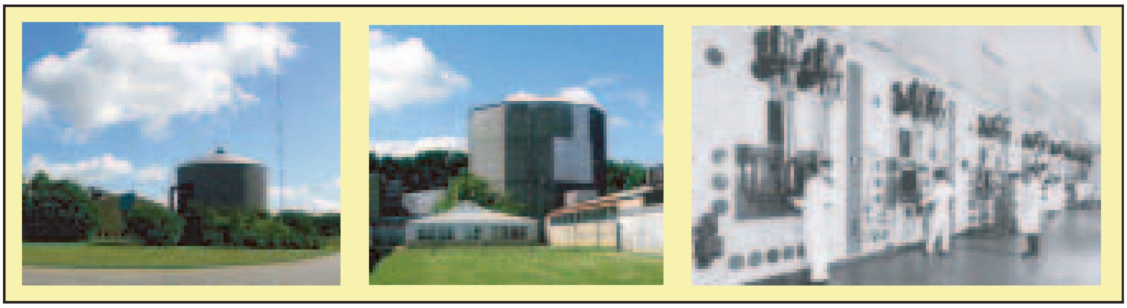


FIG. 2. From left to right, the DR 2 and DR 3 research reactors and the interior of the Hot Cell facility.

shown in Fig. 2. The activity content in each of the nuclear facilities has been estimated from both measurements and calculations and the results are shown in Table I with reference to the year 2000.

Tritium in the heavy water from the DR 3 reactor constitutes the largest single activity at the nuclear facilities, as can be seen in Table I, but it is, however, a very low toxicity radionuclide. The major potential radiological risks would arise during the decommissioning of the DR 3 reactor and the Hot Cell plant. Although the potentially largest doses could arise from exposure to waste in the storage facility for high radiation waste, this waste is safely contained in stainless steel containers and the probability of being exposed is therefore rather low.

TABLE I. ACTIVITY CONTENT IN THE NUCLEAR FACILITIES AT THE RISØ NATIONAL LABORATORY IN 2000 [1]

Nuclear facility	β/γ activity (GBq)	α activity (GBq)
Storage facility for high-radiation waste	700 000	30 000
Storage hall for waste drums	4 800	—
Waste Management Plant	8 500	10
Research reactor DR 3 (excluding fuel)	200 000	—
Hot Cell plant	3 000	100
Research reactor DR 1 (including fuel)	100	5
Research reactor DR 2	60	—
Cellar DR 2 (tritium in heavy water)	3 000 000	—

The major characteristics of each of the nuclear facilities at Risø are briefly presented in the following paragraphs. A more detailed description of these facilities can be found in a project report initiated by the Risø National Laboratory in June 2000. This report describes the nuclear facilities to be decommissioned and gives an assessment of the work to be done and the costs incurred [1].

2.1. DR 1 research reactor

DR 1 was a 2 kW thermal homogeneous solution type reactor, which used 20% enriched uranium fuel and light water as a moderator. First criticality was obtained on 15 August 1957. During the first ten years of operation the reactor was used for neutron experiments and thereafter mainly for educational purposes. In the autumn of 2000, it was decided to end the operation of the reactor.

The reactor core consists of a spherical steel vessel containing 13.4 L of uranyl sulphate dissolved in light water, which will be drained before decommissioning. Around the core there is a graphite reflector contained in a steel tank and a biological shield made of heavy concrete. The reactor is provided with various irradiation facilities. The reactor was controlled by two stainless steel control rods containing boron carbide. In addition to these major reactor components, there are connecting pipes, recombiner, lead shield, cooling coil, etc.

The main part of the activity is concentrated in the fuel solution. During 43 years of operation, it has only consumed about 1 g of ^{235}U out of a total amount of 984 g. When the core solution is removed, the recombiner, the connecting pipes and the core tank are the most active components due mainly to ^{137}Cs deposited on the inner surfaces (and small amounts of actinides). Small amounts of long lived activation products such as ^{14}C, ^{60}Co, ^{63}Ni, ^{133}Ba, ^{152}Eu and ^{154}Eu are left in the different construction parts, mainly in the core tank, the reflector tank and the concrete shield surrounding the graphite reflector.

2.2. DR 2 research reactor

DR 2 was a pool type, light water moderated and light water cooled reactor with a thermal power level of 5 MW. The reactor went critical for the first time on 19 December 1958. It has mainly been used for isotope production and neutron beam experiments. It was closed down on 31 October 1975 and partially decommissioned. After the final shutdown, the spent fuel elements were shipped back to the USA. The reactor block and the cooling system were sealed and the reactor hall was used for other purposes until 1997, when a pre-decommissioning study was commenced. DR 2 operated at full power from 1959. During its 5905 days of operation, the integrated thermal power was 7938 MW·d.

The reactor block is made of ordinary and heavy concrete and contains the reactor tank made of aluminium and a lead shield surrounding the core position. A shielded graphite column used for thermal neutron irradiation experiments is situated next to the core position. The reactor tank is 8 m in height and 2 m in diameter and has various beam and irradiation tubes. The primary cooling system, including the heat exchangers, is made of aluminium.

The major part of the residual activity in the reactor components is located in the stainless steel components and to some extent in the beam plugs and heavy concrete shield. The radionuclide activity is situated in the following parts of the reactor system: reactor tank (^{60}C), heavy concrete shield (^{133}Ba, $^{152+154}$Eu), beryllium reflector elements (^{10}Be), thermal column graphite ($^{152+154}$Eu, ^{14}C), beam plugs (^{60}Co), guide tubes and S tubes (^{60}Co), and the primary cooling system (^{60}Co, ^{137}Cs).

2.3. DR 3 research reactor

DR 3 was a 10 MW tank type reactor with heavy water as a moderator (and partly a reflector) and coolant. It was of the DIDO/PLUTO family constructed in the United Kingdom. DR 3 went critical for the first time on 16 January 1960 and has been operated since then on a four-week cycle, with 23 days of continuous operation and 5 days of shutdown. It was finally shut down in September 2000, its last period of operation ending in April 2000. After final shutdown, the fuel elements were removed and shipped to the USA and the heavy water (about 15 000 L) has been stored in stainless steel drums in the cellar of the DR 2 reactor.

The reactor has been used for materials testing, beam experiments, isotope production and silicon irradiation. The main reactor components are: reactor aluminium tank, primary cooling system (steel), graphite reflector, steel tank, lead shield and biological shield (heavy concrete). The coarse control arms (cadmium contained in stainless steel) are stored outside the reactor in the storage facility for high radiation waste. The auxiliary systems are still in place, but are presently undergoing modification or being removed. It is planned to use the active handling hall for decommissioning activities, including operations in the handling pond.

The major activity will be found in the following reactor components: reactor aluminium tank, graphite reflector, reactor steel tank, top shield, lead shield, biological shield, coarse control arms, irradiation rigs and thimbles, and experimental facilities. The main components have a total weight of about 1000 t and nearly all the residual activity will be found here, approximately 200 TBq of semi-long-lived and long lived radionuclides (year 2000). The tritium activity in the heavy water is about 3000 TBq. The residual activity in the reactor components has been estimated on the basis of calculations for the British DIDO reactor at Harwell, properly corrected for differences in reactor power and operating period.

2.4. Hot Cell facility

The Hot Cell facility was commissioned in 1964 and operated until 1989. The six concrete cells have been used for post-irradiation examination of irradiated fuel of various kinds, including plutonium enriched fuel pins. All kinds of non-destructive and destructive physical and chemical examinations have been performed. In addition, various sources for radiotherapy — mainly ^{60}Co — have been produced from irradiated pellets in DR 3. Following a partial decommissioning of the Hot Cell facility from 1990 to 1994, only the row of six concrete cells remains as a sarcophagus inside the building. The remaining part of the building has been released and is now used for other purposes.

The dimensions of the interior of the six cells are: 39 m in length, 4 m in width and 5 m in height. The cells are shielded by approximately 2 m of concrete walls with lead glass windows. The cells are lined inside with steel plates and a conveyor belt and parts of the ventilation systems still remain. Only long lived fission products and actinides remain in the cells, together with some small activated Co pellets. Alpha and gamma spectrometric analyses of smear samples and dose rate measurements have shown that the major part of the activity, i.e. more than 90%, is found in concrete cells 1–3. The total activity in the cells (1993) is about 3000 GBq b/g activity (mainly ^{137}Cs and ^{90}Sr) and about 100 GBq actinides.

2.5. Fuel Fabrication facility

The Fuel Fabrication facility has produced fuel elements for the DR 3 reactor for more than 35 years. Up to 1988, the fabrication was based on high enriched (93% ^{235}U) metallic uranium, but from then on the elements have been made from low enriched (<20% ^{235}U) U_3Si_2 powder. When all fuel material in the form of unused powder, fuel plates, samples, etc., has been transferred to the DR 3 storage room, the only activity left will be in the form of uranium contaminated equipment in the connected ventilation system and in the drain pipes in the building. It is expected that most of the contaminated equipment can rather easily be completely decontaminated.

2.6. Waste Management Plant with storage facilities

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. No final disposal of Danish produced radioactive waste has taken place and the entire collection of waste units produced since 1960 is currently stored in three interim storage facilities at the Risø site.

The decommissioning of the Waste Management Plant will have to be postponed until the decommissioning of the other nuclear facilities has been completed and suitable substitutes have been provided. After decommissioning of the nuclear facilities, there would still be a need for a system for the treatment of radioactive waste in Denmark, as radioactive isotopes will still be used in medicine, industry and research. The active part of the Waste Management Plant consists of the treatment plant for radioactive water (evaporation using steam recompression), decontamination room (mainly for protective clothing) and laboratories for control analyses and waste characterization.

The low active waste from the wastewater treatment plant is put in drums in a bituminization cell. The storage hall for low level waste drums contains about 4700 drums. The shielded storage facility for low and medium level waste contains about 80 drums of medium level waste. Each drum is a 100 L drum inside a 220 L drum with the annular space filled with cement mortar. The storage facility for high radiation waste consists of an underground concrete block with holes and pits for high radiation waste in stainless steel containers, e.g. control rods from DR 3 and α contaminated waste from the Hot Cell facility.

3. THE NATIONAL POLICY ON DECOMMISSIONING

The decision taken in September 2000 by the Risø Board of Governors to permanently close down the DR 3 research reactor, and the subsequent approval by the Minister responsible for science policy mark the starting point of the new State company Danish Decommissioning. No policy existed before September 2000 and no savings were made in the past for investments in decommissioning. Within a short period very fundamental decisions had to be taken. Firstly, it was decided to create the new organization with decommissioning as its one and only task and, secondly, it was decided to indemnify Risø for the loss.

The decision to establish a governmental organization responsible for the decommissioning was taken from the very beginning as part of the dialogue between Risø and the Ministry of Research and Information Technology. Seen from Risø's point of view, it was a matter of importance to avoid an image of 'decline and fall'. Therefore, the close-down was to be seen as a starting point for a new and offensive research strategy. The Ministry of Research and Information Technology, on the other hand, wanted to exclude any possible conflict of interest between the obligation to decommission and any future tasks.

Concerns about the impact of decommissioning upon the Risø economy became a matter of lengthy negotiations between the parties. The conclusion was an agreement with the Ministry of Finance that expenditures for decommissioning should not be a part of the Risø budget, and, consequently, there would be no connection between the financing of research and the financing of decommissioning.

The planning process for decommissioning the nuclear facilities is still evolving, which means that decommissioning of the nuclear facilities does not start from a master plan including all future steps to be taken and it most certainly does not indicate that all the pitfalls ahead are disclosed. They remain to be seen! But it does mean that a firm political decision is expected to be taken to go for complete decommissioning as fast as possible to arrive at green field status within the next 20 years. In addition, it has been decided to start — as soon as possible — a parallel process of establishing a radioactive waste disposal policy to avoid a conflict between decommissioning needs and the lack of radioactive waste storage facilities.

4. DECOMMISSIONING STRATEGIES

Many factors must be taken into account when selecting a strategy for decommissioning nuclear facilities. These include the national policy, characteristics of the facilities, health and safety, environmental protection, radioactive waste management, availability of staff, future use of the site, improvements in decommissioning technology, cost and availability of funds for the project and various social considerations. The relative importance of these factors must be assessed case by case. Three general types of strategy are normally considered:

— **DECON** (decontamination), where all components and structures that are radioactive are cleaned or dismantled, packaged and shipped to a waste disposal site, or are stored temporarily on-site. Once this task is completed and the regulatory body has terminated the license of the site owner, the site can be reused for other purposes.
— **SAFSTOR** (safe storage), where the nuclear facility is kept intact in protected storage for tens of years. 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 factor. When the activity has decayed to significantly lower levels, the unit is taken apart, similar to the DECON strategy.
— **ENTOMB** (entombment), where the radioactive structures, systems and components are entombed in a long lived substance, e.g. concrete. The entombed plant would be appropriately maintained, and be under surveillance until the activity has decayed to a level that permits termination of the plant's licence.

Three different decommissioning strategies for the nuclear facilities have been considered and some important issues that will influence the selection of the 'best' strategy have been identified:

— A prolonged cooling period (40–60 years) would not reduce the radioactive inventory in the DR 3 research reactor to a level where remotely operated tools could be avoided.
— Sufficient technology in the form of tools and knowledge is available at present for the decommissioning process.
— Concentrated planning and fast execution of the decommissioning process will give the maximum benefit from the existing staff, which possesses the relevant know-how on the existing installations and routines in handling radioactive materials and components.
— A short and continuous decommissioning process will establish the best opportunities for a rational use of the national resources, especially for Denmark with only one decommissioning project and no nuclear industry.

All estimates made so far also indicate that a continuous short decommissioning scenario is the most cost effective.

— To avoid delay in the decommissioning process awaiting planning, decision and completion of a final waste repository, a new temporary storage facility will be built at Risø to store the major part of the radioactive waste emerging from the decommissioning.

A safe storage strategy for some tens of years is considered to be inappropriate because the total costs would increase with increasing time. This is due to the fact that the costs of the actual dismantling of the facilities would remain more or less unchanged, but the surveillance costs would increase in proportion to the length of the storage period. Safe storage would also be in conflict with the well established view that problems should not be left for the coming generations to solve. The entombment strategy is considered to be quite unacceptable for several reasons, among them the very limited international experience. This strategy has been considered mostly due to a lack of facilities for the disposal of radioactive waste. It has therefore been suggested that complete decommissioning of all the nuclear facilities at Risø should be carried through to a green field status.

5. SCENARIOS AND METHODOLOGY FOR DECOMMISSIONING TO GREEN FIELD STATUS

Three different decommissioning scenarios to green field status have been considered for which the major difference is the cooling time for the DR 3 reactor from termination of operation to final dismantling. Cooling times of 10, 25 and 40 years have been considered. The total duration of the scenarios is estimated to be 20, 35 and 50 years, respectively, as indicated in (Fig. 3).

In all scenarios, it is assumed that the DR 1 and DR 2 reactors and the Hot Cells are decommissioned during the first ten years. The transfer of waste from the storage facilities at Risø to a final repository can more or less be carried out at any time after such a repository has been constructed.

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

Rough estimates have been made of the radiation doses to staff members during the decommissioning operations and are summarized in Table II for scenario 1. These estimates are rather uncertain, but better estimates require more

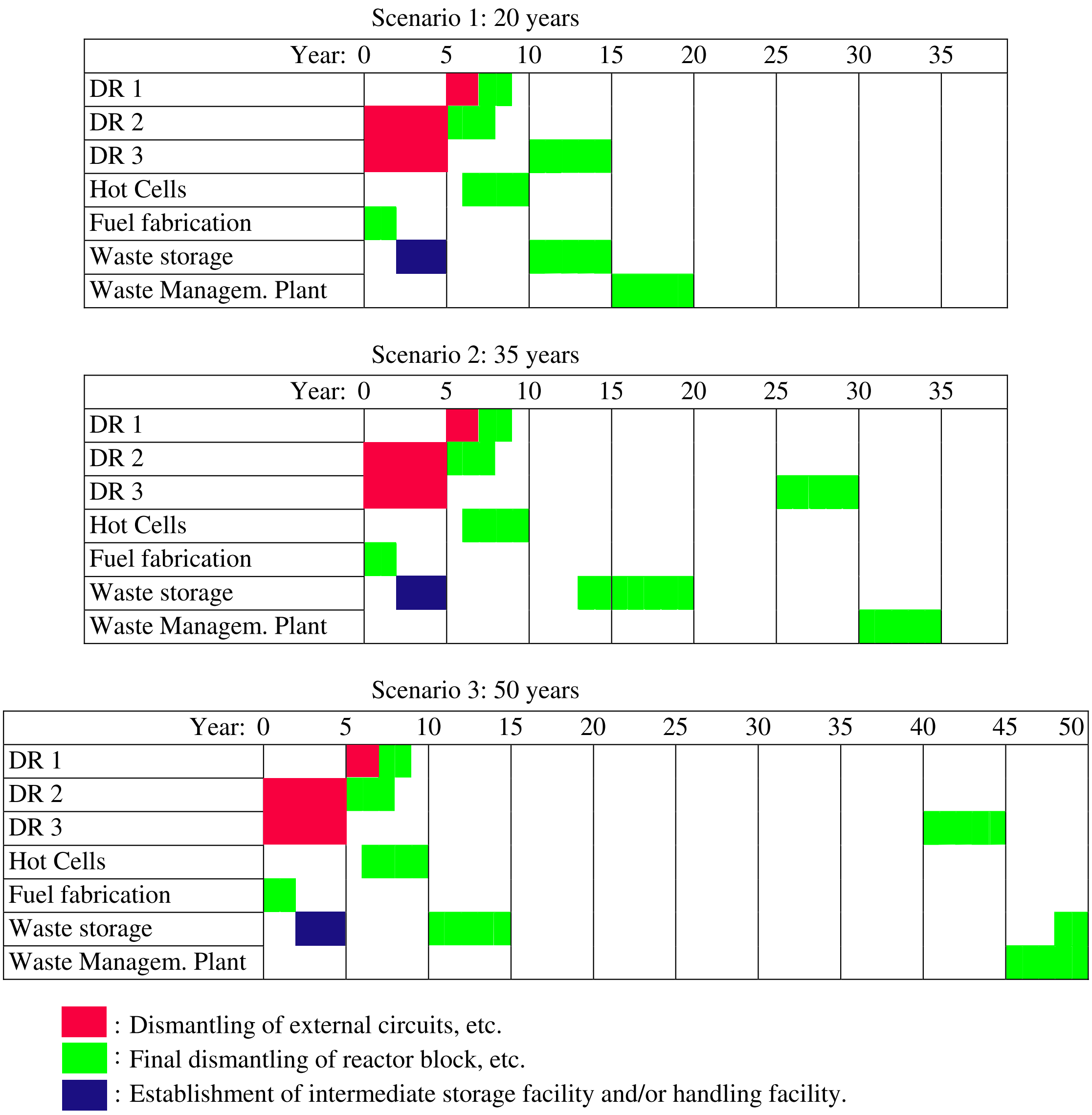


FIG. 3. Different decommissioning scenarios for Danish nuclear facilities leading to green field status.

precise assessment of the activity contents and the work operations to be performed.

'Hot operations' will, in all three scenarios, be performed by some kind of remote handling. The effect of radioactive decay on individual doses will 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 remotely, could be performed non-remotely due to the reduced activity content, the total collective dose might be higher for scenarios 2 and 3 compared with scenario 1.

TABLE II. RADIATION DOSES FROM DECOMMISSIONING OF RISØ'S NUCLEAR FACILITIES FOR SCENARIO 1
(for comparison, the collective doses registered at Risø during the later years have been ~150–200 man·mSv per year)

Nuclear facility	Estimated collective dose (man·mSv)
Reactor DR 1	25
Reactor DR 2	100
Reactor DR 3	2000
Hot Cells	300
Waste storage facilities	70
Total	~2500

There will probably not be large differences between the three scenarios with respect to the protective measures needed for the personnel carrying out decommissioning work. The costs for the three scenarios will therefore be equal in fixed prices, apart from the differences due to expenses for keeping the organization running for different periods of time and for keeping some facilities in safe storage for the longer scenarios. Total costs for the three scenarios have been estimated to be about €150 million, i.e. on average about €7–8 million per year during the periods where substantial work is being performed.

The shortest, 20 year, scenario is thus the most attractive and has therefore been recommended. This time-frame is dictated by two opposing points of view. On the one hand, a suitable cooling period for the DR 3 reactor, which was in operation until 2000, and on the other hand the best possible use of the expertise of the existing staff. The sequence for decommissioning the different facilities is dictated mainly by: (a) the activity content within the facility and the advantage of radioactive decay; and (b) the complexity of the facility. Consequently, the following sequence for decommissioning of the different nuclear facilities has been recommended:

(1) DR 1 research reactor,
(2) DR 2 research reactor,
(3) Hot Cell plant,
(4) DR 3 research reactor,
(5) Waste Management Plant with intermediate storage facilities.

The Waste Management Plant would be decommissioned at the end because operation of this facility is necessary during the decommissioning of all the other facilities.

Much of the construction materials in the nuclear facilities, e.g. the outer part of the reactor buildings and the auxiliary systems, will not be contaminated or will be only slightly contaminated. Such materials will as far as possible be sorted from the radioactive waste and removed for recycling, reuse or disposal as inactive waste. This will diminish the volume to be placed in the final disposal facility for radioactive waste. The non-active and slightly active waste will be checked for activity before and after the components have been dismantled. This, together with the origin and the known use of the components, will be used for primary sorting. A gamma scanning laboratory will be built for the final declassification measurements. The system and procedures will be quality controlled.

After completion of decommissioning, the site may need to be restored and cleaned of the remaining contamination. The selection of restoration techniques, which can be appropriately applied, will depend upon a number of factors. The major factors include: (1) the scale of the contamination problem and the radionuclides involved; (2) the contaminated medium; (3) the location of the contaminated site with respect to the local population; and (4) the location of the contaminated site with respect to a suitable waste repository for any residues. The need for restoration will be based upon a comprehensive radiological survey of the site and a dose constraint of 50 μSv/a to the critical group.

6. MANAGEMENT AND CHARACTERIZATION OF RADIOACTIVE WASTE

Low level waste (LLW) and intermediate level waste (ILW) from Danish users of radioactive materials and from operation of the three research reactors has in the last forty years been stored intermediately at Risø. Together with the waste emerging from the decommissioning of Risø's nuclear facilities, it will be transferred to a final repository to be built in the future somewhere in Denmark.

It would have been preferable if a Danish repository for low and medium level waste could have been available before initiation of a significant demolition of the more active parts of the nuclear facilities. However, the time schedule for availability of a final disposal facility is uncertain, and to be able to proceed with planning for the decommissioning the intention is to use interim storage also for the waste from the decommissioning work.

The decommissioning waste consists mainly of concrete, aluminium, ordinary steel, stainless steel and graphite. Estimates are given for expected volumes of conditioned waste from the decommissioning of the DR 1, DR 2 and DR 3 research reactors with associated buildings, the concrete cells in the Hot Cell plant, small facilities such as the Fuel Fabrication facility, and the Waste Management Plant with its storage facilities. They are shown in Table III, which also shows the approximate volume of the already existing waste in drums, etc., and as separate lines the remains from Uranium Pilot Plant (UPP) experiments with uranium extraction from ores from Greenland.

A new intermediate facility will be built at Risø for storage of the waste emerging from the decommissioning of the nuclear facilities. The facility will primarily be used for a new type of waste unit in the form of concrete containers. This waste unit will be used for decommissioning waste and also for some of the existing waste drums. The concrete containers will be designed with a multiple barrier system. It comprises backfill material, stainless steel membranes and high quality concrete. For ILW, internal shielding will be used if necessary. For very low level waste, International Organization for Standardization (ISO) containers or other containers made of steel can be used.

The concrete containers will be filled with waste at the decommissioning site. Afterwards they will be moved to the new temporary storage facility. The lids of the containers will not be sealed tightly before the remaining volume in the containers is filled up with backfill material and the final disposal facility is ready to receive the waste units. Depending on the waste types, cement or gravel will be used as backfill material.

Characterization of the activity content in the containers is important and required by the authorities. Samples from the decommissioning waste will be kept as documentation in a sample library and used for non-destructive and destructive measurements. The following analyses will be used for assessing the activity content in the radioactive waste:

— Calculation of β/γ activity concentrations from measurements of samples in the laboratory using a high efficiency germanium detector,
— Chemical determination of trace element concentrations in neutron activated waste for neutron activation calculations of radionuclide specific activity concentrations,
— Calculation of alpha activity concentrations from alpha spectrometric measurements of selected samples,
— Development of methodologies to determine ^{14}C and ^{3}H in reactor graphite and shielding concrete.

The requirements for the final disposal capacity have been determined to be between 3000 and 10 000 m^3. Probably, the facility will be a ‘near surface’

TABLE III. ESTIMATED AMOUNTS OF CONDITIONED RADIOACTIVE WASTE WITH INDICATIONS FOR CONTENTS OF SHORT AND LONG LIVED RADIONUCLIDES (EXCLUDING HIGH RADIATION WASTE AND 15 m^3 TRITIATED HEAVY WATER) IN 2010.

(The two figures in the right hand column are estimates for decommissioning waste that possibly might be released as non-active waste, and for inactive waste from the dismantling of buildings, etc.)

Nuclear facility	Volume of conditioned waste (m^3)	β/γ activity short lived (GBq)	b activity long lived $T_{1/2} > 30$ years (GBq)	a activity long lived actinides, etc. (GBq)	Mass of nearly inactive and inactive waste (t)
			Decommissioning waste		
DR 1	2	5	Low	Low	200 + 1000
DR 2	120	20	Low	≈ 0	300 + 600
DR 3 complex	1000	20 000[1]	7700[1]	≈ 0	1800 + 11 000
		20 000[2]	—		
Small facilities	6	—	Low	Low	+10
Hot Cells	50	3000	Low	100	2500
Waste Management Plant	50	1	Low	Low	100 + 3600
			Existing waste		
In drums, etc.	1800	25 000	1000	1000	
Total	3000	48 000[1]/20 000[2]	8700[1]	1100	5000 + 1600
UPP tailings	1000	Daughters	—	30 (NORM[3])	500
UPP ore	2400	Daughters	—	100 (NORM)	500

[1] The activities are based on assessments for the DIDO reactor at Harwell, UK.
[2] Tritium, mainly present in irradiated concrete shielding and generated by the $^6Li(n, \alpha)^3H$ process.
[3] NORM: naturally occurring radioactive material.

type, but the final concept has not yet been decided. The concrete containers will be constructed to withstand a certain degree of outer water pressure. Above a maximum water pressure the containers will quickly be filled with water if the facility is placed under the groundwater level. At present, construction of the final disposal facility and the process of site selection have not started.

For a final disposal facility placed outside Risø, the waste units are to be transported by road. If so, shielded transport containers will be used to comply with the guidance from the IAEA [2] and Danish regulations [3].

7. CLEARANCE OF NON-ACTIVE AND LOW ACTIVE WASTE

A large part of the waste from decommissioning will be a candidate for release as non-active waste, while a smaller part will require isolation in an appropriate radioactive waste facility.

Non-active waste can, without any restrictions, be deposited outside the Risø area as normal building or metal waste. It is, however, necessary to ensure that it contains sufficiently low activity levels so any form of post-release regulatory involvement is not required in order to verify that the public is being sufficiently protected. The point where there are no regulatory requirements has been defined as clearance, which is subject to *clearance levels being defined by six international organizations as values, established by the regulatory authority and expressed in terms of activity concentrations, at or below which sources of radiation may be released from regulatory control* [4].

Materials with activity content above clearance levels would be regarded as radioactive waste, whereas materials with activity levels at or below clearance levels would not be regarded as radioactive for regulatory purposes. In the European Union Council Directive on basic safety standards for radiation protection of the public, the disposal, recycling or reuse of materials containing radioactive substances may be released from the requirements of the directive provided they comply with clearance levels established by national competent authorities [5].

The European Union Article 31 Group of Experts has made recommendations on clearance levels for radionuclides in waste from the dismantling of nuclear installations [6]. These levels have been calculated from public exposure scenarios and a dose criterion of 10 μSv/a, corresponding to what has been defined as a trivial risk. Clearance levels for radionuclides that are expected during the decommissioning of the nuclear facilities at Risø are shown in Table IV.

The content of radionuclides in the candidate waste for release shall be documented to the regulatory authorities. A new low level laboratory with

TABLE IV. RECOMMENDED CLEARANCE LEVELS FROM THE EUROPEAN UNION [6]

	Clearance level (Bq/kg)
^{3}H	10^5
^{60}Co	10^2
^{63}Ni	10^6
^{90}Sr	10^3
^{137}Cs	10^3
^{238}U	10^3
^{239}Pu	10^2
^{241}Am	10^2

facilities to handle bulk quantities of waste and large items originating from the dismantling of the nuclear facilities will be built. The laboratory will be equipped with high efficiency germanium detectors, which will be calibrated using a sophisticated point source/volume source technique, enabling inhomogeneous activity distributions in bulky items to be determined by gamma spectroscopy analyses. In addition, analyses will be made for the content of α emitters and pure β emitters. Procedures and methods will be quality assured in accordance with existing ISO standards.

8. IMPACT OF DECOMMISSIONING ON THE ENVIRONMENT

Plans for the decommissioning of the nuclear facilities at Risø will include radiation protection of the surrounding population in the same way as during the operating phase of the facilities. Procedures will therefore be established to limit potential releases of radioactive materials to the environment during dismantling of the facilities. Existing environmental surveillance programmes will be continued or even expanded to include analyses, for example ^{14}C releases to the environment. Emergency preparedness plans to mitigate any consequences of accidental releases of radioactive materials to the environment will be continued, although at a lower level than during the operational phase.

Assessments of potential doses to the surrounding population from atmospheric releases of radioactive materials during decommissioning, both from normal operation and from accidents, require analyses that would be extremely costly. An alternative and deterministic approach has been used relating a fractional release of the activity inventory from each nuclear plant to individual radiation doses to members of the critical group in the surrounding population. With this approach it is possible to determine the maximum doses to the critical group corresponding to an (almost impossible) 100% release of the inventory, either continuously during decommissioning or over a short time period during an accident [7].

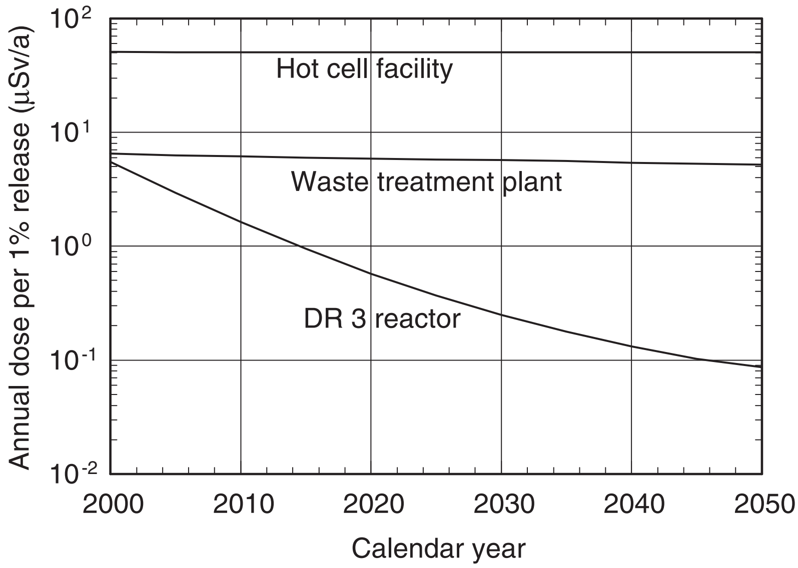
The calculated individual doses to the critical group outside the Risø area situated at a distance of 1 km from the nuclear facilities are shown in Fig. 4, both for an annual atmospheric release rate of 1% of the inventory and for an accidental atmospheric release of 1% of the inventory over a short time period. Atmospheric releases from the DR 1 and DR 2 reactors are not included in Fig. 4, as the activity content in these facilities is very low. Individual doses from aquatic releases to Roskilde Fjord will be insignificant.

It appears from Fig. 4 that the individual doses from a 1% release rate from the DR 3 reactor would decrease with time due to radioactive decay. Doses from any future releases from the Hot Cell facility and the Waste Management Plant would remain unchanged, as they would be dominated by long lived actinides.

A fractional release of 1% of the activity inventory is extremely conservative, at least for the DR 3 reactor, as the radioactive materials are distributed as activation products within the inner parts of the construction (reactor tank, top shield, etc.). For the Hot Cell facility the activity is distributed on the inner surfaces of the concrete cells as small particles and a fractional release of 1% of the activity during dismantling would be more likely, but still rather conservative. Even if a large fraction of the activity inventories were released to the atmosphere, the maximum individual doses to the critical group would be comparable to and no more than a few times the annual doses from the natural background radiation.

9. SUMMARY

All the nuclear facilities at the Risø National Laboratory except the Waste Management Plant have been closed and the plan is to decommission these facilities, including the Waste Management Plant, to green field status within the next 15–20 years. The total costs are estimated to be around €150 million, corresponding to an average annual cost of about €7–8 million for the short scenario over 15–20 years. The dominant contributor to the total decommissioning costs is the DR 3 research reactor. The costs will not be evenly distributed over the period, and investment costs for building facilities, for example remote handling and decontamination, will add to the basic costs.



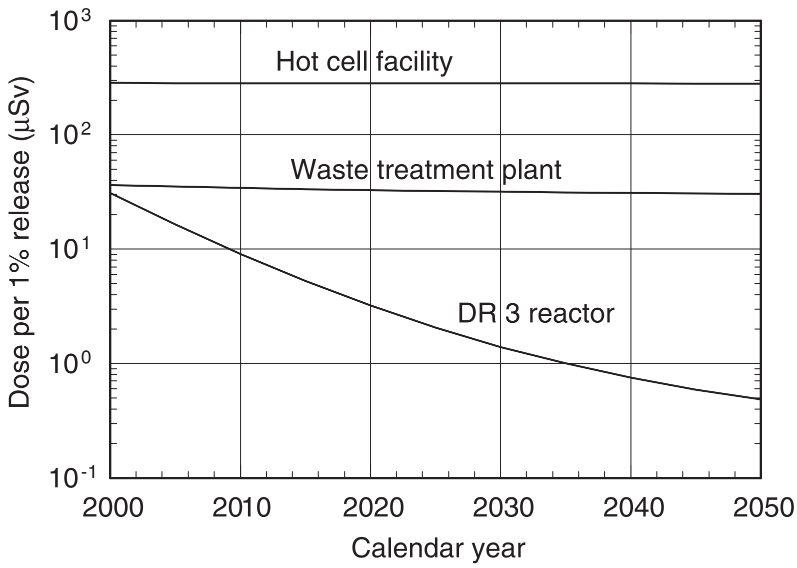


FIG. 4. Individual doses to members of the critical group from an annual release of 1% of the activity inventory (upper figure) and from an accidental release of 1% of the activity inventory in the Hot Cell facility, the DR 3 reactor and the Waste Management Plant (excluding the storage facility for high radiation waste) over a short time period under the most probable meteorological conditions (lower figure) [7].

A few alternative options to fast decommissioning to green field status have been considered. These include safe storage, where the nuclear plant is kept intact and placed in protective storage for several tens of years, and entombment, where the radioactive structures, systems and components are encased in a long lived substance such as concrete. The latter is equivalent to establishing an on-site shallow land burial waste disposal facility. It is very unlikely that any of these alternative options will be selected.

Storage and disposal facilities are needed for about 5000 m^3 of conditioned radioactive waste, including existing waste and waste produced during decommissioning. The existing storage facilities for radioactive waste are more or less filled and it is therefore planned to build a new temporary storage facility for the decommissioning waste packed into a new type of concrete waste unit. This storage facility will be used only for a relatively small number of years, with subsequent transfer of the waste units to a final repository once such a facility has been constructed.

Decommissioning of the nuclear facilities is not expected to cause any significant releases of radioactive materials to the environment but should such releases occur, only small doses comparable to doses from the naturally occurring background radiation would be the result.

REFERENCES

[1] Decommissioning of the Nuclear Facilities at Risø National Laboratory — Descriptions and Cost Assessment (LAURIDSEN, K., Ed.), Rep. Risø-R-1250(EN), Risø National Laboratory, Roskilde (2001).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material — 1996 Edition (Revised), Safety Standards Series No. TS-R-1 (ST-1, Revised), IAEA, Vienna (2000).

[3] NATIONAL INSTITUTE OF RADIATION HYGIENE, Regulation for Transport of Radioactive Substances, Reg. No. 993, Copenhagen (2001).

[4] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

[5] COMMISSION OF THE EUROPEAN COMMUNITIES, Council Directive 96/29/Euratom, 13 May 1996, Basic Safety Standards for the Protection of the Health of Workers and the General Public against the Dangers arising from Ionizing Radiation, Official Journal of the European Communities, L 159, Vol. 39, CEC, Brussels (1996).

[6] EUROPEAN COMMISSION, Recommended Radiological Protection Criteria for the Clearance of Buildings and Building Rubble from the Dismantling of Nuclear Installations, Radiation Protection 113, EC, Brussels (2000).

[7] HEDEMANN JENSEN, P., Differences in Environmental Impacts for Different Decommissioning Scenarios for the Nuclear Facilities at the Risø Site, National Laboratory, Roskilde (2002) (in Danish).

NUCLEAR WASTE MANAGEMENT PLAN OF THE FINNISH TRIGA REACTOR

S. E. J. Salmenhaara

VTT Processes
Technical Research Centre of Finland (VTT), Otakaari 3 A,
P.O. Box 1608, FI-02044 VTT (Espoo), Finland
seppo.salmenhaara@vtt.fi

Abstract

The FiR 1 –reactor, a 250 kW Triga reactor, has been in operation since 1962. The main purpose to run the reactor is now the Boron Neutron Capture Therapy (BNCT). The BNCT work dominates the current utilization of the reactor. The weekly schedule allows still one or two days for other purposes such as isotope production and neutron activation analysis.

According to the Finnish legislation the research reactor must have a nuclear waste management plan. The plan describes the methods, the schedule and the cost estimate of the whole decommissioning waste and spent fuel management procedure starting from the removal of the spent fuel, the dismantling of the reactor and ending to the final disposal of the nuclear wastes. The cost estimate of the nuclear waste management plan has to be updated annually and every fifth year the plan will be updated completely. According to the current operating license of our reactor we have to achieve a binding agreement, in 2005 at the latest, between our Research Centre and the domestic nuclear power companies about the possibility to use the Olkiluoto final disposal facility for our spent fuel. There is also the possibility to make the agreement with USDOE about the return of our spent fuel back to USA. If we want, however, to continue the reactor operation beyond the year 2006, the domestic final disposal is the only possibility.

In Finland the producer of nuclear waste is fully responsible for its nuclear waste management. The financial provisions for all nuclear waste management have been arranged through the State Nuclear Waste Management Fund. The main objective of the system is that at any time there shall be sufficient funds available to take care of the nuclear waste management measures caused by the waste produced up to that time. The system is applied also to the government institutions like FiR 1 research reactor.

1. Introduction

The FiR 1 reactor, a 250 kW Triga reactor, has been in operation since 1962. The main purpose to run the reactor has been lately the Boron Neutron Capture Therapy (BNCT). The epithermal neutrons (0.5 eV – 10 keV) needed for the irradiation of brain tumor patients are produced from the fast fission neutrons by a moderator block consisting of Al+AlF_3 (FLUENTAL™) developed and produced by VTT. The material gives excellent beam values both in intensity and quality and enables the use of a small research reactor as a neutron source for BNCT purposes [1]. Over thirty patients have been treated since May 1999, when the license for patient treatment was granted to the responsible BNCT treatment organization [2]. The treatment organization has a close connection to the Helsinki University Central Hospital. The funding of the BNCT-project is coming from a public funding organisation. The goal of the funding of the BNCT project is to develop the treatment organisation to a profit-making company. VTT as the reactor operator has a long term contract with the treatment organisation to produce epithermal neutrons for the patient treatments.

The BNCT work dominates the current utilization of the reactor: three or four days per week for BNCT purposes and the rest for other purposes such as the neutron activation analysis and isotope production. Figure 1 describes the general layout of the BNCT facility at the FiR 1 reactor. The facility gives a high epithermal neutron field, 1.1×10^{9} n/cm^{2}s with a very low fast neutron and gamma component.

During this and the next year (2004 and 2005) the back end solutions of the spent fuel management will have a very important role in our activities and in the possibility to continue the operation of the reactor. According to our current operating license we have to achieve next year (2005) a binding agreement between VTT and the domestic nuclear power plant companies about the possibility to use the final disposal facility of the nuclear power plants for our spent fuel. In this case we can continue the operation of the reactor as long as there is reasonable work to do and the funding is in order. Naturally we can also make an agreement with the USDOE within the well-known time limits.

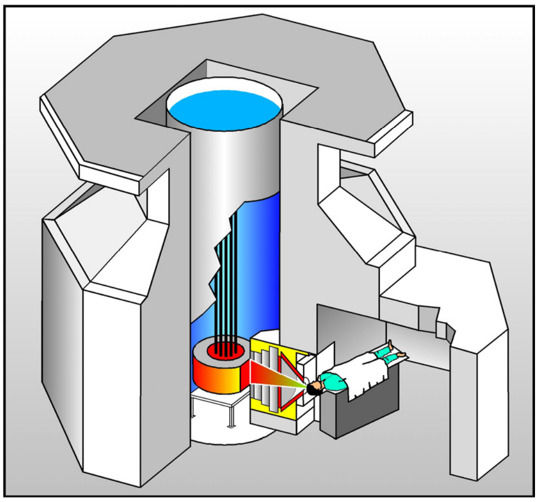


Fig 1. BNCT Facility at the FiR 1 –reactor

2. Final disposal solution of spent fuel in Finland

The Finnish nuclear power companies founded in 1995 a separate company Posiva to develop the technology and carry out safety analysis and site investigations for implementing the spent fuel final disposal. In 1999 Posiva submitted an application for a decision in principle for a final repository to be built at Olkiluoto, on the western coast of Finland. Olkiluoto is also one of the two nuclear power plant sites in Finland. At the end of the year 2000 the Finnish government approved the application and sent it to the parliament for ratification. The ratification took place in May 2001. Separate licenses still will be needed for the construction of the facility, scheduled to start in 2010, and also for the operation, 10 years later. The government alone will grant these licenses and no political aspects are supposed to involve in the licenses.

For the final repository the spent fuel will be encapsulated in airtight copper canisters and situated in the bedrock at a depth of 500 m. The safety of this deep underground repository is based on multiple natural and engineered barriers. Each canister contains 12 normal fuel assemblies from nuclear power plants. The present concept for Triga fuel elements is that the elements will be loaded in containers, which have the same outer dimensions as the nuclear power plant fuel assemblies. This ensures that the Triga fuel will be easily handled in the final disposal facility and loaded in the heavy copper canisters. Figure 2 describes the final disposal canisters.

Posiva Oy

Posiva Oy

Fig 2. Final disposal canisters

3. Nuclear waste management and spent fuel situation at the FiR 1 reactor

In Finland also the research reactor must have a nuclear waste management plan, which contains among others a part for spent fuel management. The plan describes the methods, the schedule and the cost estimate of the whole spent fuel management procedure starting from the removal of the fuel from the reactor core and ending to the final disposal. The cost estimate of the nuclear waste management plan has to be updated annually and every fifth year the plan will be updated completely. The plan has been based on the assumption that the final disposal site will be somewhere in Finland. Now we know that the final disposal facility for the spent fuel of the nuclear power plants will be situated in Olkiluoto. The final disposal facility is supposed to be in operation in 2020.

In Finland the producer of nuclear waste is fully responsible for its nuclear waste management. The financial provisions for all nuclear waste management have been arranged through the State Nuclear Waste Management Fund. The cost estimate of the nuclear waste management will be sent annually to the authorities for approval. Based on the approved cost estimate the authorities are able to determine the assessed liability and the fees to be paid to the Fund [3]. The main objective of the system is that at any time there shall be sufficient funds available to take care of the nuclear waste management measures caused by the waste produced up to that time. The system is applied also to the government institutions as FiR 1 research reactor operated by the VTT.

We have had already for fourteen years an agreement in principle about the possibility to use the final disposal facility of one of the Finnish nuclear power companies. Later this agreement was transferred to the joint nuclear waste management company Posiva. According to the current operation license of our reactor we have to achieve a binding agreement between our Research Centre and either Posiva or USDOE about the back end solution of the spent fuel. This means that the said agreement in principle is not sufficient any more. The binding agreement with Posiva is the only alternative, when we want to continue the reactor operation beyond the year 2006. Obviously the idea is that the binding agreement has to be established during the time when there are still two possible agreement partners left. Before we can start the real negotiations about the final disposal of our spent fuel with Posiva, we have to prepare a safety study about the behaviour of the Triga fuel in the final disposal surroundings.

The current operation license of our reactor will expire in 2011. It is possible to apply a new license at that time. In every case it is very probable that there will be certain waiting time from the shut down of the reactor to the opening of the final disposal facility. Therefore there have to be a sufficient interim storage for the spent fuel before the transportation to the final disposal facility. After enlargement work of the spent fuel storage in 1997 we have sufficiently storage capacity for the fuel in the reactor building. So far we have used it as dry storage. In addition to the domestic final disposal solution there is still the USDOE alternative available until 2006.

4. Safety of the Triga fuel in the final disposal repository

For later negotiations aiming to the binding agreement we are making safety studies about the long term behaviour of the spent TRIGA fuel in the final disposal surroundings. The main safety aspects, which have to be analyzed and compared to the spent fuel coming from nuclear power plants, are the criticality safety, the solubility of the fuel ($UZrH_x$) in water and the existence of some moving and long-lived radioactive isotopes. The TRIGA fuel is much more reactive compared to the spent fuel coming from nuclear power plants and therefore the TRIGA fuel can not be situated so tightly in the final disposal canister. The Triga containers will be situated in the outer zone of the canister and the inner zone will be left empty. In practice the empty positions will be loaded with dummy assemblies made of cast iron. The criticality safety calculations show, however, that it is possible to load safely all the TRIGA fuel elements in one final disposal canister. This is important, because if the criticality safety would demand the fuel to be divided to two or more canisters, the expenses would also be about twice or more compared to the one canister alternative.

5. Final disposal of the decommissioning waste

The nuclear waste management plan is based on immediate dismantlement after the final shutdown of the reactor. Experienced personnel will be still available to conduct the decommissioning work. The decommissioning waste is supposed to be disposed of in the repository constructed in the bedrock of the Loviisa nuclear power plant site at the depth of 110 m. At the moment preparatory work has been done to clarify the possible problems of the decommissioning waste of our reactor in the surroundings of decommissioning waste of the nuclear power plant. Our goal is to work out an agreement between VTT and the Loviisa NPP about the final disposal of our decommissioning waste in the said repository.

The decommissioning waste studies concentrate mainly on the long term safety of the decommissioning waste disposal. The main part of the active reactor components will be packed in concrete packages in the waste disposal facility, which means an additional barrier against the ground water flow. Among others the amount and behaviour of some long-lived radioactive isotopes like ^{14}C belong to these studies. Triga reactors have typically plenty of irradiated graphite in many components.

6. Conclusions

At the moment, when the BNCT and other irradiations develop satisfactorily and the funding of the reactor is in order, the primary alternative for the spent fuel management is naturally the domestic one. It is, however, reasonable to keep so far both of the possibilities still open: the domestic final disposal and the return to the USA offered by USDOE. The cost estimates of the both possibilities are on the same order of magnitude. At the end of this year (2004) we will be ready to have an opinion about the future of the BNCT and the reactor. Consequently we will be able to decide, which of the spent fuel policies will be obeyed. Meanwhile the necessary safety assessment concerning the behaviour of the spent fuel in the final disposal surroundings will be completed and based on the safety assessment the draft of the binding agreement will be written.

References

[1] AUTERINEN I., SALMENHAARA S.E.J. FiR 1 reactor in service for boron neutron capture therapy (BNCT) and isotope production. Proceedings of an International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, 10-14 November 2003, Santiago, Chile (in press)

[2] JOENSUU, H., KANKAANRANTA, L., SEPPÄLÄ, T., et al., Boron neutron capture therapy of brain tumors, Clinical trials at the Finnish facility using boronophenylalanine. Journal of Neuro-Oncology, **62** (2003) 123-34

[3] VÄÄTÄINEN A.E., MANNINEN J. Funding of future dismantling and decommissioning costs in the Finnish State Nuclear Waste Management Fund. Safe Decommissioning for Nuclear Activities. Proceedings of an International Conference Berlin, 14–18 October 2002, IAEA, Vienna (2003)

This paper was presented at the **2nd World Triga Users Conference 15. - 18.9.2004, Vienna**

PS

At the end of the year 2004 USDOE extended the acceptance policy of spent research reactor fuel by ten years until 2016

Appendix D

FUNDING OF FUTURE DISMANTLING AND DECOMMISSIONING COSTS IN THE FINNISH NUCLEAR WASTE MANAGEMENT FUND

by
A E VÄÄTÄINEN
and
J MANNINEN
Finnish Ministry of Trade and Industry

SAFE DECOMMISSIONING FOR NUCLEAR ACTIVITIES
PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
ON SAFE DECOMMISSIONING FOR NUCLEAR ACTIVITIES
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HOSTED BY THE GOVERNMENT OF GERMANY
THROUGH THE BUNDESAMT FÜR STRAHLENSCHUTZ
AND HELD IN
BERLIN, 14–18 OCTOBER 2002
INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2003

:

FUNDING OF FUTURE DISMANTLING AND DECOMMISSIONING COSTS IN THE FINNISH STATE NUCLEAR WASTE MANAGEMENT FUND

A.E. VÄÄTÄINEN, J. MANNINEN
Ministry of Trade and Industry,
Helsinki, Finland
E-mail: anne.vaatainen@ktm.fi

Abstract

The financial provisions for all nuclear waste management, including dismantling and decommissioning (D&D), in Finland have been arranged through the State Nuclear Waste Management Fund, which was founded in 1988. A producer of nuclear waste is fully responsible for its nuclear waste management, including D&D. The main objectives of the system, created through the legislation, are: (a) at any time there shall be sufficient funds available to take care of the nuclear waste management measures caused by the waste produced up to that time; and (b) the financial burden caused by the production of wastes shall, in a timely manner, be reflected in the cost of electricity produced through the activity giving rise to those wastes. The part of liability that is not covered by money in the Fund must always be fully guaranteed. The State Nuclear Waste Management Fund is a special purpose fund, segregated from the State budget. The licence holders are entitled to borrow back 75% of the capital of the Fund against the provision of full guarantees and at current interest rates. In addition, the State has the right to borrow the rest of the capital. Plans and cost estimates for the remaining nuclear waste management measures are updated yearly by the nuclear power companies and approved by the authorities. The assessed liability and fees to be paid into the Fund by the companies are then confirmed. No discounting is used. The funding system in Finland seems to work well and so far no serious problems have arisen as regards the future availability of sufficient capital for nuclear power plant D&D.

1. GENERAL FRAMEWORK

In Finland presently some 27% of all electricity is produced by nuclear power. The total capacity of the four nuclear power units, situated at two different sites, is 2656 MW. Teollisuuden Voima Oy (TVO) operates the Olkiluoto power plant with two 840 MW(e) BWR units supplied by Asea-Atom and commissioned in 1979 and 1982. Fortum Power and Heat Oy (the former IVO) operates two 488 MW(e) Russian type PWR units commissioned in 1977 and 1981 at the Loviisa site. In addition, there is one small research reactor.

This statistical information already reveals two factors that have had a decisive influence on the system through which funds are collected for dismantling and decommissioning (D&D) of nuclear facilities in Finland. The first of these factors is that the Atomic Energy Act promulgated in 1957, i.e. ten years before the order for the first nuclear power plant unit was placed, declared that any company or organization that met the requirements set out in the legislation was eligible to produce nuclear energy. In other words, production of nuclear energy was not to be a State monopoly. Electricity production in general has never been a State monopoly in Finland. This starting point, considering nuclear energy production as a commercial activity, has also been maintained in the Nuclear Energy Act, replacing the old Atomic Energy Act in 1988.

The other factor is the relatively small size of the Finnish 'nuclear plant fleet'. This has indirectly influenced the strategies for nuclear waste management and decommissioning. At an early stage it became obvious that the reprocessing of spent fuel in Finland was not, in practice, an option. Furthermore, in spite of the small scale of the Finnish nuclear programme, there seemed to be no guarantee of finding suitable foreign reprocessing or disposal services for all the spent fuel generated in Finland. Both the high prices of these services and the non-proliferation aspects were seen as potential obstacles. Thus, decommissioning was not seen as the only financial liability of the nuclear facilities. It was found to be quite possible that in the future, after the nuclear power plants were closed down, a significant task of disposal of spent nuclear fuel would still have to be carried out. Consequently, decommissioning was seen to be only a part of the major question of nuclear waste management and not a separate undertaking.

Finland is one of the countries that consider nuclear power to be a viable option for electricity production. This was recently demonstrated by the Finnish Parliament when it gave, by ratifying a so-called 'decision-in-principle', political acceptance for the construction of a new nuclear power plant unit in Finland. The operator of this new unit will be TVO. As for the existing nuclear power plant units, their planned lifetimes are at least 40 years. The current operating licences are in force until the end of 2007 (for Loviisa) and 2018 (for Olkiluoto). This means that there will probably be nuclear power plants in operation in Finland for a long time. On the other hand, there is still no decommissioned nuclear facility in Finland. And the experimental uranium mining effort did not really take off.

2. BASIC PRINCIPLES OF FINANCIAL PROVISIONS FOR THE COSTS OF NUCLEAR WASTE MANAGEMENT

The old Atomic Energy Act included only very general provisions on nuclear waste management, since waste management was not considered a significant issue in the 1950s. Fortunately, the Act gave extensive powers to the authorities to draw up licence conditions on arrangements for nuclear waste management and decommissioning, and on collecting reserves to cover the respective costs and include these conditions in the operating licences of the nuclear facilities. In that connection, it was, however, seen that a stronger legal basis for provisions for the costs of nuclear waste management was needed. This was one of the important reasons to start, at the end of the 1970s, the drafting of new nuclear legislation. However, due to both substantial disagreements and legislative problems, the new act, the Nuclear Energy Act, did not enter into force until 1988.

When drafting the legislation for financial provisions for the costs of nuclear waste management in Finland the following two, now almost globally accepted, principles were chosen as starting points:

- The costs of management of any quantity of nuclear waste should be reflected in the cost of the nuclear electricity production giving rise to those wastes (timeliness);
- The funds collected should be available when waste management operations are carried out and they should be sufficient for that purpose.

In the Finnish solution, the manner of implementing the principle of availability and sufficiency strongly influenced the manner of implementing the timeliness principle.

From the political point of view, the administration of the funds to be collected was an important question. Two views were competing: on one side those who, at least partly for ideological reasons, saw that the funds should be administered by the State, and on the other side those who considered that the State was the most unreliable trustee of the capital. Several alternative funding methods were studied. For example:

- Internal funding of nuclear companies;
- Internal funding of nuclear companies plus full guarantees to be furnished to the State;
- Internal funding of nuclear companies, plus a bank deposit on a blocked account in the Bank of Finland;
- External funding without the right of borrowing back;

— External funding with the right of borrowing back with or without the obligation to provide guarantees;
— Annual transfer of the funds to the State budget.

The outcome was a compromise, according to which an external segregated fund, the "State Nuclear Waste Management Fund", was established and detailed legislation was created for it. The nuclear companies were entitled to borrow back, at the market interest rate, 75% of the capital of the fund against the provision of full guarantees. The State was to have the right to borrow the remaining capital, i.e. at least 25%, at the same market interest rate. One factor contributing to this compromise was that the companies had already collected, pursuant to the then existing obligations, a relatively significant amount of money and a sudden transfer of that money into the Fund would have been complicated.

As mentioned above, the primary responsibility for nuclear waste management is assigned to the licence holders while the State has a supportive backup role only. Consequently, it was considered that it would not be appropriate to collect funds from the licence holders through a system based on a levy. Instead, the system selected was based on the requirement that at any moment there shall be, in the Fund, sufficient funds available to cover the remaining waste management measures necessary for the waste produced up to that time. Accordingly, the capital of the Fund is annually adjusted, normally with additional contributions from the licence holders. However, repayments from the Fund to the operators are also possible.

It is worth stressing that the Fund does not pay for the waste management measures, but continues to keep the money corresponding to the costs of the remaining measures. Theoretically, all the funds have been returned to the operators when they carried out all the necessary waste management operations. For these reasons, the Fund could be described as a "guarantee fund".

No obligation of balance sheet specifications to control the source of the money paid into the Fund has been set for the licence holders. Consequently, on the basis of the funding system it is not possible to consider precisely the effect of waste management costs on the cost of nuclear electricity. (It is worth noting that today the price of electricity is determined by market conditions.)

The cost of D&D immediately turns attention to the 'remaining waste management costs' when a facility is taken into operation. If such a large sum, forming a considerable portion of the total cost of waste management, were immediately transferred to the Fund, the effect of the costs would not be included on a timely basis and correctly in the production costs of electricity. Also, the construction costs of final disposal facilities for spent fuel constitute a type of significant investment cost which is completely discharged only in the distant future. When creating the funding system, this problem was solved by a provision that allows, during the first 25 years of operation of a nuclear facility, the collection of funds as a gradually increasing fraction of the calculated costs. However, in order to cover the total liability, the licence holder must give full guarantees to the State to cover the difference of the liability and the amount of the funded capital. For the existing four nuclear power plant units in Finland, the 25 year distribution period is now over.

In a way, one can say that each licence holder has its own ‘account’ in the Fund and the State authorities regularly establish the required balance of that account. According to the Nuclear Energy Act, the transfer of a nuclear facility to another legal person does not automatically transfer the obligation of waste management or the ‘account’ to the new owner; rather, the transferee has to open an account of its own. However, with the consent of the authorities the obligation of waste management and the ‘account’ can be transferred. In the case where the licence holder with an obligation of waste management is no longer capable of taking care of its obligation for financial reasons and/or measures of waste management, the State can take over both the waste and the ‘account’. The guarantees furnished by the licence holder to the Fund ensure that the Fund can return money to the State in time with the actual waste management measures.

According to the Nuclear Energy Act, the legal ‘person’ whose activities produce nuclear waste is fully responsible for nuclear waste management, including D&D. It can be released from that obligation only by the consent of the Government. If a nuclear power company ceases to exist or becomes unable to fulfil its obligation, the task is transferred to the State.

In theory at least, if a nuclear facility should for any reason stop its operation and also stop the production of more waste, the money accumulated in the Fund and the securities given to the State would together suffice to handle the situation and take care of the management of all the existing waste and the D&D of the plant. As the actual waste management measures would not be taken immediately, the interest accrued in the meantime by this existing capital is used to compensate for inflation and cost escalation.

3. OPERATION OF THE FINNISH NUCLEAR WASTE MANAGEMENT FUNDING SYSTEM

3.1. Organizations involved and their roles

The Ministry of Trade and Industry is responsible for nuclear energy in Finland. One of its duties is to ensure that the plans for waste management by the nuclear power companies and the implementation of these plans comply with the national policy. Each year the Ministry also determines, through various decisions, the amount of money each licence holder must have in the State Nuclear Waste Management Fund. The Ministry also makes sure that the operation of the Fund complies with legislation.

The State Nuclear Waste Management Fund is responsible for the management of the capital collected for nuclear waste management. The Fund has a Board of four members nominated by the Government. The Board has to include representatives from the Ministry of Trade and Industry, Ministry of Finance and the State Treasury. The current Chairman comes from outside the public administration. The Fund has two auditors, one of whom is selected by the nuclear power utilities. It also has a Managing Director, secretary and accountant, all part-time. Currently, the Fund's capital amounts to about €1200 million. In 2001, the profit of the Fund was €47 million. The annual administrative costs of the Fund have been about €50 000.

The Radiation and Nuclear Safety Authority (STUK) reviews, especially from the safety point of view, proposals on the basis of which the assessed remaining liabilities are established, and gives its opinion to the Ministry of Trade and Industry. In addition, the VTT Technical Research Centre of Finland reviews the proposals and cost estimates and gives the Ministry its opinion.

3.2. Assessment of liabilities

As mentioned above, the financial provisions for the future management of nuclear waste are based on the principle that the funds, covering the cost of the remaining operations needed to manage the waste that has already been produced, are available at any moment. Accordingly, the payments to the State Nuclear Waste Management Fund are based on the estimated costs for the future management of the currently existing nuclear wastes.

In practice, these estimates are based on proposals provided annually by each licence holder and confirmed, after scrutiny, and sometimes negotiations, by the Ministry of Trade and Industry. The cost estimates are always calculated in current prices, on the basis of current plans and technology. No discounting is used. These confirmed estimates or assessed liabilities form the basis for establishing the amount of money that each licence holder should have in the Fund. This amount that the Ministry also confirms each year is called the ‘fund target’. It is then up to the Fund to see that the licence holder’s share of the money in the Fund is balanced with the fund target.

To take into account the ‘fixed costs’, i.e. costs the total amount of which is not at all or rather weakly linked to the life cycle of the facility, the fund target is gradually increased during the first 25 years in proportion to the years of operation completed, so that the capital reaches the assessed liability sufficiently early before the estimated cessation of operation of the nuclear facility. From a licence holder’s point of view, the gradual collection method supports the evenly distributed transfer of waste management costs to the cost of electricity.

The detailed instructions for determining the fund target as a fraction of the liability are given in a Decision by the Council of State (Cabinet). The fund target depends on the energy produced, but there is a minimum target that must be reached even with no energy output.

It is worth noting that the assessed liability is not equal to the total cost of waste management, but is based on the estimated costs of the remaining measures. These estimates may change considerably during a year. Firstly, they are made according to current plans and technology. Thus, changes or corrections in plans, possible innovations and changes in the cost level as well as changes in national policy may influence the assessed liability. An example of the policy changes is the requirement, introduced at the beginning of 1995, of final disposal of all spent fuel in Finland. Secondly, the waste management operations carried out by a licence holder decrease the liability and sometimes these operations can be very costly. Actual examples of these kinds of changes are the completion of disposal facilities for low and intermediate level wastes. There are also other reasons that may give rise to sudden changes.

Due to the fact that the Fund targets are confirmed on the basis of assessed liabilities, these sudden changes can conflict with the aim that the cost of nuclear waste management should be smoothly transferred into the cost of electricity. To take this into account, the Nuclear Energy Act allows handling of an exceptionally large, sudden increase or decrease in the assessed liability, under certain precautions, by confirming temporarily (for a maximum of five years) the final liability that is lower/higher than the assessed liability.

Because of the method assumed to handle the high fixed costs and also major changes, the fund target can be less than the assessed liability. As a precaution against insolvency, the part of the assessed liability that is not covered by the money in the Fund must be covered with guarantees furnished by the licence holder. These guarantees are given to the Ministry of Trade and Industry, not to the Fund. They can, according to the Nuclear Energy Act, be a credit insurance provided by an insurance company, direct liability guarantees provided by a Finnish commercial bank, real estate mortgages or direct liability guarantees provided by a Finnish association. Mortgages on a nuclear power plant itself cannot be accepted. Each security has to be separately accepted by the Ministry of Trade and Industry. In practice, TVO has used direct liability guarantees of its shareholders and Fortum real estate mortgages related to its conventional power plants. As an additional precaution against unforeseen events, supplementary guarantees covering 10% of the assessed liability must be given to the Ministry.

3.3. Administration of the Fund capital

The State Nuclear Waste Management Fund manages the funds collected to guarantee future nuclear waste management. The Fund is to maintain and increase the value of this capital through a cautious lending policy and under the limitations set by the nuclear energy legislation. Any interest earned is added to the capital and in this way benefits the licence holders by decreasing the payments. On the other hand, all financial losses suffered by the Fund will be deducted from the capital of the Fund, a fact that introduces an element of collective liability into the system.

The share of each licence holder of the capital of the Fund or the amount of money each licence holder actually has in the Fund is called ‘fund holding’. The fund holding is made up of the payments by the licence holder, its relative share of the accumulated interests of the capital and also potentially of its share of the losses. The fund holding varies during the year and can be regarded as the daily balance of a licence holder’s ‘account’ in the Fund.

The fund holding related to the last day of the previous calendar year is compared by the Fund with the fund target determined by the Ministry of Trade and Industry; the difference is defined either as a fee to be paid to the Fund or as a refund to be paid to the licence holder. Refunds to the licence holders will be more probable now, when the accumulation period of 25 years is over and waste management plans and measures are being actively implemented. However, some returns have been occasionally paid due to changes in waste management plans and high real interest rates.

The accumulated capital is lent out by the Fund. A licence holder, or its shareholders, can borrow back up to 75% of its fund holding against full guarantees given to the Fund. The Board of the Fund must in each case approve these securities, which should not be mixed with the guarantees given to the Ministry. TVO normally provides direct liability guarantees of its shareholders and Fortum uses shares it owns in a hydropower company.

In normal cases, the fixed period of a loan is five years. The interest rate is presently fixed by legislation to be Euribor +0.15%.

The remaining Fund capital, consequently at least 25%, is offered to the State as a loan with the same interest rate. The part of the capital that the licence holders, their shareholders or the State do not want to borrow is to be invested against full guarantees in some other way yielding the best possible return. The utilities and the State have normally borrowed the amounts they have been entitled to. Only earlier, during a certain period when the fixed interest rate at that time was rather high, did the State not fully use its right to a loan. The total amount of money borrowed by the State is today some €250 million.

The Ministry of Trade and Industry confirms, at the end of January, the assessed liabilities as of 31 December and determines the corresponding Fund targets. The State Nuclear Waste Management Fund then determines, in February, the fund holding of each licence holder at the end of the previous year and the balance between this fund holding and the fund target. On 1 April, all payments to and from the Fund, including those connected with the issuing and repaying of the loans, are made simultaneously, in practice largely compensating each other. Thus, the actual money flows are often much smaller than the determined fees.

In the licence holder's (company's) balance sheet, a payment to the Fund is an expense, and a received payment from the Fund is an income. This expenditure or income is included into the balance sheet of the calendar year ending before the payment is actually made since it reflects the situation at the end of that year. The annual waste management fee is treated as a deductible expense and the possible return from the Fund is taxable income. However, the costs of waste management measures carried out by the company during the previous calendar year and which reduce the remaining waste management costs, in that way either having a decreasing effect on the fee or causing a payment from the Fund, are treated as deductible expenses. Thus, at least in theory, the actual expenses are balanced by the return from the Fund.

4. SPECIFIC ISSUES CONNECTED WITH THE COSTS OF D&D

4.1. Dismantling and decommissioning plans for the power plants

The four nuclear power reactors in Finland were put into operation between 1977 and 1982, while the current operation licences will be in force until the end of 2007 (Loviisa) and 2018 (Olkiluoto), as mentioned earlier. The decommissioning plan for the Loviisa power plant is based on immediate dismantling in less than ten years from the shutdown of the reactors, excluding facilities needed for spent fuel storage. The current basic plan for the Olkiluoto power plant envisages a 30 year safe storage period prior to dismantling of the reactors. When the planned life cycle for all the units is at a minimum of some 40 years, and if D&D plans are carried out following the current plans, the D&D period of the existing plants would start approximately in 2030 and be completed in 2060 or later, depending on the final life cycle.

According to a policy implemented by decisions of the authorities, the licence holders have, since 1983, been obligated to update their decommissioning plans every five years. These plans aim at ensuring that decommissioning can be appropriately performed when needed and that the estimates for the decommissioning costs are realistic. The latest updates of these decommissioning plans were published at the end of 1998. So the next updating will take place by the end of 2003.

The Finnish decommissioning plans cover dismantling of only structures and components that exceed the clearance constraints. Similarly, the funding system covers only radioactive waste from the dismantling. The ‘green field’ option is not required. The estimated amount of waste to be disposed of is 15 000 m^3 for the Loviisa plant and 28 000 m^3 for the Olkiluoto plant.

Some essential technical details of the decommissioning plans have not been fixed so far. For instance TVO, in spite of its primary option of delayed dismantling, is also studying the immediate dismantling option. Furthermore, the company has not decided finally whether the pressure vessels will be disposed of in pieces or as a whole.

Both nuclear companies plan on-site disposal of dismantling waste. The existing underground repositories for operating low and intermediate level waste would be expanded for the disposal of dismantling waste. In addition to technical benefits, on-site disposal is estimated to be much more cost effective compared with other alternatives. The decommissioning waste disposal plans include fairly comprehensive safety assessments.

4.2. Cost estimates

The cost estimate of D&D using the current price level is €192 million for Loviisa and €156 million for Olkiluoto. Accordingly, the total sum of provisions for D&D is now about €350 million, or about one third of the total sum of provisions for nuclear waste management in Finland.

In international comparisons, the estimated costs of D&D in Finland seem to be relatively low. Many reasons for this can be identified. First of all, the basis of calculations varies significantly from one country to another. The fact that dismantling according to Finnish legislation involves the contaminated parts of the facility only naturally limits the cost of dismantling compared with that of the green field option. Secondly, considerable cost reductions are assumed to be achieved through effective arrangements at the site, and especially from the on-site final disposal of decommissioning waste.

The critical question, however, is not the exactness of the cost estimate today, but how the system takes into account the difficulty of arriving at reliable estimates. As a nuclear company may at any time, at least in theory, lose its capability for, or interest in, the orderly management of D&D, the Finnish funding system contains some built-in features to minimize the risk of the State having to contribute additional funds to carry out these operations.

It is obvious that the estimates of D&D costs have, especially in the past, been mostly based on theoretical considerations. However, the system continuously requires new, updated estimates that must take into account the practical experience accumulating worldwide. The estimates must not rely on improvements in waste management methods, but must, according to the law, always be based on the technology currently available. In addition, the law also requires that the uncertainty of available information about prices and costs shall be taken into account, in a reasonable manner, as raising the estimated liability.

The transfer of funds on the account of a licence holder to the State has already been mentioned. In this situation, the Fund has full rights to require the licence holder to pay its loans back to the Fund or, alternatively, to realize the securities. The interest of this capital is also available to the State and is assumed to compensate for inflation and related cost escalation. The State can also, if there is a need, realize the 10% supplementary securities.

5. CONCLUDING REMARKS

The Finnish nuclear waste management funding system has been in operation for almost 15 years and has worked smoothly, to the satisfaction of all parties. The real test is, however, still ahead. This will be experienced sometime in the future if and when a nuclear company has ceased to exist and neglects all its financial obligations. Then one will see whether society is willing to use all the strong means it has at its disposal under legislation to extract the necessary funds from the securities. It is also worth remembering that repayment of the funds loaned to the State have to be collected from the taxpayers.

Appendix E

KÄRNREAKTORN R1
-
ETT STYCKE HÖGTEKNOLOGISK PIONJÄRHISTORIA

av
Karl-Erik Larsson

Daedalus 1981,
Tekniska Museets Årsbok, årgång femtio, Stockholm 1981.

Författaren, Professor Karl-Erik Larsson, samt Tekniska Museet i Stockholm tackas för välvilligt samtycke till denna publicering.

Kärnreaktorn R1 – ett stycke högteknologisk pionjärhistoria

Av Karl-Erik Larsson

Bakgrunden

Den 6 augusti 1945 revs förlåten undan med stor dramatisk kraft och blottade en scen, som ytterst få personer omedelbart förstod: atombomben föll över Hiroshima och därmed sattes snabbt punkten för ett krig, som i sex år pinat mänskligheten. Denna dramatiska start av atomåldern drog med sig en förbannelse, som det fredliga atomära verksamhetsområdet sedan ej har lyckats bli förlöst ifrån. Atombomben alstrades av hat och skapade permanent fruktan. Men låt oss skaka av oss dessa obotliga olustkänslor och granska den fredliga utvecklingen inom atomområdet och särskilt utvecklingen i vårt land. Det leder till skildringen av utvecklingen av en ny, högteknologisk verksamhetsgren inom svensk industri.

De första stapplande stegen mot utvecklingen av denna högteknologi togs i byggnader, som ligger centralt i Stockholm, alldeles i närheten av Tekniska Högskolan (KTH). Här utvecklades och byggdes under åren 1948–54 Sveriges första kärnreaktor, som kom att kallas R1 (R som i *r*eaktor).

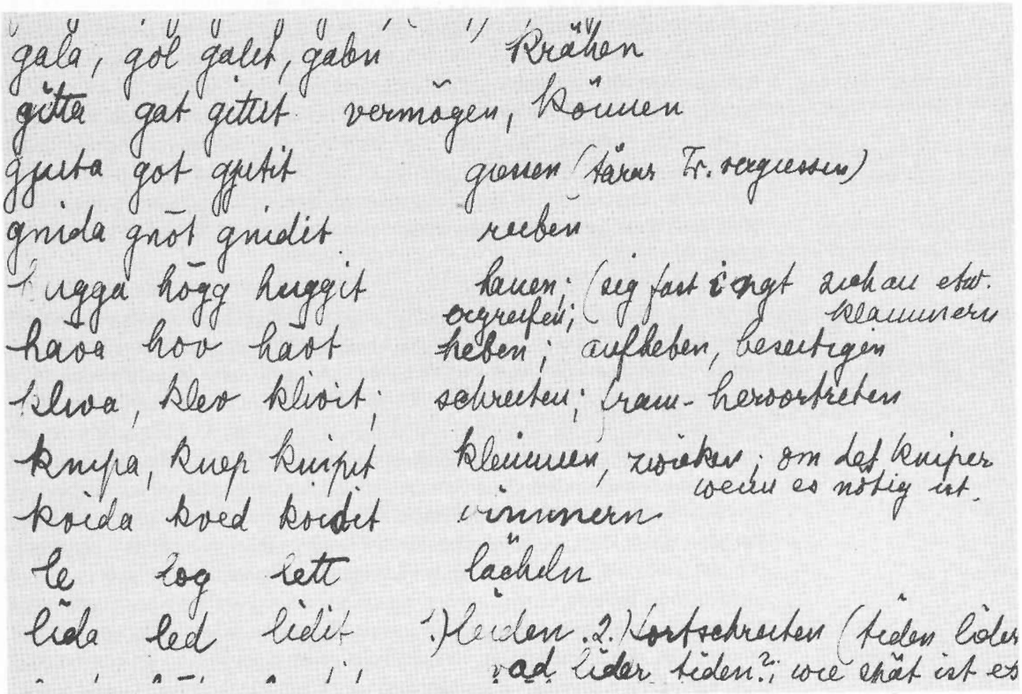
Som bakgrund till den svenska utvecklingen på detta område får man erinra sig inledningen. Det var tydligt redan 1945, att i världens vapenarsenaler skulle komma att ingå ett nytt vapenslag: kärnvapnen. I anledning härav måste även Sveriges regering se om sitt hus på försvarets område och inlemma studier av de nya vapnens verkan och eventuellt framställning inom ramen för försvarets forskning organiserad i Försvarets Forskningsanstalt (FOA). Den från USA tillgängliga officiella informationen på området var minimal och av den typ, som återfinnes i Smythes rapport av år 1945. (Officiell Amerikansk Rapport över Atombomben 1945.) Så mycket stod dock klart, att inkörsporten till det nya teknikområdet – civilt och militärt – var en atomstapel som kärnreaktorn först kom att kallas. I civil användning lovade kärnreaktorn att bli en energikälla av en hittills icke skådad kompakthet. Tidningarna innehöll vid den tiden lockande notiser t ex om att energiinnehållet i en mängd uran, som ryms i en tändsticksask, var nog att driva en atlantångare tur och retur Göteborg–New York.

1. Lise Meitner, den tyska forskare, som verksamt bidrog till upptäckten av kärnklyvningen, vistades från årsskiftet 1938–39 i Sverige som flykting.

Bilden visar utdrag ur hennes personliga anteckningsböcker förda redan 1939 i Sverige. Därav framgår hur hon

a. strävade efter att lära sig svenska
b. fortsatte sin experimentella forskning i Sverige
c. spekulerade över hur kärnklyvning egentligen var möjlig.

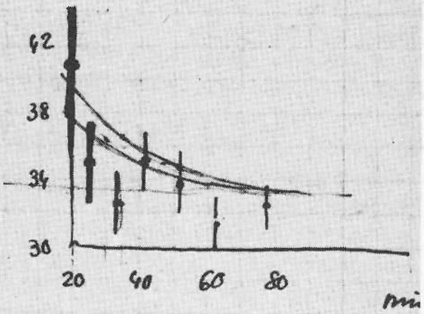
(Ur Lise Meitners efterlämnade kvarlåtenskap. Studsvik Energiteknik AB, Biblioteket)



gala, gol galit, gabn — Krähen
gitta gat gittit — vermögen, können
gjuta göt gjutit — giessen (Särnr. Fr. vergiessen)
gnida gnöt gnidit — reiben
hugga högg huggit — hauen; (sig fast i ngt sich an etw. klammern) ergreifen;
hava hov havt — heben; aufheben, beseitigen
kliva, klev klivit; — schreiten; fram- hervortreten
knipa, knep knipit — kneifen, zwicken; om det kniper wenn es nötig ist.
kvida kved kvidit — wimmern
le log lett — lächeln
lida led lidit — 1.) leiden 2.) fortschreiten (tiden lider vad lider tiden?; wie spät ist es

a.

$22^{h}17$ 18 Min ß	8265			
22	8429	164	41,0 ± 3.2	
28	8641	212	35,3 ± 2.4	
36	8900	259	32,4 ± 2	
46	9250	350	35.0 ± 1.9	
56	9588	338	33.8 ± 1.8	
67	9933	345	31.4 ± 1.7	
87	0578	645	32.3 ± 1.3	
$12^{h}05$ 124	1814	1236/37	33,3	



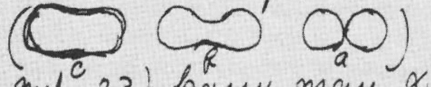
$\boxed{29.\text{II}.39.}$

Z_0 (neues Cellophan)

b.

$\alpha_4 = -243/595\ \alpha_2^2$ 23)

(das Minuszeichen ist eine Verkürzung) in Übereinstimmung mit der Tatsache, dass da die kritische Form mit abnehmendem Z^2/A mehr länglich wird sich eine Konkavität ausbilden muss um den Aequatorial Gürtel, um kontinuierlich mit Änderung der Ladung zu der dumpbell shaped figure zu kommen ( )



mit 23) kann man α_4 aus 22 eliminieren und dann sehen für welches α_2 ΔE ein Max wird, wodurch man E_f erhält, dass eine Deformation an der Grenze der Kernes erzeugt!)

$$E_f/4\pi r_0^2 \sigma A^{2/3} = f(x) = 98(1-x)^3/135 - 11368(1-x)^4/34425 + \dots$$

durch geeignete Interpolation kann man jetzt $f(x)$ oder E_f als ft von $Z^2/A / Z^2/A$ Limit darstellen (Fig 4)

es wird zwischen $f(0)$ und $f(1)$ ($x = 0$ bzw $= x = 1$, $Z = 0$ " Z^2/A sehr nahe (Z^2/A) limit) eine glatte Kurve gezogen.

c.

107

:

Nya forsknings-uppgifter

För att ta hand om utvecklingen av atomstaplar, för att ta fram bränsle och annat material till sådana samt för att bedriva härav betingad forskningsverksamhet grundades år 1947 det halvstatliga bolaget AB Atomenergi (AE). Så sakta började nya lovande forskare och tekniker att anställas i det nya bolaget, inom vilket bl a arbeten att utvinna uran ur våra svenska skiffrar utfördes. Forskning på det området drevs parallellt och allra först inom FOA. Det stod tidigt klart, att man vid konstruktion av en reaktor stod inför en rad olösta frågor, varför forskning och utveckling behövdes inte bara inom den kemiska delen med framtagning av uran som huvudmål utan även inom materialtekniken i övrigt t ex i samband med kapsling av uranet i en reaktor och i början även inom grundläggande delar av fysikområdet. Unga tekniska fysiker och universitetsfysiker sögs in i det nya området, ibland innan de ännu ej avlagt sin examen. Många av dessa anställdes vid FOA, avdelning 2, andra direkt vid AB Atomenergi. Forskningen inom de två organisationerna samordnades och leddes av dåvarande laboratorn vid FOA, fil dr Sigvard Eklund. Det kan vara av intresse att veta att en av de redan etablerade forskare, som kraftfullt deltog i värvning av ledande personal till FOA och AE-befattningar var professor Hannes Alfvén vid KTH, som under 1970-talet ju inte precis gjort sig bekant som kärnenergifältets främjare åtminstone vad gäller den gren av kärnenergi, som vi kallar fissionsenergi.

Osäkerheten inför de nya gigantiska arbetsuppgifterna var stor bland alla oss unga kring åren 1947–49, som tog itu med att lära oss vad vi egentligen skulle göra. Men vad som brast i kunskap ersattes av entusiasm och vilja att åstadkomma det nya, att ta det första steget in i atomåldern. Den arbetsuppgift man fick var kanske föga väldefinierad, t ex "gör något med neutroner". Inte lätt för en okunnig, mycket ung fysiker, som nätt och jämt visste att neutroner fanns. Men det var inte bättre ställt bland den stora allmänheten. Det kunde långt fram på 50-talet hända, att man tillfrågades om var man egentligen kunde "bryta atomer". De enda neutroner vi kunde åstadkomma var sådana, som frigjordes från små radioaktiva preparat inköpta från det mäktiga Union Minière de Haut Katanga, en belgisk firma, som även var en viktig uranproducent för USA, samt så småningom från små acceleratorer, ur vilka neutroner erhålls genom kärnreaktioner. Allt material var svårt att få tag i, men många firmor ville lyckligtvis sälja t ex mycket små mängder tung vätgas eller tungt vatten.

Forskningens organisation

Från 1 juli 1950 övergick en större grupp unga forskare och tekniker från FOA till AE. Därefter organiserades inom AE:s fysiska forskningsavdelning under Eklund arbetet på att ta fram den första reaktorn så, att sex olika sektioner formades med ansvar för olika revir sträckande sig ut i den okända värld som vi nu beträdde. Den intellektuella upptäcktsresan kunde börja. Vår tids svenska Columbus hade sjösatt sitt skepp. Dessa sektioner var i princip:

1. Experimentell reaktor- och neutronfysik (Guy von Dardel)
2. Teori. Reaktorns teori och beräkning (Nils Svartholm, Gunnar Holte)
3. Kärn- och reaktorkemi (Erik Haeffner)

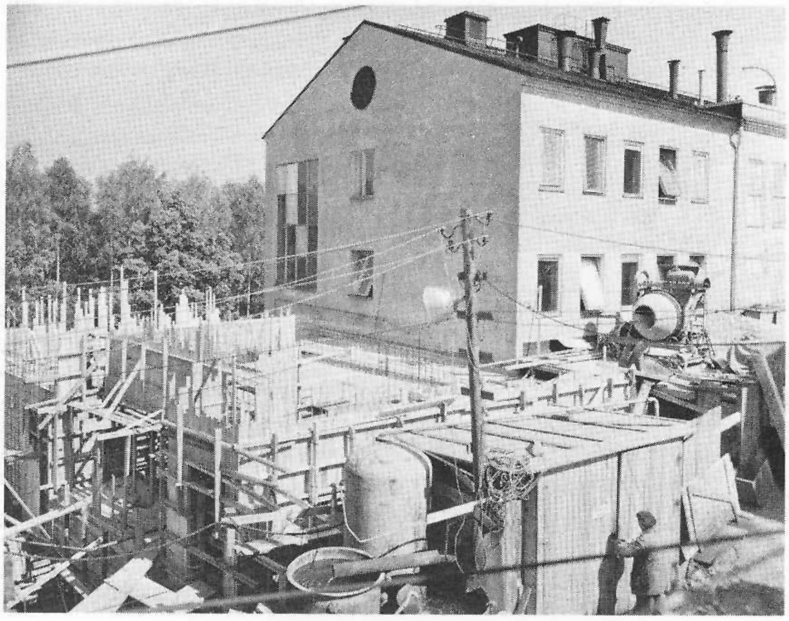
4. Instrumentering, styrning och kontroll av reaktor (Robert Vestergaard)
5. Konstruktion och byggnadsplanering (Gösta Lindberg)
6. Strålskydd och dosimetri (Lars Carlbom)

Namnen inom parentes anger personer som var sektionschefer. Listan över sektioner ser imponerande ut. Men vi var i själva verket inte så många omkring 1950. En handfull personer inom varje sektion. Arbetet inom varje sektion drevs dock med kraft och energi, och även om man knappt visste vad en reaktor var och än mindre kände till alla nödvändiga data, så arbetade varje grupp som om problemen inom angränsande grupper vore lösta. Reaktorn började konstrueras och byggas, innan man visste om dess grundläggande fysik och kemi. Men nog gick det sakta i början, tyckte vi otåliga, och det avspeglades bl a i de olika namnförslag, som dök upp för den första reaktorn. I Harwell, Englands atomcentrum, fanns en förstlingsreaktor med grafit och naturligt uran. Den kallades GLEEP = *G*raphite *L*ow *E*nergy *E*xperimental *P*ile. I analogi härmed föreslogs för vår första svenska benämningen SLEEP = *S*wedish *L*ow *E*nergy *E*xperimental *P*ile. Eller t o m SLURK = *S*vensk *L*ågenergi *U*ran *R*eaktor *K*anske. Namnförslagen speglar våra känslor vid början av 50-talet.

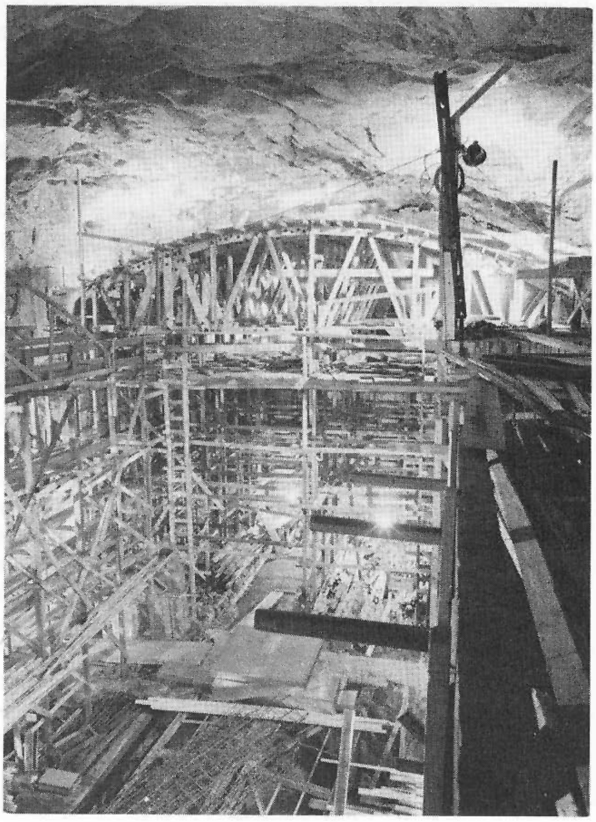
Koncessions-proceduren

Den tilltänkta reaktorns placering är värd några ord. Det stod klart för ledningen, att det värdefullaste man kan ha, är tillgång till skicklig personal samt en nära kontakt med vetenskaplig och teknisk forskning. En placering av den första reaktorn nära KTH bedömdes som värdefull. Helst i omedelbar anslutning till den byggnad vid Drottning Kristinas Väg 47–49, som kallades IVA:s försöksstation, i vilken gökungen AB Atomenergi snabbt trängde ut andra mindre snabbväxande bokamrater över kanten. På den tiden susade Djurgårdens björkar skönt över markerna norr och öster om försöksstationen. KTH hade ännu inte erövrat så stora delar av denna gröna oas i norra Stockholm. Andra skäl till denna placering av R1 var att man slapp bygga nya laboratorielokaler m m.

Undret inträffade. Tillstånd erhölls att bygga reaktorn i nordändan av försöksstationen. Den skulle dock inte komma att störa omgivningen, då den av olika skäl, t ex säkerhetsskäl, byggdes 25 meter under jord i präktigt berg. Drottning Kristinas Väg 51 skulle komma att bli adressen till hiss- och vakthuset och nedgången till R1. Numera måste var och en, som vill starta en reaktor, genomgå en oerhörd byråkrati i form av koncessionsansökningar och tillstånd. Särskilt med hänsyn till eventuell förekomst av radioaktiv strålning måste dock ett visst tillstånd erhållas även kring 1950. Men då fanns varken kärnkraftsinspektion eller strålskyddsinstitut utan myndigheterna, som utfärdade villkoren och löftet om start, var medicinalstyrelsen och radiofysiska institutionen vid Karolinska Sjukhuset. Denna leddes av den frejdade radiologiske forskaren och organisatören professor Rolf Sievert. Det var han och medarbetare, som avgjorde villkoren för R:1s start ur denna synpunkt, och som bestämde de radiologiska kontroller som skulle behövas. Vi, som arbetade här på 40-talets slut och början av 50-talet, glömmer inte dåvarande laborator Sven Benner, medarbetare till Sievert som regelbundet besökte oss och mätte våra strålnivåer med ganska enkla metoder.



2. Från byggnadsarbetet kring reaktorn. Överst trapphuset vid skorstenen under byggnad. Därunder tar bergrummet med platsen för reaktorn form. Bilden tagen 1953 i juli resp april. (Studsvik Energiteknik AB, Biblioteket)



Säkerhetsnormer

I samband med detta omnämnande av den enkla koncessionsproceduren skall här en annan viktig faktor framhållas i det pionjärarbete, som pågick. Alla vi, som arbetade i radiologiskt arbete, gjorde oss i hög grad till självansvariga för utveckling av säkerhetsnormer. Teknikerna tog själva initiativ till den nödvändiga säkerhetsbedömningen med avseende på den radioaktiva strålningen.

Säkerhetsbedömningen bakades samman med den tekniska och ekonomiska utvärderingen av projekten. Ja, säkerheten för allmänhet och anställda sattes som första punkt. Det torde innebära ett unikt och nytt initiativ i utveckling av en ny tung teknologisk verksamhet. I den debatt som förts under 1970-talet, och där yngre professionella miljö-aktivister varit pådrivande, har det låtit, som om teknikerna glömde allmänhetens säkerhet. Inget är mera felaktigt. Trots tidspress, stor innovationshöjd och brist på vissa fundamentala kunskaper intog skydds- och säkerhetsaspekterna en styrande roll i det dagliga arbetet, i konstruktion av reaktorn och i kontakterna med allmänheten.

Ni, som idag är skeptiska till kärnenergi, spärrar kanske förfärat upp ögonen vid genomläsning av denna beskrivning. Hur kunde man vara så släpphänt att medge en placering av en reaktor mitt i Stockholm? Svaret är: På denna tid härskade i vårt land liksom i stora delar av den tidigare krigshärjade världen en stark vilja att förbättra livsvillkoren för alla. Tekniken erbjöd en väg ut ur de små resursernas eller rent av fattigdomens samhälle. Allt som främjade utvecklingen mot en ny värld var välkommet. Ett bättre liv var också ett mera spännande sådant. Dynamiskt istället för statiskt. Ordet "atom" hade positiva förtecken. En generation, som njutit frukterna av denna allmänna utveckling till förbättrade materiella villkor, synes till stora delar ha glömt dessa sammanhang. Erfarenheten visar för övrigt nu efteråt att alla säkerhets- och miljöbedömningar, som utfördes, var i stort sett korrekta.

Ingen kom någonsin till skada genom arbetet med R1 under åren 1950–1970. Det hör också till saken, att R1 långt senare, när moderna kontrollmyndigheter utformats, fick en förnyad och efter alla konstens regler utformad och bedömd koncessionsansökan bifallen.

Den "svenska linjen"

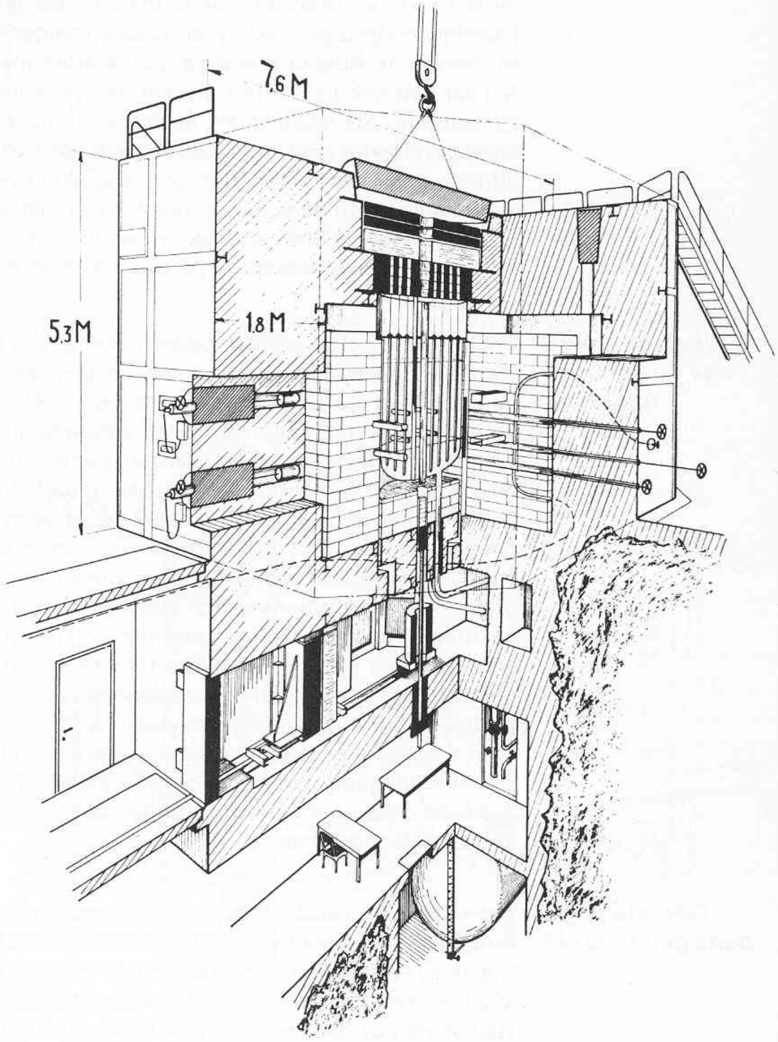
Vilken sorts reaktor skulle vi nu bygga? I själva verket var möjligheterna inte många omkring 1950. En s k termisk reaktor (namnet termisk därför att den drives av termiska och långsamma neutroner) arbetar med uran som bränsle och t ex grafit, tungt vatten eller lätt vatten för att bromsa ned de snabba neutroner, som kommer från fissionsprocessen, så att de blir långsamma eller termiska. Uranet består som bekant av de två atomslagen, uran-238, som är tyngre, och uran-235, som är lättare. Det *naturliga uran*, som vi finner t ex i vår skiffer (Kvarntorp, Ranstad) eller i berg eller i havsvattnet innehåller 99,3 % uran-238 och 0,7 % uran-235. Det är i stort sett endast uran-235, som driver reaktorn, och ur vilket energi utvinnes. Uran-238 bidrar till en mindre del genom att det omvandlas till uran-239, vilket därefter omvandlas genom ett par radioaktiva sönderfall till plutonium-239. Detta senare ämne, plutonium-239 är jämte uran-235 bränslet i en reaktor. (Det kan noteras, att exakt samma ämnen också är materialet i atombomber, men då skall de vara renodlade.) De olika materialens egenskaper är sådana, att en reaktor kan fås att fungera med *naturligt uran*, endast om man använder tungt vatten eller grafit som nedbromsningsmaterial för de neutroner, som kommer ut ur uranet vid kärnklyvningen. Uran med annan sammansättning av uran-235 och uran-238 än den naturliga, s k anrikat uran, var otänkbar att komma över 1950. En fungerande reaktor med naturligt uran blir mindre i dimensionerna och drar mindre urankvantiteter, om den utföres med tungt vatten som neutron-nedbromsare (moderator) än med grafit. Därför blev också valet för R1: naturligt uran och tungt vatten. Den kombinationen kom sedan att lanseras som den "svenska linjen". Liknande reaktorer fanns dock i början av 50-talet i bl a USA, Canada, Frankrike och Norge. Linjen har fram till dags dato följts upp endast av Canada, som utvecklat den s k Candu-reaktorn.

Fanns det andra argument att välja den kombinationen för R1? Ja, innan man definitivt avfärdat planer på svenskt kärnvapen, så var tungvattenreaktorn den bästa plutoniumproducenten. Den har ett högt konversionsförhållande, vilket betyder, att den effektivt omvandlar uran-238 till plutonium-239. Överhuvudtaget om större mängder kvalitetsplutonium önskas för andra reaktorändamål, t ex bridreaktorer, så erbjuder tungvattenreaktorn en bra start. (Bridreaktor = en reaktor som producerar minst lika mycket eller mer brännbart material än den förbrukar.)

Fransk experimentreaktormodell

Modell för den första svenska experimentreaktorn, R1, var en fransk motsvarighet kallad ZOÉ. Från AE:s sida upprätthölls mycket goda kontakter genom Eklund som drivande kraft med den internationella atomforskningssocietén. Och inte minst med den franska. Viktig information strömmade på det sättet in till oss i det svenska laget. Hemlighållandet av atomuppgifter var emellertid mycket strikt fram till 1955. På det reaktorfysiska- och teknologiska området var enkla översiktssammanfattningar av typen Goodmans "Nuclear Science and Engineering" enda tillgängliga information tillsammans med s k "declassified documents" från amerikanska atomenergikommissionen. Det var därför mycket värdefullt, när amerikanska experter som t ex Milton C. Edlund kom över till oss från USA och höll föreläsningar ur det, som sedermera blev den berömda boken "Nuclear Reactor Theory" av Glasstone och Edlund.

Samarbetet med Frankrike och med vårt grannland Norge blev det, som ledde till att R1 möjliggjordes. Frankrike kom nämligen att till Sverige överlåta de ca 3 ton naturligt, metalliskt uran kapslat i aluminiumrör med ca 3 cm diameter, som utgjorde bränslet. Från Norge inhandlades under mycket stor sekretess de ca 5 ton tungt vatten, som behövdes i reaktorn. Man arbetade visserligen framgångsrikt inom AE, under den driftige ingenjören Erik Svenke som ledare, med att laka ut uran ur våra svenska skiffrar, men till R1 kunde inte svenskt bränsle bli färdigt. Det skall också noteras, att bränslet i R1 var uranmetall. Detta innebar en begränsning, då man med sådana bränslestavar inte kunde gå upp i några nämnvärda temperaturhöjningar av moderatorvattnet. För att nå högre temperaturer som i de kommande värme- och kraftalstrande reaktorerna måste annat och termodynamiskt stabilare bränsle användas, nämligen keramiskt ma-



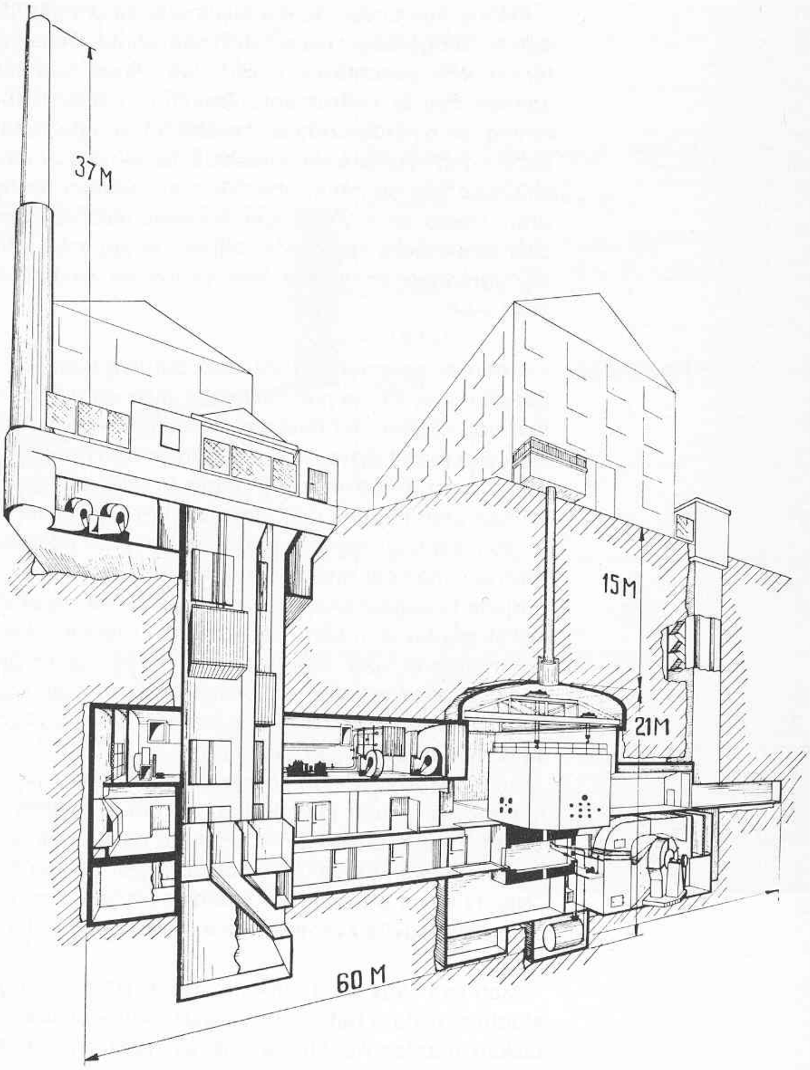
3. Tvärsnitt av reaktorn. I mitten ses reaktortanken, som innehåller 5 ton tungt vatten och 3 ton uran i form av 126 stavar. Tanken är på sidor och botten omgiven av 90 cm grafit. Till skydd mot radioaktiv strålning omges reaktorn av 1,8 m tjocka betongväggar. Reaktortanken är åtkomlig uppifrån med hjälp av en travers. Kärnreaktionerna inuti reaktorn regleras med två plattor av kadmium, som löper mellan tankväggen och grafiten. För experiment, mätändamål och framställning av radioaktiva isotoper finnas ett stort antal horisontella och vertikala kanaler. Reaktorn startades den 13 juli 1954. (AB Atomenergi) terial i form av uranoxid. Detta utvecklades tillsammans med det nya kapslingsmaterialet zirkalloy, en zirkoniumlegering, långt senare. (Numera professorn Roland Kiessling gjorde stora insatser på detta område.) Allt var nytt och okänt. Hur mycket de nya materialen skulle tåla var en nyckelfråga. Att reaktorn R1 ej skulle kunna köras upp i några högre termiska effekter stod klart. Den skulle bli en experimentreaktor. Den lilla effekt, som bildades i reaktorn, skulle effektivt kylas bort med luftkylning av det tunga vattnet, som från reaktorn cirkulerades ut till värmeväxlare. En arbetshypotes blev, att reaktorn skulle kunna drivas vid maximalt 100 kW (jfr dagens kraftreaktorer på ca 2.700.000 kW termisk effekt).

Arbetsgruppernas forskningsuppgifter

Det mörker de olika arbetsgrupperna kring R1:s tillblivelse svävade i från starten i slutet av 40-talet skingrades relativt snabbt. Inom fysikergrupperna bekantade man sig effektivt med neutroner och kärnfysikaliska effekter i reaktorn, inom kemisektionen arbetade man med frågor om anrikning (man framställde exempelvis anrikat bor-10, som var mycket bra att ha i neutronräknare), man studerade upparbetningsfrågor och plutoniumkemi, man lärde sig att rena tungt vatten, som blivit förorenat i reaktor, man studerade strålningsinverkan på vatten och vattenlösningar, s k strålningskemi, och man ägnade sig allmänt åt bränslets behandling från produktion till utbränning och upparbetning. Konstruktörerna arbetade på utformningen av den nya maskinen och från juli 1951 sprängdes de två schakten ned i berget vid IVA:s försöksstation. Instrument fanns bara delvis att köpa, varför en omfattande instrumentutveckling ägde rum. De grupper, som konstruerade det elektroniska kontrollsystemet hade att ta fram nya instrument och introducera den då rätt okända reglertekniken på det raffinerade styrproblemet för reaktorer. Parallellt härmed beräknades olika detaljer av reaktorns förutserbara beteenden, dess kritiska storlek etc av teorigrupperna.

Tvärvetenskapens genombrott

Det vore fel att påstå, att denna skapelseprocess försiggick utan våndor. Precis som vi än idag vid t ex en teknisk högskola har representerat hela skalan av vetenskap och teknik, från de mest grundläggande vetenskapsidkande över det tekniskt nyskapande idéstadiet och slutligen till verkställigheten av den tekniska processen med hänsynstagande till ekonomi, säkerhet, miljö i de mest tillämpade ämnesområdena, så fanns detta register representerat vid vår R1-verksamhet. Och visst uppstod konflikter. Förespråkare för mera grundläggande tillvägagångssätt med analys av många nyckelprojekt från den fysikaliska eller kemiska elementarnivån kom ofta att stå mot de praktiskt och strikt ingenjörsmässigt inställda, som mera med tumregeltillämpningar och delvis obestyrkta approximationer men med stöd av teknisk intuition och "lagom med vetenskap" ville snabbt nå resultat. Det fick kompromissas och ofta vann nog den driftiga ingenjörssidan. Resultatet blev till slut gott: reaktorn fungerade bra, och man fick nöja sig med att i efterhand studera de mera grundläggande frågorna av fysisk, kemisk, materialteknisk eller annan art som lämnats bakom. Detta blev sedermera olika specialiserade forskares uppgifter.



4. Tvärsnitt av den underjordiska byggnaden med reaktorn och tillhörande laboratorier, ventilationsaggregat och värmeväxlare. Bergrummet är förbundet med markytan genom två schakt, det ena med två hissar och det andra med lufttrummor för värmeväxlaren och en trappa som reservutgång. Ventilationsluften utblåses genom skorstenen.

Sprängningen av bergrummet började i juli 1951 och avslutades i oktober 1952. Den utsprängda volymen är 13 000 m³. (AB Atomenergi)

Många fick under denna fascinerande pionjärtid ägna sig åt ett betydande mångsyssleri både i stort som smått. Detta var verkligen ett tvärvetenskapens genombrott i stort. Jag erinrar mig exempelvis hur jag bl a ägnade mig åt instrumentutveckling, precisionsmätning av neutroner, rening av bortrifluoridgas, beräkning av neutronflöden i reaktorer, utveckling av viss strålskyddsteknik, beräkning av utbredning av radioaktiv strålning från ett moln, utveckling av metoder för flygprospektering efter uran. Dessa senare metoder testades med ett gammalt Storch-flygplan över Kvarntorps skifferfält. Sällan har jag mått så illa som under dessa lågflygningar. Så kunde livet te sig för unga forskare i dessa dagar 1948—54.

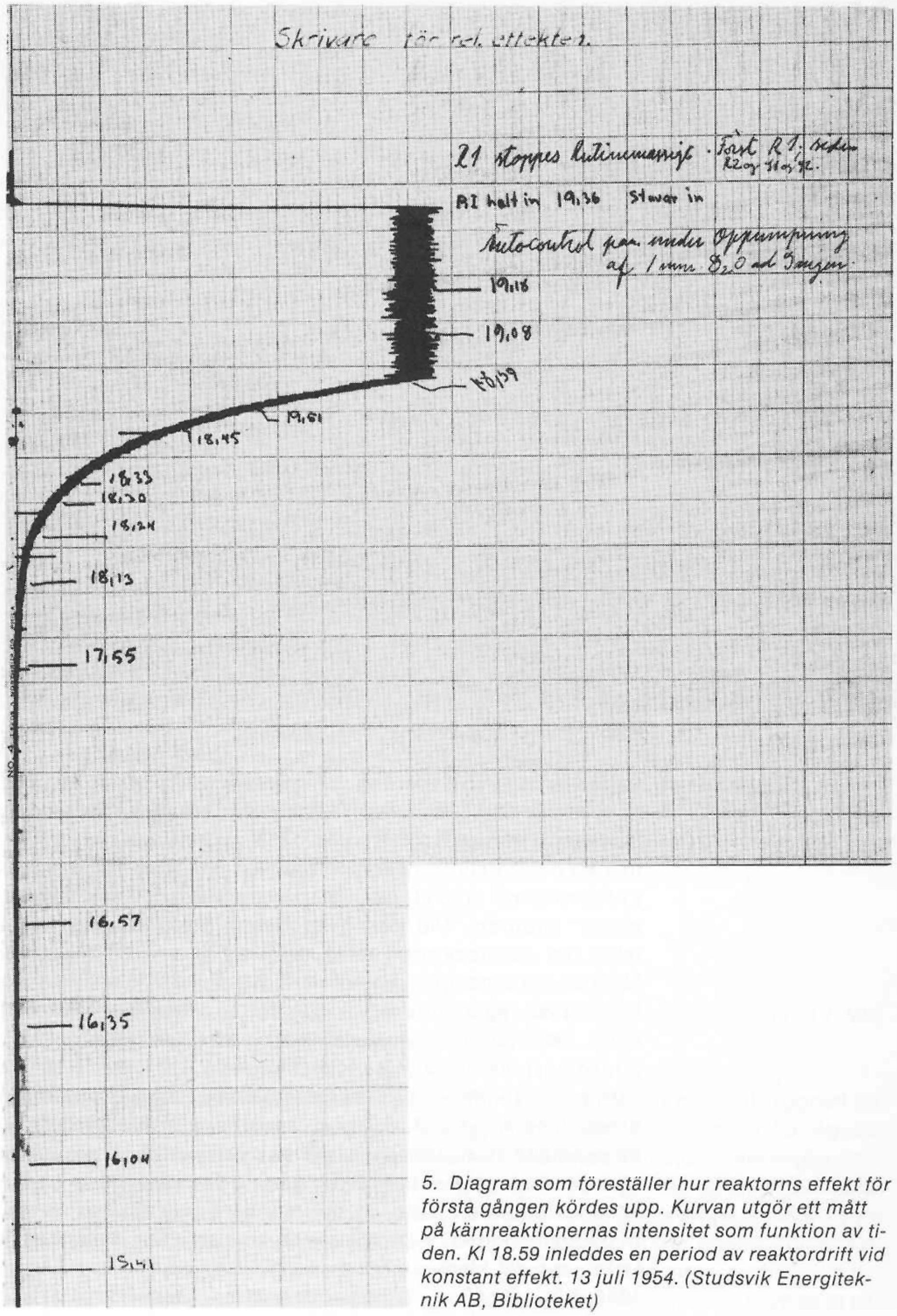
R1 på plats

I mitten av sommaren 1954 stod det nya tekniska underverket på plats. Bergrummet var vackert blåmålat med en mängd ljuspunkter i taket av den höga salen. Det hela gav illusionen av en oföränderlig stjärnhimmel över den febrilt dynamiska verksamheten i reaktorhallen. Den 13 juli 1954 började det tunga vattnet pumpas in i den aluminiumtank, i vilken de tre tonnen uran hängde ned i form av 126 uranstavar. De bortrifluoridräknare, som känsligt registrerar neutroner, gav i början en eller annan impuls ifrån sig, men när nivån av tungt vatten nått ett par meter upp i tanken, så började tickandet öka först sakta, sakta, så snabbare: reaktorn hade nått kritisk storlek och kärnreaktionerna i reaktorn drev nu sitt eget dock väl kontrollerade spel. Sverige hade inträtt i atom-åldern, ca elva och ett halvt år efter att den italienske fysikern och nobelpristagaren Enrico Fermi startat världens första reaktor, CP-1, i Chicago den 2 december 1942. AE var då ca 7 år gammalt.

En ledande och stolt konstruktionsingenjör vid R1 yttrade en gång då reaktorn var under tillverkning: ''Vi gör ju faktiskt om Fermis bedrift från 1942 då han startade CP-1''. Eklund hade dock den kolossalt stora skillnaden mellan Fermis CP-1 och vår egen R1 klar för sig, då han svarade: ''Nej, ty vi vet att reaktorn kommer att gå''. Den svenska seglatsen över det intellektuella äventyrets ocean var inte den första, men dock en tidig sådan.

Man kan säga att det första året av R1:s drift ägnades åt ett noggrant studium av dess beteendemönster, driftegenskaper och prestanda. Människan utanför AE kunde inte se reaktorns existens på annat sätt än genom existensen av den höga skorstenen vid Drottning Kristinas Väg 51. Reaktortanken omgavs av en reflektor av grafit och denna kyldes genom luftströmmar. Kylluften blåstes ut genom skorstenen. För att kontrollera att denna inte innehöll några otillåtna nivåer av radioaktivt argon och annat, så fanns strålningsdetektorer utplacerade på husen på Röda Korsets sjukhem och uppe på husen sydväst om Östra station. Allt fungerade utmärkt och inga strålningsproblem noterades.

Aptiten växer medan man äter, och vid R1 uppstod önskan att höja effekten. Den ursprungliga effekten av 100 kW höjdes gradvis, och man försökte sig på höjning upp mot 600 kW. Man kunde inte tillåta, att temperaturen på det uppvärmda tunga vattnet steg till mer än ca 40°C, annars skulle tankkonstruktionen och uranstavar kunna komma i fara. Det visade sig, att man inte kunde nå 600 kW utan ca 525 kW som högst,

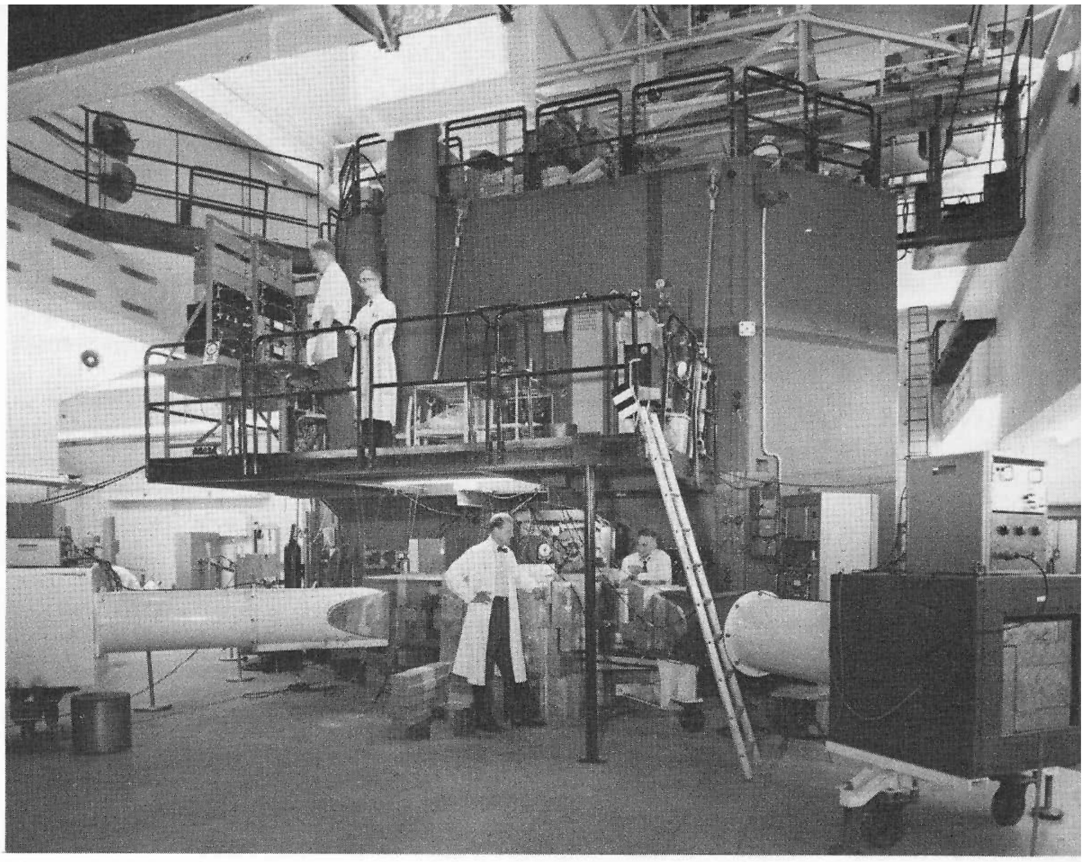


5. Diagram som föreställer hur reaktorns effekt för första gången kördes upp. Kurvan utgör ett mått på kärnreaktionernas intensitet som funktion av tiden. Kl 18.59 inleddes en period av reaktordrift vid konstant effekt. 13 juli 1954. (Studsvik Energiteknik AB, Biblioteket) då lufttemperaturen ute – kylluften – var låg och mindre än 525 kW på sommaren. Uppfinningsrikedomen var emellertid stor under den påhittige driftschefen Nils Berglund, och man fann på att vattenbegjuta värmeväxlaren. Effekten kunde då höjas ytterligare till en effekt av 800 kW. Absolutkalibreringen av reaktoreffekten var ett problem, och man trodde faktiskt under denna tid 1958–1959 att man drev reaktorn vid 1000 kW. Det sista steget i effekthöjning ägde rum 1966, då man ersatte luftkylningen med vattenkylning, lätt vatten kylande tungt vatten. Sedan drevs reaktorn konstant vid en verklig termisk effekt av 1000 kW under resten av sin drifttid.

Reaktorns användning

Hur användes nu reaktorn? Till att börja med för att lära sig hur en reaktor fungerar. Tämligen snabbt utvecklades en isotoptillverkning vid R1, som skedde behändigt genom ett automatiserat rörpostsystem, som konstruktören kunde känna sig stolt över.

Radioaktiva isotoper levererades till sjukhus och forskningsinstitutioner. Reaktorn var naturligtvis landets förnämsta neutronkälla, och från slutet av 1956 utvecklades vid R1 teknik och kunnande i att dra ut strålar av neutroner ur reaktorn och med dessas hjälp undersöka olika sorters beteende av atomer och molekyler i de bestrålade materialen. Här infördes sålunda den s k neutronspridningstekniken i Sverige. Resultat av världsformat producerades från 1957–1960 med rätt små medel. Forskargrupper från KTH, Chalmers och Uppsala Universitet utnyttjade strålningen från R1. Som ett mått på den betydelse kärnenergifältet nu fått så inrättades från 1 juli 1961 vid KTH och även något år tidigare vid CTH professurer i bl a reaktorfysik och kärnkemi. Dessa institutioner, särskilt de i reaktorfysik kom att under hela reaktorns livstid flitigt utnyttja den för undervisning och forskning. Intresset för själva reaktorn avtog emellertid snabbt från 1955, trots att reaktorn var en succé. Anledningen till det sjunkande intresset för R1 från 1955 var, att detta år hölls i september månad den första Genève-konferensen om "Den fredliga användningen av kärnenergi" som ett resultat av bl a president Eisenhowers "Atoms for peace" program. Vid denna konferens presenterades några hyllmeter fakta om kärnreaktorer, fakta, som tidigare varit delvis hemlighållna. Många ekonomer och i ämnet mindre bevandrade tekniker trodde nu, att härmed var vägen öppen för exploatering av den nya energikällan, kärnenergi, i stor och lönsam skala. I vårt land lanserades tankar på inte bara forskningsreaktor av materialtestningstyp, R2 i Studsvik, utan även på värme och elkraftproducerande reaktorer. Statens vattenfallsverk i samarbete med ASEA och den nya industriellt inriktade avdelningen av AE framkastade storvulna planer på reaktorerna R3-Adam och R4-Eva. Denna industriella yra varade dock endast tills man stötte pannan mot den kalla verkligheten: dels var det inte så enkelt tekniskt sett att utveckla högeffektreaktorer, och dels kunde lönsamheten i hela idén ifrågasättas. Detta stod allt klarare efter den andra Genève-konferensen, som avhölls 1958. När det gällde kärnvapenutveckling, så blev det vid denna tid klart, att Sverige inte skulle inveckla sig i framtagandet av ett sådant. Efter 1958 krymptes reaktorerna Adam och Eva till ett enda tungvattenprojekt känt som Ågesta-reaktorn i Stockholm, vilken startades 1963 och levererade



6. Experimentverksamhet vid R1. Bilden visar experimentalapparaturer uppställda framför de olika bestrålningskanaler som genomträngde betongskydd och grafit och gick in mot reaktortanken eller in i densamma. Reaktorn arbetar som strålningskälla för forskning. (Studsvik Energiteknik AB, Biblioteket)

55 MW värme-energi till kringliggande bostadsområdena och 10 MW el-energi till nätet under tio års tid till 1973.

Motstånd mot kärnkraft-utbyggnad

Rent parentetiskt skall noteras, att motstånd mot kärnkraftutbyggnad för första gången dök upp i Sverige, då utbyggnadsplanerna för Ågesta presenterades i början av 60-talet. Dåvarande villaägareföreningen protesterade mot utsläpp i sjön Magelungen. Det skall här emellertid konstateras, att samma villaägareförening såg mycket positivt på reaktorn som värmekälla tio år senare i början av 70-talet och t o m gjorde uttalanden för bibehållande av reaktorn. Den var så mycket renare visavi sin omgivning via utsläpp än ersättningsverket, som var oljeeldat och spydde ut sitt sot och sin svaveloxid över omgivningen. Det var en ödets ironi, att reaktorn lades ned 1973 på grund av olönsamhet, just när oljepriserna började sin raska marsch uppåt.

Omkring årsskiftet 1960–1961 fattades ett annat beslut, som fick betydelse för landets tungvattenlinje, den s k svenska linjen. Då beslöts om uppförande av Marviken-reaktorn, som småningom utvecklades till ett avancerat projekt med s k nukleär överhettning av den producerade ångan för att driva upp verkningsgraden. Reaktorn skulle ha arbetat vid en termisk effekt av mer än 400 000 kW. Som är väl bekant blev konstruktionen alltför invecklad och projektet lades ned efter knappt tio års planering, utveckling, konstruktion och byggnad. Därmed gick den svenska tungvattenlinjen i graven.

Det kan dock nämnas, att man inom denna tid även haft planer på att införa ett antal varmvattenalstrande reaktorer i landet. Även detta projekt skrinlades. I stället utvecklades de el-kraftproducerande lättvattenmodererade reaktorerna med bränsle anrikat på uran-235, som kunde köpas ur USA:s överskottslager. Det var i princip en reaktor, som utvecklats under en amerikansk officers ledning, amiral Rickover, för den amerikanska flottans räkning, som nu landbaserades och utvecklades till en kraftreaktor. I och med att den linjen valdes i Sverige övergavs planerna på vårt framtida energiberoende för en lång tid framåt. Med tungt vatten och naturligt uran finns i Sverige en god energiförsörjning på nationell oberoende-basis. Med lätt vatten och anrikat uran är vi beroende av anrikningstjänster från stormakterna.

R1:s nedläggning

Emellertid ledde den skisserade utvecklingen till att dels minskade AE:s ekonomiska resurser mot slutet av 60-talet och dels var helt allmänt intresset för en tungvattenreaktor i bottenläge. Beslutet om R1:s nedläggning togs av AE 1969 och det kunde trots uppvaktning hos dåvarande utbildningsministern, Olof Palme, inte förhindras. Den 6 juni 1970 sänktes säkerhetsstavarna för sista gången i denna Sveriges första reaktor. Särskilt KTH gjorde därmed en betydande förlust. Genom sitt centrala läge var reaktorn ett mycket värdefullt forsknings- och utbildningsinstrument. Man kan jämföra med Studsvik, som bl a genom sitt isolerade läge aldrig blivit det planerade stora forskningscentret trots sina avsevärda tunga resurser.

Nu tio år senare har R1 åter blivit aktuell. Reaktorhallen har stått oanvänd sedan 1970. Uranstavarna förvaras i Studsvik och det tunga vattnet är sålt. Det kostar pengar att hålla reaktorhallen även tom. Tanken har väckts att riva R1, dvs den järn- och betongkonstruktion med grafit som finns kvar. Radioaktiviteten i materialet runt den gamla reaktorn är nu låg. Så blir då kanske R1 pionjär ytterligare en gång: nu får man öva sig i konsten att riva en uttjänt reaktoranläggning. I samband med debatten om våra kraftreaktorer har ju hanteringen av uttjänta kärnkraftverk varit en diskussionspunkt. R1 kommer tydligen att leverera demonstrationsmaterial. Det har sagts, att rivning skulle kosta 25 miljoner kronor i dag. Uppbyggnad av hela anläggningen kostade ca 20 miljoner kronor 1951–1954. Sedan dess har penningvärdet fallit sex gånger till 1981. Det skulle betyda, att de 25 miljonerna hade varit 4 miljoner 1954, dvs rivningen kostar runt 20 % av hela anläggningskostnaden. Härav kan kanske lärdom dragas för framtiden. Pionjären förblev pionjär till sin förintelse.

Appendix F

EARLY STAGE COST CALCULATIONS FOR DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR RESEARCH FACILITIES

Rolf Sjöblom[1], Staffan Lindskog[2],
Cecilia Sjöö[1] and Anna Cato[2],

1 Tekedo AB
2 Swedish Nuclear Power Inspectorate

Presented at ICEM'05:
The 10th International Conference on Environmental Remediation and Radioactive Waste Management. September 4-8, 2005,
Scottish Exhibition & Conference Centre, Glasgow, Scotland

ABSTRACT

The Storage for Old Intermediate Level Waste (SOILW) at Studsvik has been used for interim storage of intermediate and high level radioactive waste from various activities at the Studsvik site including post irradiation investigations. The SOILW facility was in operation during the years 1961 –1984. The waste was stored in tube positions in concrete blocks and in concrete vaults. In some instances, radioactive debris and liquid has contaminated the storage positions as well as the underlying ventilation space.

The Interim Store for Spent Nuclear Fuel (ISSNF) at Studsvik was built in 1962-64 and has been used since for wet storage of spent fuel from the Ågesta Nuclear Power Plant and the Studsvik R2 research reactor. It comprises three cylindrical pools together with facilities and equipment for handling and decontamination.

In the Swedish finance system, adequate funds need to be accumulated long before (most) decommissioning operations take place. Thus, precise cost calculations are needed already at an early stage of planning.

The primary purpose of the present work is to improve and extend the present knowledge basis for cost estimates for decommissioning, with the SOILW and ISSNF facilities as reference cases. The main objective has been to explore the possibilities to improve the reliability and accuracy of capital budgeting for decommissioning costs. The work has comprised review of previous cost estimates, visits to facilities and information searches.

The following conclusions were made:

- IAEA and OECD/NEA documents provide invaluable advice for pertinent approaches.
- Adequate radiological surveying is needed before precise cost calculations can be made.
- The same can be said about technical planning including selection of techniques to be used.
- It is proposed that separate analyses be made regarding the probabilities for conceivable features and events which could lead to significantly higher costs than expected.
- It is expected that the need for precise cost estimates will dictate the pace of the radiological surveying and technical planning, at least in the early stages.
- It is important that the validity structure for early cost estimates with regard to type of facility be fully appreciated. E g, the precision is usually less for research facilities.
- The summation method is treacherous and leads to systematical underestimations in early stages unless compensation is made for the fact that not all items are included.
- Comparison between different facilities can be made when there is access to information from plants at different stages of planning and when accommodation can be made with regard to differences in features.

- A simple approach is presented for "calibration" of a cost estimate against one or more completed projects.
- Information exchange and co-operations between different plant owners is highly desirable.

BACKGROUND

In the nineteen fifties and sixties, Sweden had a comprehensive program for utilization of nuclear power including uranium mining, fuel fabrication, reprocessing (the plans for reprocessing were never carried out) and domestically developed heavy water reactors. Only one of these was actually taken in operation, the Ågesta reactor, which generated a thermal power of 65 MW of which 10 MW was used for electricity generation and 55 MW for district heating. It was shut down in 1973. The program also included a materials and fuel testing reactor, R2, with light water and heavily enriched uranium fuel. It has a thermal power of 50 MW and is being shut down this year (2005). There is also a hot cell laboratory for post-irradiation investigations still in operation.

The residues from the hot cell laboratory were put in steel boxes which in turn were stored in the Storage for Old Intermediate Level Waste (SOILW). The spent fuel from the Ågesta reactor was kept at the Interim Store for Spent Nuclear Fuel (ISSNF) which is a pool storage comprising three cylindrical concrete tanks.

The development work described above lead to the present nuclear programme comprising 12 modern light water reactors, eleven of which are in operation at present. One more reactor will be taken out of operation this year (2005).

THE SYSTEM FOR FINANCING

The facilities used in the development work described above will need to be decommissioned. It has been decided that it is those who benefit from the electricity generated by the modern nuclear power plants who shall pay the costs for the decommissioning, decontamination, dismantling and waste management which is required when the old research facilities are no longer needed.

Thus, the Law on financing of the management of certain radioactive waste e t c (SFS 1988:1597) states (§1) that *"fee shall be paid to the Government in accordance with this law as a cost contribution*" to amongst other things "*decontamination and decommissioning of*" … "*the Storage for Old Intermediate Level Waste (SOILW)*" ... and ... "*the Interim Store for Spent Nuclear Fuel (ISSNF)*".

The Ordinance (SFS 1988:1598) on financing of the handling of certain radioactive waste e t c states (§4) that the funds collected should be paid to cover the costs incurred. It also states (§4) that "*payment will be carried out only for costs which are needed for" the decontamination and decommissioning "and which have been included in the cost estimates*" required.

According to the Law on financing of the management of certain radioactive waste e t c (SFS 1988:1597, §5), cost calculations shall be submitted to the Swedish Nuclear Power Inspectorate (SKI) each year. They shall comprise estimates of the total costs as well as the costs expected to be incurred in the future with special emphasis on the subsequent three years.

The SKI has the responsibility (SFS1988:1598, §5) to review the cost estimates and to report to the Government if there is a need to change the level of the fee. The SKI also has the responsibility (SFS 1988:1598, §4) to decide on the payments to be made.
It might be added that according to its instruction (SFS 1988:523, §2) SKI also has the responsibility "*in particular ... to take initiative to such ... research which is needed in order for the Inspectorate to fulfil its obligations*".

RATIONALE FOR THE PRESENT WORK

It is thus a solid prerequisite for the responsible planning and management of the decommissioning of the various research facilities concerned that realistic and reliable cost estimates can be made.

The estimates must be based on a sufficiently ambitious program to guarantee that all the pertinent requirements of the society are met. At the same time, unjustified fees should not be levied on the users of the nuclear electricity.

It is actually far from trivial to meet these requirements. It is not unusual that cost estimates be raised each time they are updated as further details become apparent.

Therefore, high requirements apply to cost estimates themselves as well as to the knowledge base on which they rely. In particular, there is a need to identify in what way feedback of experience might be utilized in order to achieve sufficiently robust estimates.

The purpose of the presently reported work is to identify methodology to be used in order to achieve the precision and reliability required. The purpose is also to identify what knowledge might be required in order for such methodology to be successfully applied.

This is achieved by going through two reference cases: the Storage for *Old Intermediate Level Waste* (SOILW) and the *Interim Store for Spent Nuclear Fuel* (ISSNF). Details of these cases can be found in [1] and [2], respectively, and references therein.
Previous cost calculations rely on data on contamination levels, assumptions on methods to be used and on estimates of various volumes of work and waste based on drawings. The methodology applied is similar to that used for nuclear power plants and utilizes a summation type of methodology. The experience from such calculations is that the costs estimated increase with the level of detail, and thus escalate as the work progresses and time passes. The scope thus includes to attempt to identify time and stage invariant methodology.

The work has comprised the following activities:

- To review previous cost estimate reports
- To visit facilities and meet with those responsible
- To carry out various information searches

GUIDANCE DOCUMENTS

The financial planning relies heavily on an appropriate technical planning. Invaluable advice in this regard is provided in IAEA[3-7] and OECD/NEA[8] guidelines and similar.

In these guidelines, so-called "critical decommissioning tasks" are identified. They include the following:

1 *Characterization of the facility.*
   - A survey of the radiological and non-radiological hazards which is used as an input for the safety assessment for decommissioning and for implementing a safe approach during the work.
   - An adequate number of radiation and contamination surveys should be conducted to determine the radionuclides, maximum average dose rates, and contamination levels for inner and outer surfaces throughout the facility.
   - A survey of all hazardous material in the facility.

2 *Removal of the residual process material.*

3 *Decontamination,* including selection of technique with regard to effectiveness and to potential for reducing the total exposure

4 *Dismantling,* including an analysis of each dismantling task and the most effective and safe method to perform it.

5 *Demolition, surveillance and maintenance, and final radiological survey.*

It is also stated that the cost estimate should reflect all activities described in the decommissioning plan, including e g development of specific technology.

It should be recognized, however, that these guidelines - at least for the most part - are issued with regard to the technical planning and its pertinent logistic and timing constraints. This implies that unless a comprehensive view is taken - including the financial planning requirements - data may be insufficient for cost estimates having the precision required as presented above. Therefore, iteration is required between steps 1 - 5 above and the cost estimates. This may well imply that the timing of the technical planning is dictated by the need for sufficient precision in the cost estimates, at least in the early stages of planning.

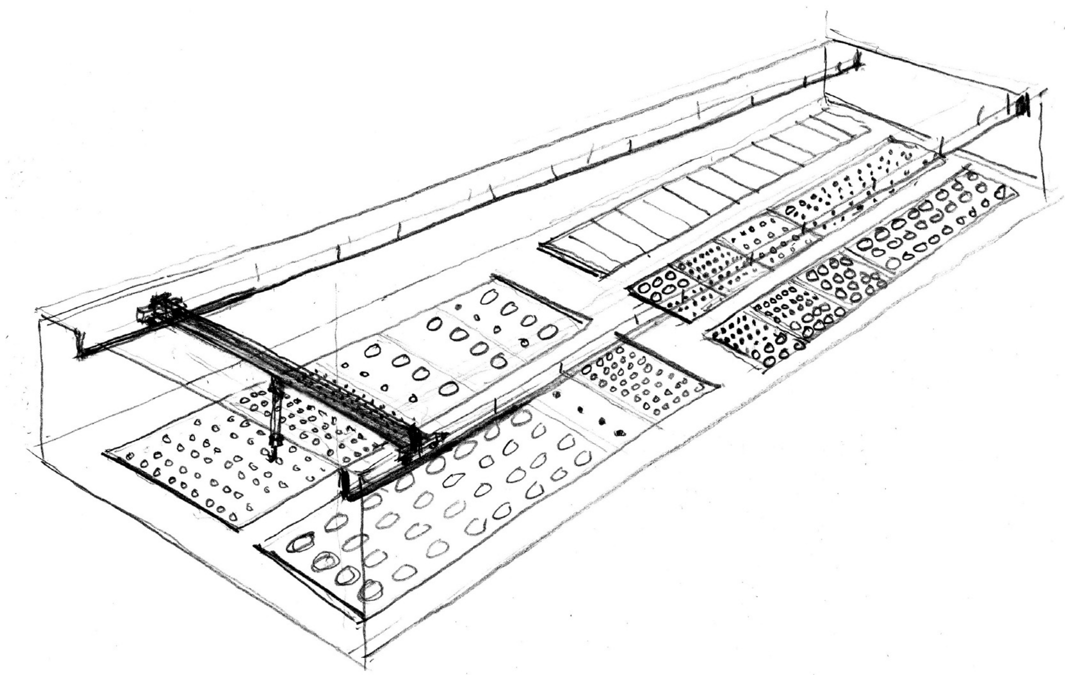


Figure 1. Layout of the Storage for Old Intermediate Level Waste (SOILW) at Studsvik (artist's view). Compartments having large lids are open inside and most compartments with circular lids contain vertical pipes in concrete blocks which are about 3 meters thick.

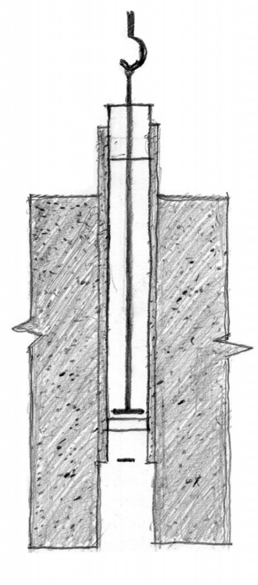


Figure 2. A pipe position in the Storage for Old Intermediate Level Waste (SOILW) at Studsvik showing the removal of a pipe after an overcoring operation (artist's view).

STORAGE FOR OLD INTERMEDIATE LEVEL WASTE

Plant description

The Storage for Old Intermediate Level Waste (SOILW) was commissioned in 1961 and waste emplacement was discontinued in 1984. All waste was removed in 2001.

Further detail on the information below can be found in [1] and references therein.

SOILW has been used for interim storage of intermediate and high level radioactive waste from various activities at the Studsvik site, including test reactor and hot cell laboratory operation. Some of the waste came from outside Studsvik, e g the Swedish Military.

Much of the high level waste originated from fuel tests and subsequent post irradiation investigations. It comprised fuel debris and in some cases also slurry used for polishing of specimens. The material was packed in tins made from sheet metal.

An overview of the SOILW facility is shown in Figure 1. The SOILW comprised a number of storage compartments of two kinds, concrete blocks with vertical pipes for storage of tins as just described and compartments with no internal structures for storage of intermediate level waste of various kinds. At the bottom, the vertical pipes enter into a ventilation area which is about 5 – 10 centimeters high. All storage compartments have thick concrete lids for radiation shielding. The facility has been emptied from radioactive waste but not cleaned. Significant levels of contamination are believed to exist on the surfaces of the vertical pipes and at the bottoms of the compartments.

The handling space above the compartments and the concrete lids is classified as “yellow” which implies that the surface contamination is between 40 and 1000 kBq/m2 for beta plus gamma radiation and between 4and 100 kBq/m2 for alpha.

The dose rates in the compartments with no internal structures are on the order of 0,5 mSv/h which is too high for work by man in situ (except possibly for very limited periods of time).

The dose rate in the pipes used for stacking tins is believed to be high, at least at certain locations. The reason is that the tins contained not only fuel debris but also liquid, supposedly absorbed in vermiculite, containing nitric acid which caused corrosion of the tins as well as leakage and contamination of the pipe shafts. Also, it is known that small objects have dropped down to the ventilation area underneath and possibly caused contamination.

Present plan for decontamination and dismantling

It is assumed that the insides of the vertical pipes are heavily contaminated by leakage from the cans containing the high level waste. Thus, the plan is to decontaminate them by using carbon dioxide jets. It is assumed that all pipes having welds will become clean enough for unconditional release but that those that are spirally welded will not become completely clean in the seams (or on the outside).

It is thus assumed that those pipes that have seams – which comprise the vast majority – need to be removed by core drilling (after decontamination of the inside). The drilling is to be made using a conventional drilling rig and water as coolant and lubricant.

The floor underneath the ventilation space under (most) concrete blocks having vertical pipes is expected to hold surface contamination. Thus, any operation that may involve accessing this area will need special consideration. It is anticipated that some preliminary removal of specimens and vacuum cleaning in this area will take place as a first step. Then, plugs are inserted at the bottoms of the pipes, whereafter the decontamination is carried out of the insides of the pipes. The core drilling is wet only to immediately before penetration, at which stage dry drilling is applied instead. The removal of an overcored pipe position is shown in Figure 2.

After the vertical pipes have been removed – alternatively cleaned completely – the concrete blocks are to be size reduced into pieces which can be handled by the crane which is at most 10 tons.

It is anticipated that the surfaces of the concrete blocks be relatively clean at this stage. A positive factor in this regard is the fact that there is a steel sheet metal plate at the bottom of the blocks. This implies that it may be feasible to clean the bottom surfaces from whatever contamination they might have.

The breaking up of the blocks is intended to be made by means of drilling and mechanical fracturing. Once the blocks have been removed, surfaces become accessible for (further) cleaning and for removal of the contaminated surfaces of concrete. Such cleaning and removal of surface material is also expected to be warranted for those compartments which did not have any interior structures. It is assumed in the report that a surface layer of 3 centimeters will have to be removed by using hand tools.

Regarding level and precision of calculated costs

It is obvious from the above cited guidance documents that a radiological mapping of a facility provides the necessary basis for technical planning and precise cost calculations. Such a survey should include the presence of hot spots, approximate radionuclide distribution and at least to some extent also the penetration depth.

A highly important factor for the cost level and precision is the selection of technology. For large and flat surfaces remotely controlled billing may be preferential to manual billing. If the penetration depth is small (e g less than 5 millimeters) a laser based technique might be considered.

Splitting of blocks using the technique put forward might be difficult due to lack of tools of the appropriate length on the market. Therefore use of expanding concrete might be warranted instead.

The literature survey conducted revealed[9] the existence of a similar but completed project: the East Map Tube Facility at Argonne National Laboratory in Illinois. Further information has been compiled subsequently.[10]

The approach applied was rather similar to that described above for SOILW. The experience is briefly as follows.

A concrete coring rig was used to cut each pipe from the concrete matrix. Each pipe was cut from the structure in one continuous 21 foot long coring operation through solid concrete. To reduce waste quantities, the core diameter was selected to minimize the amount of concrete removed along with the pipe. Careful control of the coring operation was required to prevent the core tool from cutting into the pipe or joint.

It became apparent during the operation that the pipes were not quite vertical in orientation. It was therefore deemed desirable to angle the coring, but attempts to this effect were unsuccessful. Eventually the drilling was carried out strictly vertically using a larger diameter drill.

The coring drill originally used was too light to maintain the orientation of the core and therefore a larger rig had to be brought in.

On several occasions, voids as well as incidental objects were encountered in the concrete. Loss of cooling liquid took place at a number of occasions so that injection of fresh concrete had to be applied.

Small or moderate amounts of activity were transferred to the drill water slurry. However, the potential for such contamination is substantial in an operation of the present kind.

INTERIM STORE FOR SPENT NUCLEAR FUEL

Plant description

The Interim Store for Spent Nuclear Fuel (ISSNF) was commissioned around 1964 and is still in operation. It has been used for the interim storage of spent fuel from the Ågesta nuclear power plant and the R2 research reactor. The former had incidents of severe fuel damage[11] although it appears that at least some of the most damaged fuel was sent to Eurochemic for reprocessing and accordingly never received at ISSNF[12]. The plant comprises a main hall with three cylindrical pools for spent fuel storage and a drained stainless steel surface for decontamination, see Figure3. The insides of the tanks are covered with glass fibre impregnated epoxy which has become deteriorated in patches. The hall also contains an overhead crane and equipment for shielded handling of the fuel.

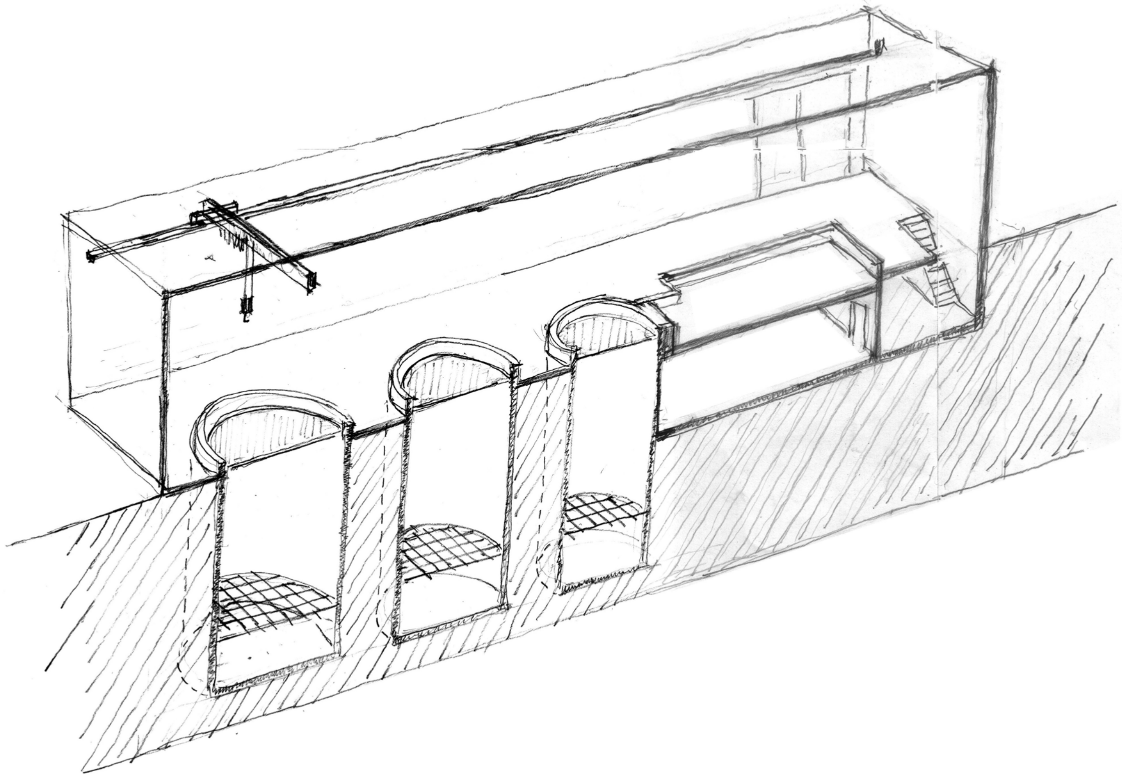


Figure 3. The spent fuel store at Studsvik showing the main hall as well as the interface between the building structrues and the underlying soil and rock (artist's view). (Lifting device shown is not that used for fuel transfer.)

The basement contains equipment for water management including purification.
The handling space above the compartments and the concrete lids is classified as "yellow" which implies that the surface contamination is between 40 and 1000 kBq/m2 for beta plus gamma radiation and between 4 and 100 kBq/m2 for alpha.

The pool water has historically had activity concentrations on the order of MBq/m3. Recent levels of activity concentrations are as follows (in kBq/ m3):

total alpha	< 1,3
total beta	614
Cs-13422	
Cs-13796	

Present plan for decontamination and dismantling

The existing plans give only a general idea of the methodology to be used for the decontamination and dismantling of the ISSNF facility, see [2] and references therein. This might not appear unreasonable in view of the low levels of activity detected.
It is assumed that part of the surfaces of the insides of the concrete tanks will have to be billed. The same is also assumed for part of the concrete surface under the sheet metal in the decontamination unit.

The dismantling of the pipe systems will be made based on dose rates on the outsides of the pipes and components.

Regarding level and precision of calculated costs

The few radiological measurements made can probably form a basis for a reasonable technical approach to the decontamination and dismantling, at least in general terms. However, it is again clear from the guidance documents cited above that a detailed mapping is required in order for a precise cost calculation to be made. In particular, it is important to know the alpha to gamma ratios as well as the present of any contaminated sludge and deposits in the water system.

Since the pool system is old, it does not have the redundancy of barriers against leakage to the surrounding soil that modern systems do. An example the types of events that might take place in an old system is presented in [13] where potentially contaminated water was released to ground and surface waters. One source for this was the foundation drainage from the Oak Ridge Research Reactor which was mistakenly pumped to a storm drain, and the other was a leak to groundwater from underground coolant pipes.

Similarly to the case of SOILW, cost may be strongly affected by the choices made regarding technology as well as unexpected features encountered.

DISCUSSION AND CONCLUSIONS

The above examples illustrate the significance of making appropriate radiological surveying and mapping as well as technology selection before sufficiently precise cost calculations can be performed.

Actually, it may well be the need for cost calculation precision that dictates the comprehensiveness and timing of such activities, at least in the early stages of planning.

Moreover, uncertainty in cost calculations may occur in a manner similar to that of a risk for an accident. Thus some sort of risk assessment may be warranted in which conceivable more severe but presumably less likely cases are identified and their probability characteristics evaluated.

The above presented real cases on completed projects illustrate how unexpected events might come about, and when they do, costs will usually escalate. Such features of the cost probability structure are of particular interest in cases where adequate funds are to be collected long before costs are to be incurred.

It is not necessarily so that an unexpected event has a low probability as might be the case for the hypothetical leak in a fuel storage tank. In the case of the drilling with overcoring, the frequency of deviation was substantial. Many pipes deviated from strictly vertical orientation and 10 out of 129 pipe positions had to be temporarily abandoned and grout injected in the core hole to fill the voids before the overcoring could be completed.

It is therefore proposed that some sort of deviation risk assessment be carried out as a part of the critical decommissioning tasks presented in the guidelines[3-8].

The methodology to be used may well resemble those of ordinary hazard evaluation[14].

Frequently, cost calculations for research facilities are made using calculation tools developed for the case of nuclear power plants. This may be appropriate if the differences in character are fully appreciated and accordingly compensated for. Nuclear power plants are huge facilities with large components that lend themselves to detailed analysis. They also have auxiliary facilities with large volumes of similar equipment where per unit economic data may be applied successfully.

Research and test facilities, on the other hand, are widely different in character. Radionuclide distribution patterns and contamination patterns vary and so do also the technologies that are suitable to apply.

Examples of application of this philosophy can be found in [15].

It should be realized that the precision of cost calculations vary strongly between different types of facilities[16]. Deviations are also more likely to be increases than decreases. Deviations are more likely for unusual projects such as research and/or test facilities. The larger the step in technological development, the greater is the deviation. The main reason for this is that “surprises” are encountered in the process.

In conventional cost calculations for new technical facilities five stages of calculation are identified[17]. In the first stage, predesign cost estimates, the analysis is based mainly on comparison between similar plants and the probable accuracy is typically larger than 30 %. In the last one, contractor’s estimate, the accuracy is perhaps 5 % and the calculation is based on summation over essentially known items.

Application of the summation method at early stages gives rise to systematic errors which lead to underestimated costs since not all items have been identified. Nonetheless, it is not unusual that calculations of costs for research facilities at early stages of planning are carried out using the summation method based on methodology and cost parameters for nuclear power plants. Such an approach will invariably lead to calculated costs that increase for each calculation.

It is therefore highly desirable to somehow “calibrate” results of early estimates against known costs of already completed projects of similar kind. One simple approach to this may be as follows[2].

Let the cost for a plant be given by the equation:

$$K^c = \sum_i p_i \qquad (1)$$

Where
Kc = the total calculated cost
p = cost item, and
i = index for cost item

A fit to actual cost K^a for a completed project can be made using the weighing factors w_i and a scaling factor s according to the following equation:

$$K^a - K^c = s \sum_i w_i p_i \qquad (2)$$

The weighing factors may be obtained by assessment of which items should have a small, intermediate, large or very large influence on the difference between calculated and actual values. For instance, a weighing factor can be given one of the values 1, 2, 4 or 8. The scaling factor can then be calculated using the equation:

$$s = (K^a - K^c) / \sum_i w_i p_i \qquad (3)$$

For a plant for which a refined cost calculation is to be made, the cost items can be calculated first, and then the total cost according to the equation (1) above.

After that, an adjusted calculated total cost can be calculated using the equation:

$$K^{adjusted} = \sum_i (1 + sw_i) p_i \qquad (4)$$

where s and w_i have been derived from a similar reference plant and pi for the plant for which a refined calculation is to be made.

The application of equation (4) implies an improvement compared to a simple over all scaling since differences in the assessed cost structure influences the result.

In view of the need for comparisons between different research and test facilities in different stages of planning and decommissioning, the Swedish Nuclear Power Inspectorate has taken initiative to a now (2005) ongoing project within the framework of the *Nordic Nuclear Safety Research*. The main purpose of the work is to find improved methodology for accurate cost calculations at early stages of planning by preparing guidance documents, by making plant data available to the participants and by establishing a network for communication and co-operation.

ACKNOWLEDGMENTS

The authors wish to express their gratitude to the members of staff at Studsvik and SVAFO who have generously shared their knowledge and expertise on the SOILW and ISSNF facilities.

Any errors or misinterpretations are the sole responsibility of the present authors, however. Figures 1-3 are based on drawings presented in [1-2,9] as interpreted by Fabian Sjöblom, Tekedo AB, who is an architect and has not seen the plants. The sketches are intended to illustrate general principles only and must not be consulted for any technical detail.

REFERENCES

1 Sjöblom R and Lindskog S. *An applied study of the storage for old intermediate level waste at the Studsvik site.* SKI Report 2004:11. The Swedish Nuclear Power Inspectorate, February 2004.

2 Sjöblom R, Sjöö C, Lindskog S and Cato A. *Cost studies on decontamination and dismantling of nuclear facilities* (in Swedish). The Swedish Nuclear Power Inspectorate, in print.

3 *Decommissioning of nuclear power plants and research reactors.* Safety Guide. IAEA safety standard series No WS-G.2.1.6

4 *Decommissioning of Nuclear fuel cycle facilities. Safety Guide.* IAEA safety standard series No WS-G.2.4.7

5 *Decommissioning of medical, industrial and research facilities.* Safety Guide. IAEA safety standard series No WS-G.2.2.8

6 *Predisposal management of radioactive waste including decommissioning.* Safety Requirements. IAEA safety standard series No GS-R-2.

7 *Transition from operation to decommissioning of nuclear installations.* Technical report series 420. IAEA April 2004.

8 *Decommissioning Nuclear Power Plants: Policies, Strategies and Costs. Nuclear Energy Agency,* Organization for Economic Co-operation and development, OECD / NEA, 2003.

9 Moos L P, Wescott J B, Flemming D L, Subbaraman G and Horton P, *Decontamination and dismantlement of the Argonne National Laboratory - East Map Tube Facility.* One of the papers in [15],volume 1, pp 164-167.

10 Varley G. *An applied study on the decontamination and decommissioning of the map tube facility, 317 area*, Argonne National Laboratory, USA. Swedish Nuclear Power Inspectorate, in print.

11 Sandstedt K E and Mogard H. *Operating experience with the Ågesta nuclear power plant.* Proceedings of the fourth international conference on peaceful uses of atomic energy. Jointly sponsored by the United Nations and the International Atomic Energy Agency and held in Geneva 6 - 16 September, 1971.

12 Österlundh C G. Former member of staff at Studsvik. *Private communication*.

13 Stover R L, Anderson G E, Finger J M and Skipper D D *Decontamination and decommissioning preparation of Oak Ridge National Laboratory*. In Editors Gerry White M, Norman N and Thomas R. Decommissioning, Decontamination, and Environmental Restoration at Contaminated Nuclear Sites (DDER-'94), Vol. 1, American Nuclear Society, 1995.

14 *Guidelines for hazard evaluation procedures.* Center for Chemical process safety of the American Institute of Chemical Engineers, 1992. ISBN 0-8169-0491-X.

15 Lauridsen, K. *Assessment of the costs of decommissioning the nuclear facilities at Risø national laboratory.* Nuclear Decom 2001, Ensuring Safe, Secure, and Successful Decommissioning. IMechE Conference Transactions 2001-8. Professional Engineering Publishing, 2001.

16 Segelod E. *Comparison between planed and actual costs for large projects.* Presented at A seminar on Nuclear waste - costs and financing held by the Swedish National Council for Nuclear Waste at Gimo Herrgård, Östhammars kommun on February 16th, 2005. (Can be obtained from siv.milton@sustainable.ministry.se).

17 Peters M S and Timmerhaus K D. *Plant design and economics for chemical engineers*, 4th edition. McGraw-Hill, Inc., New York 1991. ISBN 0-07-049613-7.

Title	Cost Calculations for Decommissioning and Dismantling of Nuclear Research Facilities
Author(s)	Inga Andersson[1], Steinar Backe[2], Anna Cato[3], Henrik Efraimsson[4], Klaus Iversen[5], Staffan Lindskog[3], Seppo Salmenhaara[6] and Rolf Sjöblom[7]
Affiliation(s)	[1] Studsvik Nuclear AB, Sweden [2] Institute for Energy Technology, Norway [3] Swedish Nuclear Power Inspectorate, Sweden [4] Swedish Radiation Protection Authority, Sweden [5] Danish Decommissioning, Denmark [6] VTT Technical Research Centre of Finland, Finland [7] Tekedo AB, Sweden
ISBN	978-87-7893-230-3
Date	July 2008
Project	NKS-R / CostCalc
No. of pages	Vol I: 119, Vol II: 83
No. of tables	Vol I: 5, Vol II: 5
No. of illustrations	Vol I: 19, Vol II: 13
No. of references	85

Abstract

Today, it is recommended that planning of decommission should form an integral part of the activities over the life cycle of a nuclear facility (planning, building and operation), but it was only in the nineteen seventies that the waste issue really surface. Actually, the IAEA guidelines on decommissioning have been issued as recently as over the last ten years, and international advice on finance of decommissioning is even younger. No general international guideline on cost calculations exists at present.

This implies that cost calculations cannot be performed with any accuracy or credibility without a relatively detailed consideration of the radiological prerequisites. Consequently, any cost estimates based mainly on the particulars of the building structures and installations are likely to be gross underestimations.

The present study has come about on initiative by the Swedish Nuclear Power Inspectorate (SKI) and is based on a common need in Denmark, Finland, Norway and Sweden.

The content of the report may be briefly summarised as follows. The background covers design and operation prerequisites as well as an overview of the various nuclear research facilities in the four participating countries: Denmark, Finland, Norway and Sweden.

The purpose of the work has been to identify, compile and exchange information on facilities and on methodologies for cost calculation with the aim of achieving an 80 % level of confidence.

The scope has been as follows:
• to establish a Nordic network
• to compile dedicated guidance documents on radiological surveying, technical planning and financial risk identification and assessment
• to compile and describe techniques for precise cost calculations at early stages
• to compile plant and other relevant data

A separate section is devoted in the report to good practice for the specific purpose of early but precise cost calculations for research facilities, and a separate section is devoted to techniques for assessment of cost.

Examples are provided for each of the countries of relevant projects. They are as follows:
• Research reactor DR1 in Denmark
• The TRIGA research reactor in Finland
• The uranium reprocessing plant in Norway
• Research reactor R1 in Sweden

The following conclusions were made:
• IAEA and OECD/NEA documents provide invaluable advice for pertinent approaches.
• Adequate radiological surveying is needed before precise cost calculations can be made.
• The same can be said about technical planning including selection of techniques to be used.
• It is proposed that separate analyses be made regarding the probabilities for conceivable features and events which could lead to significantly higher costs than expected.
• It is expected that the need for precise cost estimates will dictate the pace of the radiological surveying and technical planning, at least in the early stages.
• It is important that the validity structure for early cost estimates with regard to type of facility be fully appreciated. E g, the precision is usually less for research facilities.
• The summation method is treacherous and leads to systematical underestimations in early stages unless compensation is made for the fact that not all items are included.
• Comparison between different facilities can be made when there is access to information from plants at different stages of planning and when accommodation can be made with regard to differences in features.
• A simple approach is presented for “calibration” of a cost estimate against one or more completed projects.
• Information exchange and co-operations between different plant owners is highly desirable.

Key words decommissioning, cost calculations, nuclear, research facilities, fund, nordic

---

Available on request from the NKS Secretariat, P.O.Box 49, DK-4000 Roskilde, Denmark.
Phone (+45) 4677 4045, fax (+45) 4677 4046, e-mail nks@nks.org, www.nks.org