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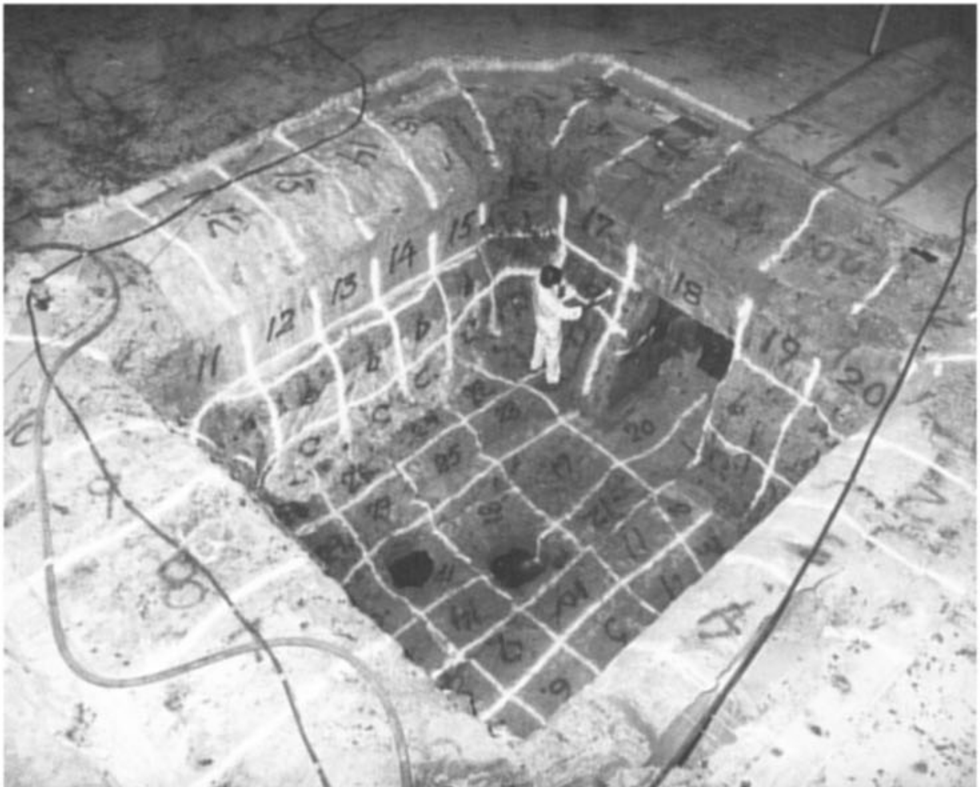
SOME STUDIES RELATED TO DECOMMISSIONING OF NUCLEAR REACTORS



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SOME STUDIES RELATED TO

DECOMMISSIONING OF NUCLEAR REACTORS

Report from NKA-Project KAV 350

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The cover picture shows the final measurements after dismantling of the biological shield in the R1 research reactor in Stockholm (photo Studsvik Nuclear).

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ABSTRACT

Decommissioning of large nuclear reactors has not yet taken place in the Nordic countries. Small nuclear installations, however, have been dismantled. This NKA-programme has dealt with some interesting and important factors which have to be analysed before a large scale decommissioning programme starts.

Prior to decommissioning, knowledge is required regarding the radionuclide inventory in various parts of the reactor. Measurements were performed in regions close to the reactor tank and the biological shield. These experimental data are used to verify theoretical calculations.

All radioactive waste generated during decommissioning will have to be transported to a repository. Studies show that in all the Nordic countries there are adequate transport systems with which decommissioning waste can be transported.

Another requirement for orderly decommissioning planning is that sufficient information about the plant and its operation history must be available. It appears that if properly handled and sorted, all such information can be extracted from existing documentation.

Key words: decommissioning, radioactive waste, radionuclide, transport, radionuclide inventory, documentation, nordic cooperation

SUMMARY

The life span of nuclear power reactors is governed by safety related factors and by economic considerations. World wide, several hundred reactors are estimated to have reached the end of their working lives by the year 2025. In the Nordic countries, each country has its own policy in this area. In Sweden, Acts of Parliament require all the 12 reactors in service to be taken out of operation no later than the year 2010. The first two reactors are supposed to be stopped in 1995 and 1996. According to the plans of the reactor owners the dismantling will not start until after 2010. In Finland, there are no plans for phasing out nuclear power but the nuclear utilities have had to present decommissioning plans with cost estimates for the four reactors presently in operation. There are no nuclear power reactors in operation or being planned in Denmark or Norway, but research reactors are in operation in both countries.

The decommissioning process for reactors can vary from case to case. For environmental and public safety reasons, an early dismantling of a station would be advantageous. However, these advantages must be weighed against the higher radiation doses to the dismantling workers and the larger quantities of radioactive waste that could be a consequence of early dismantling. Even other factors such as the availability of a final repository for radioactive waste can affect the decommissioning process. The decommissioning of a plant can take place in several steps. Three different stages of decommissioning can be defined, where "stage 3" is the most far-reaching, involving complete removal of all radioactive material.

Considerable research and development efforts have been made in the field of decommissioning of nuclear installations. Field experience has been on decommissioning of early, small-scale plants in many countries including the Nordic countries. In Sweden the research reactor R1 in Stockholm (see cover picture) as well as some zero-power facilities and laboratories at Studsvik have been totally dismantled (to "stage 3") and the Ågesta reactor outside Stockholm is decommissioned to "stage 1". The DR2 research reactor at Risö in Denmark, the JEEP 1 and NORA research reactors at Kjeller in Norway have all been decommissioned to "stage 2" and for the fuel reprocessing pilot plant at Kjeller the dismantling has started.

No full size nuclear power plants have up to now been decommissioned and thus no experience of the real costs exists, but from the small size reactors decommissioned some estimations also for full size reactors can be made. In many countries the utilities have to pay fees to build up funds to cover future decommissioning costs. Especially in those cases have the cost calculations been done with great care and have also been scrutinized by independent reviewers. This has been the case in Finland and Sweden. The calculations show that the

total cost for decommissioning is roughly 10 % of the construction cost for a nuclear power plant.

Before decommissioning of a nuclear installation starts the whole sequence of actions leading to final dismantling must be analysed. Such a sequence involves activities of technical, economic, legal and social character.

Thus, prior to decommissioning, knowledge is required regarding the radioactive inventory in various parts of a reactor. In order to verify the theoretical calculations that can be made, measurements were performed in regions close to the reactor tank and the biological shield in the Swedish Oskarshamn 1 reactor. Hereby an experimental basis has been obtained to permit evaluation of the calculational methods that are currently available.

Another requirement for orderly decommissioning planning is that sufficient information must be available in a well documented form about the plant and its history up to the time where decommissioning is to start. It appears that if properly handled and sorted, all such information can be extracted from documentation already in existance at the plants.

The waste from decommissioning that must be handled as being radioactive needs to be transported and disposed of safely. Its volume is roughly equivalent to what is generated during the life time of the plant. It was found that transportation of decommissioning waste can be done with the systems in existance in the Nordic countries.

Internationally, substantial work is under way regarding techniques for decommissioning. In the Nordic countries, the nuclear utilities as well as the safety authorities join this work and hereby benefit from the development work during the period until decommissioning dates have been fixed.

SAMMANFATTNING

Livslängden för en kärnkraftreaktor bestäms av säkerhetsrelaterade faktorer och av ekonomiska överväganden. I världen som helhet uppskattas några hundra reaktorer ha uppnått slutet av sin livslängd år 2025. I de nordiska länderna varierar policyn avseende hur lång tid reaktorerna bör drivas. I Sverige skall samtliga 12 reaktorer som nu är i drift stängas av före 2010. De två första reaktorerna avses tas ur drift 1995 och 1996. Enligt industrins planer skall rivningen påbörjas först efter 2010. I Finland finns inte några planer på att avveckla kärnkraftreaktorerna. Industrin har däremot varit ålagd att presentera nedläggningsplaner inkluderande kostnadsuppskattningar för de fyra reaktorerna som är i drift idag. Varken Danmark eller Norge har kärnkraftreaktorer eller planerar för sådana. Däremot finns forskningsreaktorer i drift i bägge dessa länder.

Strategin för nedläggning av reaktorer kan vara olika från fall till fall. Utifrån miljöhänsyn och skydd av allmänheten så är en tidig rivning av anläggningen att föredra. Fördelarna måste emellertid vägas mot de högre stråldoserna till den personal som utför rivningen och de större avfallsmängder som kan bli en följd av en tidig rivning. Också faktorer som tillgången till slutlig förvaringsplats för det radioaktiva avfallet kan påverka hur rivningen genomförs. Nedläggningen kan äga rum i flera steg. Ofta talas om tre steg, där "steg 3" är det mest långtgående. Detta innebär att allt material avlägsnats från reaktorområdet.

Omfattande FoU-insatser har redan utförts inom nedläggningsområdet. Erfarenhet från nedläggning av små anläggningar (forskningsreaktorer) finns i många länder, också i de nordiska länderna. I Sverige har forskningsreaktorn R1 i Stockholm (se omslagsbilden) samt några noll-effekt reaktorer och laboratorier i Studsvik genomgått en fullständig rivning (till "steg 3"). Ågestareaktorn utanför Stockholm har rivits till "steg 1". Forskningsreaktorn DR2 på Risø i Danmark, forskningsreaktorerna JEEP 1 och NORA vid Kjeller i Norge har samtliga rivits till "steg 2". Rivningen av en pilotanläggning för bränsleupparbetning vid Kjeller har nyligen påbörjats.

Nedläggning av en kraftproducerande reaktoranläggning har ännu inte ägt rum och därför saknas också erfarenhet av den verkliga kostnaden för en sådan rivning. En uppskattning av kostnaden kan dock göras utifrån erfarenheter som erhållits från nedläggning av mindre reaktorer. I många länder måste anläggningsägarna betala in avgifter till fonder avsedda att täcka kostnaderna för framtida nedläggningar. I dessa fall har noggranna kostnadsuppskattningar gjorts vilka också granskats av oberoende kontrollorgan. Så är fallet i Finland och Sverige. Beräkningarna visar att den totala kostnaden för en nedläggning av en kärnkraftreaktor är cirka 10 % av anläggningskostnaden.

Innan en anläggning börjar rivas så måste den totala nedläggningsinsatsen analyseras. En sådan analys omfattar frågor av teknisk, ekonomisk, juridisk och social natur.

Före rivningen behövs kunskaper om förekomsten av radioaktiva ämnen i olika reaktorsystem. För att verifiera teoretiska beräkningar så har mätningar utförts nära reaktortanken och det biologiska skyddet på den svenska reaktorn Oskarshamn 1. Härigenom har ett experimentellt underlag tagits fram som medger en utvärdering av de beräkningsmetoder som för närvarande används.

För att kunna genomföra en nedläggning på ett tillfredställande sätt så måste det också finnas tillräcklig information, väl dokumenterad, om anläggningen och dess drifthistoria. En genomförd studie visar att sådan information tycks finnas vid anläggningarna men att det kan krävas särskilda insatser för att ordna denna i en lämplig form.

De radioaktiva nedläggninsavfallet måste också kunna transporteras och förvaras på ett säkert sätt. Volymen nedläggningsavfall motsvarar ungefär det avfall som produceras under anläggningens livstid. En studie visade att transporter av nedläggningsavfallet kan genomföras med de transportsystem som idag är i bruk i de nordiska länderna.

Avsevärda insatser genomförs internationellt för att ta fram den teknik som behövs för en nedläggning. Industri och myndigheter i de nordiska länderna följer och drar nytta av detta utvecklingsarbete.

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SOME STUDIES RELATED TO DECOMMISSIONING OF NUCLEAR REACTORS.

1. Introduction

In the Nordic programme for 1985-1989 the studies related to decommissioning of nuclear reactors have been focused on estimation of the radionuclide inventory in various parts of the reactor. But work have also been made on documentation, transport of decommissioning waste, and a case study.

The goal of the study on radionuclide inventory is to verify and validate the methods used to estimate the inventory. It has been done in many steps as illustrated in Fig 1. This report summarizes the following reports:

- Documentation Required for Decommissioning of a Nuclear Power Plant (ref 1)

- Radionuclides Important in the Decommissioning of Nuclear Power Plants (ref 2)

- Activity Inventory in Reactor Systems at End of Operation of Nuclear Power Plants; A Prestudy (ref 3)

- Transport of Decommissioning Waste (ref 4)

- Gammaspectrometric Measurements on Foils and Cement Specimens at the Oskarshamn 1 Reactor (ref 5)

- Complementary Dose Rate and MADAC Measurement at Oskarshamn 2 in 1989 (ref 6)

- Decommissioning of a Small Nuclear Reactor; A Model Study (ref 7)

No full size nuclear power plants have up to now been decommissioned and thus no experience of the real costs exists, but from the small size reactors decommissioned some estimations of costs also for full size reactors can be made.

In a Swedish study, which was made by the Swedish Nuclear Fuel and Waste Management Company (SKB) and which has been approved by the authorities, the dismantling costs was estimated to slightly more than 8 000 million SEK (1 million SEK ~ 0.15 million USD) for the 12 reactors. In addition to that another 1 000 million SEK was calculated for the closing

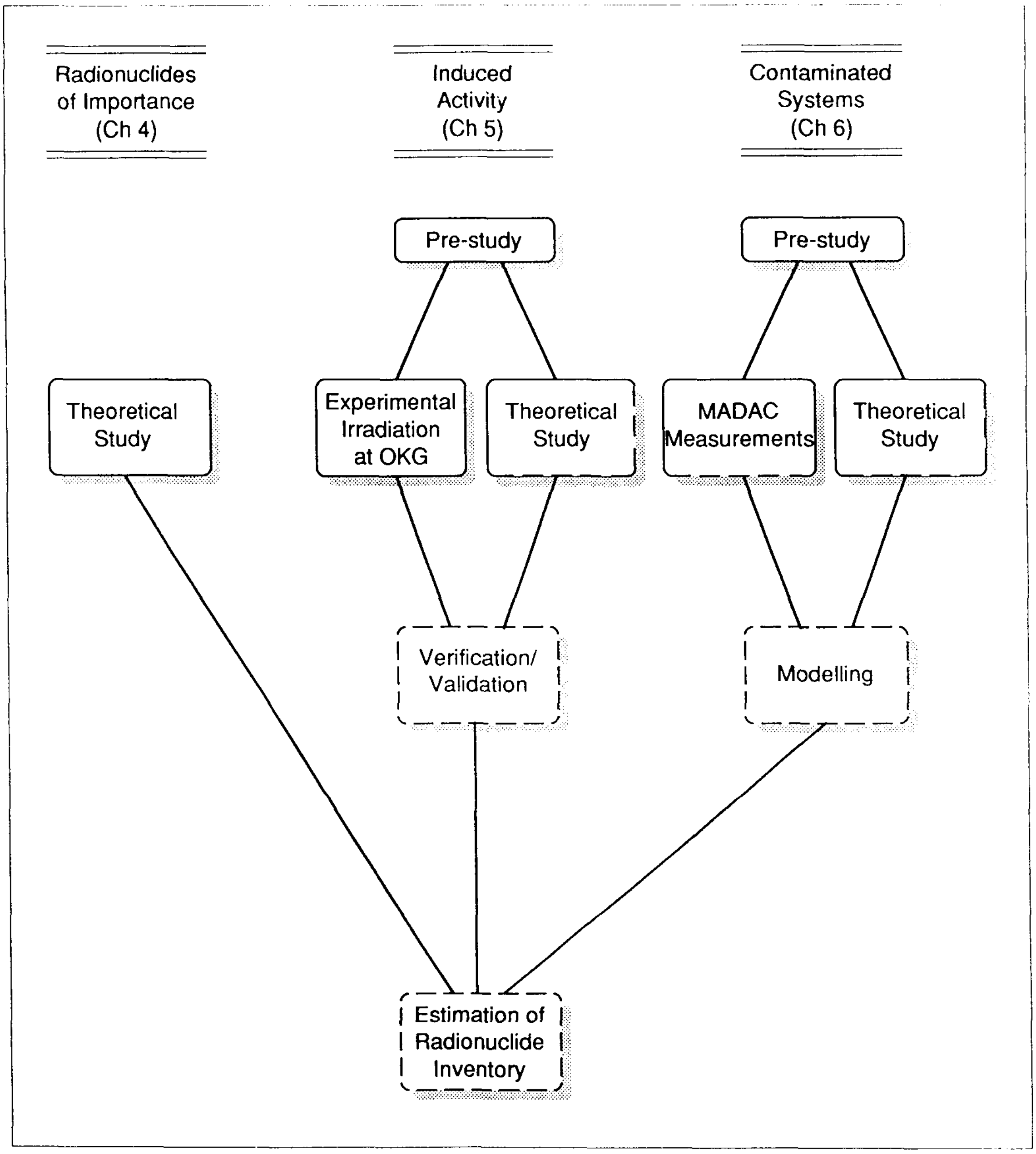


Fig. 1. Studies to be done for accurate estimation of radionuclide inventory. The studies in dashed frame are not finished within this NKA programme.

down operation and surveillance until dismantling is started and 500 million SEK for disposal of the decommissioning waste. In total the cost for decommissioning is roughly 10 % of the construction costs for the plant.

In Finland, Imatran Voima Oy in 1987 reported a detailed plan and cost estimate on decommissioning of the Loviisa power plant and the disposal of the decommissioning waste (ref 8). The total cost estimate for dismantling the two units (2 x 445 MW, PWR) is about 900 million FIM (1 million FIM ~ 0.22 USD) in the 1989 price level. Teollisuuden Voima Oy is preparing a similar plan for the Olkiluoto power plant to be presented in 1990.

As a background, this summary report starts with a brief overview of some other research and development work in the Nordic countries and some other significant projects elsewhere in the world.

2. R & D on Decommissioning

The Finnish power industry has, according to the Finnish Nuclear Energy Act, the responsibility to safely dismantle the reactors after they have been taken out of service. The utilities have studied methods for dismantling activated and contaminated systems. They have established the waste quantities and designed repositories for wastes. Studies on radionuclide inventories in the Olkiluoto and Loviisa power plants (ref 9 and 10) have also been made.

In Sweden SKB is, on behalf of the Swedish nuclear power industry, studying questions related to decommissioning. So far the studies have been limited to conceptual decommissioning cost and technology studies (ref 11). The SKB, however, is actively following international work in the area e.g. in a large OECD/NEA (Organisation for Economic Cooperation and Development, Nuclear Energy Agency) programme mentioned below. The SSI has initiated an overview of the R & D work needed in the near future in Sweden which is to be performed in cooperation with other authorities and the industry.

Since Denmark is member of the European Community, Denmark has full access to the CEC (Commission of the European Communities) work in this field although the work done in Denmark is fairly small. Also in Norway the R & D on decommissioning is limited, but some work is in progress and also there is an increased interest in these questions.

Within the framework of the IAEA, work is going on to produce an integrated data base that would systematically cover the technical, legal and administrative questions that could arise in connection with decommissioning. For this purpose, committees of experts have been set up to collate information and data and, thereafter, issue guidelines, recommendations and technical reports. The radiation protection and nuclear safety authorities of the various Nordic countries participate in some of this work.

IAEA technical reports and recommendations are issued after long discussions and thorough examination in the technical expert committees. A technical report can take between 3 to 4 years between the first committee meeting and its issuing. These reports are however very useful as reference material. The latest technical developments are, however, rarely covered. Examples of IAEA publications in the field are given in ref 12-18.

Within the European Community, 17 nuclear plants have already been decommissioned. This number is expected to reach 50 by the year 2000. The Community is currently coming to the end of its second five-year programme of research on decommissioning. The first programme was reported at a conference in Luxemburg in 1984 (ref 19) and the second five-year programme in a conference in Brussels in October 1989. A third five-year programme is being launched.

One aim of the programme is to develop a common European Community policy on decommissioning. The programme with a budget of 90 million SEK has the following projects:

- Long term integrity of buildings and systems.
- Decontamination for decommissioning purposes.
- Dismantling techniques.
- Treatment of specific waste material such as steel, concrete and graphite.
- Large containers for radioactive waste.
- Estimation of quantities of radioactive waste.
- Influence of design features on decommissioning.

The programme also has a group of projects where new techniques are beeing tested on a large scale under real conditions.

Within the OECD Nuclear Energy Agency many projects related to decommissioning have been run, e.g. a feasibility study (ref 20). An important programme was established in 1985 for the exchange of scientific and technical information from a number of on-going major decommissioning projects. Ten countries participate in the programme which covers fourteen projects. Although not participating with a decommissioning object, Sweden, represented by SKB, has accepted the coordinating function and is hereby guaranteed access to all results. The reactors being decommissioned are of many different types: light and heavy water cooled as well as gas and sodium cooled. The decommissioning of these reactors is also to varying stages: some to stage 1, others to stage 2; four of the reactors will be totally dismantled and reach stage 3. The OECD/NEA programme also includes some fuel reprocessing facilities. Major research and development work is being carried out in connection with some of the projects in the programme. A list of the participants and decommissioning objects is shown in Table 1. Status reports from the programme are published regularly (ref 21).

Table 1. Participating projects in the OECD/NEA CPD-project.

Facility	Type	Operation	Decommissioning option*
1 Eurochemic Reprocessing Plant	Reprocessing of fuel	1966-74	
2 Gentilly-1, Canada	Heavy water moderated/boiling light water cooled prototype reactor	1967-82	Stage 2
3 NPD	PHWR CANDU prototype	1962-87	Variant of stage 1
4 Rapsodie, France	Experimental sodium cooled fast reactor	1967-82	Stage 2
5 G2, France	Electricity and nuclear materials production	1958-80	Stage 2
6 AT1, France	Pilot reprocessing plant for FBR	1969-79	Stage 3
7 Kernkraftwerk Neider-aichbach (KKN), Federal Republic of Germany	Gas-cooled heavy water moderated	1972-74	Stage 3
8 Kernkraftwerk Lingen, Federal Republic of Germany	BWR (with super-heater)	1968-77	Stage 1
9 Garigliano, Italy	BWR (dual-cycle)	1964-78	Stage 1 for main containment
10 Japan Power Demonstration Reactor (JPDR), Japan	BWR, research	1963-76	Stage 3
11 Windscale Advanced Gas Cooled Reactor, United Kingdom	AGR	1962-81	Stage 3
12 BNFL, Co-precipitation plant	Production of mixed Pu and UO_2 fuel	1969-76	Stage 3
13 Shippingport, United States of America	PWR	1957-82	Stage 3
14 West Valley Demonstration Project, United States of America	Reprocessing plant for LWR fuel	1966-72	

3. Documentation for Decommissioning

For the effective planning and execution of the decommissioning of a nuclear power plant, a wide range of data and documentation is a pre-requisite. An NKA study on this subject (ref 1) has identified the following main items important for documentation:

- A detailed description of the plant together with up-to-date drawings. This should include such details as surface finishes and concrete reinforcement.
- Inventory of radionuclides and dose-rates.
- Operational history (power levels, experiences during operation/maintenance, incidents).
- Inventory of waste.
- Regulatory requirements (including activity limits for unrestricted release of material).
- Experiences from other decommissioning projects.

One main finding of the study was that no new information needs to be produced for the decommissioning of a plant. The information that is necessary is in existence at the operators of the Nordic reactors but needs to be sorted, analysed and rearranged for producing the documentation for the planning, licensing and execution of decommissioning.

A significant part of the costs of decommissioning is for special dismantling tools and operations. The most important document in the above listing is therefore the detailed technical description of the plant. The normal operational documentation available at start-up complemented by selective reports from later in its life can serve as a useful basis for this technical description.

The NKA report was circulated for comments from other bodies and institutions in the Nordic countries. Some of these comments are summarized below:

- Documentation and experiences in connection with the replacement of large components during the service life of the reactor should be very useful.

- The availability of correct and updated documentation takes on a special significance in case the dismantling does not take place soon after the end of operation. If the dismantling is carried out after a long period of dormancy, operational staff with an intimate knowledge of the site will no longer be available. In such cases all planning must be based exclusively on documents.

- The documentation should be selective so that information relevant for decommissioning is stored separately. Otherwise, the volume of information could make it cumbersome and difficult to manage.

4. Radionuclides of significance for Reactor Decommissioning

In an NKA-report on this subject (ref 2), an attempt has been made to identify all possible radionuclides that can be produced during the lifetime of the reactor and to determine their significance for:

- occupational exposure during decommissioning
- population exposure due to migration of nuclides relevant for long term exposure from a decommissioning waste repository.

In the first case, it is the relatively short-lived gamma-emitting nuclides that determine the doses to the decommissioning staff if dismantling takes place within about 10 years of end of operation. The "dormancy" period between shut-down of nuclear operations and dismantling can vary widely depending on national policies in waste management. In most cases it is probable that dismantling will take place from five to fifty years after shut-down. However, the question of timing is complicated and many factors influence the choice between immediate and deferred dismantling after shut-down of a reactor. Some of these are:

- need for the land
- decay of the radionuclides (resulting in lower doserates and lower quantity of radioactive waste)
- cost of waste management
- availability of competent personnel
- technology development
- status of auxiliary systems
- availability of financial funds
- burden on future generations

A more thorough discussion of these aspects can be found in ref 22 and 23.

Population exposure due to nuclide migration from a repository to the biosphere is governed by the long lived nuclides in the waste.

In the study a computer programme (ACTDECOM) was written to cover all possible activation reactions.

The programme output gives:

- names and half-lives of nuclides
- activation reaction
- critical concentration

The operational history of the LWR in the study was: 0.8 years of full power followed by 0.2 years at zero power and this sequence for a period of 40 years. After this two dormancy periods were studied, one of 5 years and one of 30 years.

Calculations have been performed for evaluating the significant nuclides during the decommissioning (occupational exposure situation) and the disposal (long term exposure situation) periods respectively. Calculations were performed for the following material:

- core component (stainless steel)
- pressure vessel cladding (stainless steel)
- pressure vessel wall (carbon steel)
- biological shield (concrete)

For the 30 years "dormancy period", an additional calculation was carried out for the reinforcement steel in the biological shield.

4.1 Presented results

In the report (ref 2) the main results from the four cases mentioned above are given in table form referring to a 5 year (relevant to the occupational exposure situation) and 30 year (relevant to long term exposure) cooling time, respectively. Examples of results from the study are given in Tables 2 and 3. Here the main reaction producing each radionuclide is shown together with the half-life of the radionuclide. In each case a principal activity is calculated which is dominating in that material. Co-60 and Ni-59 were chosen as indicated in the respective tables. The critical concentration C_c is the concentration needed to arrive at 1 % of the principal activity. The reference concentration C_r is a typical concentration of the target element in that specific material. The concentrations are given in parts per million (ppm). In the tables only γ-emitting nuclides are included. If more than 10^6 ppm of the target element was needed to produce the critical activity of the radionuclide, the nuclide has been omitted. If the calculated critical concentration is larger than the reference concentration by a factor of 10 it has been marked with a circle ("o") in the comment column, and if the critical concentration is larger than 1000 ppm and no reference concentration is given, it has been marked with an asterisk ("*"). If a radionuclide has no mark in the comment column, it is

Table 2. Biological shield (concrete), 5 year cooling time (for explanation see 4.1). Principal activity: 2.3×10^7 Bq/kg of ^{60}Co.

Nuclide	Reaction	$t_{1/2}$	c_c	c_r	Comment
Cl-36	Cl-35 (n,γ)	3×10^5	790	493	
Sc-46	Sc-45 (n,γ)	83.9 d	3.6×10^5	10.9	o
Zn-65	Zn-64 (n,γ)	243.8 d	1.8×10^3	75	o
Se-75	Se-74 (n,γ)	120 d	3.4×10^5	0.92	o
Se-79	Se-78 (n,γ)	6.5×10^4	2.0×10^5	0.92	o
Nb-94	Nb-93 (n,γ)	2×10^4	3.7×10^3	24.9	o
Rh-102m	Rh-103 (n,2n)	2.89	1.1×10^5	-	*
Ag-108m	Ag-107 (n,γ)	127	52	0.8	o
Ag-110m	Ag-109 (n,γ)	250.4 d	250	0.8	o
Cd-109	Cd-108 (n,γ)	453 d	8.8×10^3	0.3	o
Cd-113m	Cd-112 (n,γ)	13.6	23	0.3	o
Sn-119m	Sn-118 (n,γ)	250 d	5.6×10^4	7	o
Sn-121m	Sn-120 (n,γ)	50	61	7	
Sb-125	Sn-124 (n,γ)	2.6	120	7	o
Cs-134	Cs-133 (n,γ)	2.05	1.4	5.0	
Ba-133	Ba-132 (n,γ)	7.5	1.8×10^3	950	
Pm-145	Sm-144 (n,γ)	18	2.9×10^3	8	o
Sm-151	Sm-150 (n,γ)	87	5.0	8	
Eu-150m	Eu-151 (n,2n)	5	4.1×10^4	1.18	o
Eu-152	Eu-151 (n,γ)	12.5	5.9×10^{-3}	1.18	+

Table 2. Continued.

Nuclide	Reaction	$t_{1/2}$	c_c	c_r	Comment
Eu-154	Eu-153 (n,γ)	16	0.072	1.18	+
Eu-155	Sm-154 (n,γ)	1.7	58	8	
Gd-153	Gd-152 (n,γ)	120 d	810	-	
Tb-157	Dy-156 (n,γ)	150	1.3×10^3	2.3	o
Tb-158	Tb-159 (n,2n)	150	7.2×10^4	0.65	o
Dy-159	Dy-158 (n,2n)	144 d	7.4×10^5	2.3	o
Ho-166m	Ho-165 (n,γ)	1200	47	0.9	o
Tm-170	Tm-169 (n,γ)	127 d	1.7×10^3	-	*
Tm-171	Er-170 (n,γ)	1.9	72	-	
Lu-174	Lu-175 (n,2n)	3.6	2.0×10^4	0.42	o
Lu-177m	Lu-176 (n,γ)	155 d	4.6×10^4	0.42	o
Ta-182	Ta-181 (n,γ)	115 d	2.2×10^4	0.85	o
Ir-192	Ir-191 (n,γ)	74.2 d	7.7×10^5	0.033	o
Ir-192m2	Ir-191 (n,γ)	650	21	0.033	o
Ir-194m	Ir-193 (n,γ)	171 d	250	0.033	o
Tl-204	Tl-203 (n,γ)	3.9	8.4		

Table 3. Core component (stainless steel) 30 year cooling time (for explanation see 4.1).
Principal activity: 2.3×10^{10} Bq/kg of ^{59}Ni.

Nuclide	Reaction	$t_{1/2}$	c_c	c_r	Comment
H-3	Li-6 (n,α)	12.26	2.9×10^{-3}	0.13	+
Cl-36	Cl-35 (n,γ)	3×10^5	220	70	
Fe-55	Fe-54 (n,γ)	2.6	2.1×10^4	6.85×10^5	+
Co-60	Co-59 (n,γ)	5.26 d	1.5	2000	+
Ni-63	Ni-62 (n,γ)	120	18	1.0×10^5	+
Se-79	Se-78 (n,γ)	6.5×10^4	5.5×10^4	200	o
Nb-94	Nb-93 (n,γ)	2×10^4	1.0×10^3	160	
Mo-93	Mo-92 (n,γ)	100	200	2600	+
Tc-99	Mo-98 (n,γ)	2.1×10^5	2.7×10^5	2600	o
Rh-102m	Rh-101 (n,γ)	2.89	8.6×10^5	-	*
Pd-107	Pd-106 (n,γ)	7×10^6	1.4×10^5	-	*
Ag-108m	Ag-107 (n,γ)	127	17	2	
Cd-113m	Cd-112 (n,γ)	13.6	23	-	
Sn-121m	Sn-120 (n,γ)	50	24	100	
Sb-125	Sn-124 (n,γ)	2.6	2.7×10^4	100	o
Cs-134	Cs-133 (n,γ)	2.05	1.8×10^3	0.3	o
Cs-137	Ba-137 (n,p)	30.0	8.7×10^5	500	o
Ba-133	Ba-132 (n,γ)	7.5	5.2×10^3	500	o

Table 3. Continued.

Nuclide	Reaction	$t_{1/2}$	c_c	c_r	Comment
La-137	Ce-136 (n,γ)	6×10^5	4.8×10^5	371	o
Pm-145	Sm-144 (n,γ)	18	2.2×10^3	0.1	o
Sm-151	Sm-150 (n,γ)	87	1.7	0.1	o
Eu-150m	Eu-151 (n,2n)	5	2.6×10^4	1	o
Eu-152	Eu-151 (n,γ)	12.5	6.7×10^{-3}	1	+
Eu-154	Eu-153 (n,γ)	16	0.59	1	
Eu-155	Sm-154 (n,γ)	1.7	4.3×10^5	0.1	o
Tb-157	Dy-156 (n,γ)	150	420	1	o
Tb-158	Tb-159 (n,2n)	150	1.6×10^3	0.47	o
Ho-163	Er-162 (n,γ)	1000	550	-	
Ho-166m	Ho-165 (n,γ)	1200	13	1	o
Tm-171	Er-170 (n,γ)	1.9	1.9×10^5	-	*
Lu-174	Lu-175 (n,2n)	3.6	4.9×10^4	0.8	o
Ir-192m2	Ir-191 (n,γ)	650	6.0	-	
Tl-204	Tl-203 (n,γ)	3.9	200	-	

potentially relevant, and if it is marked with a plus sign ("+"), it is an essential part of the radioactive inventory in the material; this sign indicates that the activity contribution of the nuclide is 10 % or more of the principal activity of the case.

4.2 Occupational exposure

The most significant nuclide by far in the stainless steel parts of the LWR is Co-60. The significance of Eu-152 and Eu-154 is reduced by their large burn-up cross sections and consequent conversion to Eu-155 with a short half-life.

In the carbon steel of the pressure vessel wall, Cs-134, Eu-152 and Eu-154 can be important. Co-60 is the predominant radionuclide in the biological shield with Ba-133 having a certain relevance here. The only radionuclide of importance in the reinforcement steel, apart from Co-60, is Cs-134. Thus from the viewpoint of occupational exposure after 5 years of dormancy, Co-60 determines the dose rate on all activated LWR materials, with small contributions from the Eu-nuclides in steel, Cs-134 in carbon steel and concrete and Ba-133 in concrete.

4.3 Long Term Exposure

All long-lived radionuclides are significant in the evaluation of population exposure caused by nuclide migration to the biosphere from a repository for decommissioning waste. Of the nuclides Fe-55, Ni-59, Co-60, Ni-63, Mo-93 and Eu-152, which have about the same order of magnitude of activity in steels, the importance of Fe-55 and Co-60 are reduced by their relative short half-lives while Ni-53 gives the largest activity contribution, it is Ni-59 that has been used as the principal nuclide because of its longer half-life.

5. Induced Activity in Structural Material

Several computer codes are available for calculating the induced activity in reactor materials. An important input in these calculations are the neutron flux densities and energy spectrum. As far as the core and the region nearest to it is concerned, most of such programmes have been validated by reactor physical calculations. The induced activity in regions further away from the core is more difficult to calculate exactly because of the large attenuation of the flux from the core.

A proposal was made by ABB Atom for making measurements in the biological shielding of the Oskarshamn 1 reactor in order to verify the calculated neutron flux densities. Foils of various materials would be chosen for exposure to the neutron flux for a pre-determined time. Measurement of the activity induced in the foils would make it possible to determine the neutron flux densities in the various neutron energy intervals. A comparison of the results of these measurements with the results of various computer calculations would show how successfully the programmes could calculate neutron flux densities in the regions in question.

It was also proposed to expose cement samples (i.e. concrete without ballast), with known composition, together with the neutron foils in order to obtain an experimental determination of the neutron induced activity in concrete. By using such an exposure geometry a "real-life" environment would be achieved and the neutron spectrum would be minimally disturbed by the experimental arrangements. The same computer codes as those used for the determination of experimental neutron flux densities could be used for calculating the induced activity in the exposed concrete samples. Suitable cross sections would have to be used as well as a special computer programme for taking into account the irradiation history of each sample. A comparison between measured and calculated values of induced activity would permit verification of the calculation codes. The measured induced activity values could also be compared with earlier calculated values for activity in the biological shield for verification.

5.1 Experimental work

The proposal was accepted and accordingly two chains of foils and cement specimens were exposed at Oskarshamn 1 during the operative year 1987-88 for a period corresponding to 7 359 effective full power hours (ref 5).

The test chains were lowered into two positions:

1/ into the annulus between the reactor vessel and the biological shield

2/ into an Intermediate Range Monitor (IRM) channel in the biological shield itself.

Each chain consisted of a number of aluminium capsules with foils of Sc, Ta, Co, Ag, Mn, Ni, Ti, Fe and Cu as well as cement samples of two types: ("Kolari LH" and "Limhamns LH").

In the IRM channel, the aluminium capsules and cement samples could be placed on a test chain, built of concrete cylinders with fixed measuring positions. A schematic sketch of the test layout is shown in Fig. 2.

The test chains were removed from the reactor in June 1988. Activity measurements were carried out with the so-called MADAC equipment (Mobile Analyzer for Detection of Activity in Crud) during the period October 1988 to April 1989. The activities were tabulated with corrections introduced for time between removal from the reactor and measurement, etc. and presented in tables and graphs with specific activity plotted against core height for 163 foils and 20 cement specimens. Examples of the results are shown in Fig. 3. The full details are given in ref 5.

5.2 Theoretical work

A study is in progress to compare calculated and measured values. Briefly, the work includes:

- calculation of neutron flux densities by "two dimensional ANISN"-code calculations based on a cylindrical symmetry. They comprise one axial and 6 radial calculations and will result in neutron flux densities as a function of radius and axial position along the heights of the reactor-core,

- calculation of induced activity in the foils and cement specimens exposed in the Oskarshamn 1 reactor during 1987-1988 based on the calculated neutron flux densities. The calculated and measured activity concentrations are then compared in order to obtain a quantitative estimate of how well the neutron flux densities have been calculated.

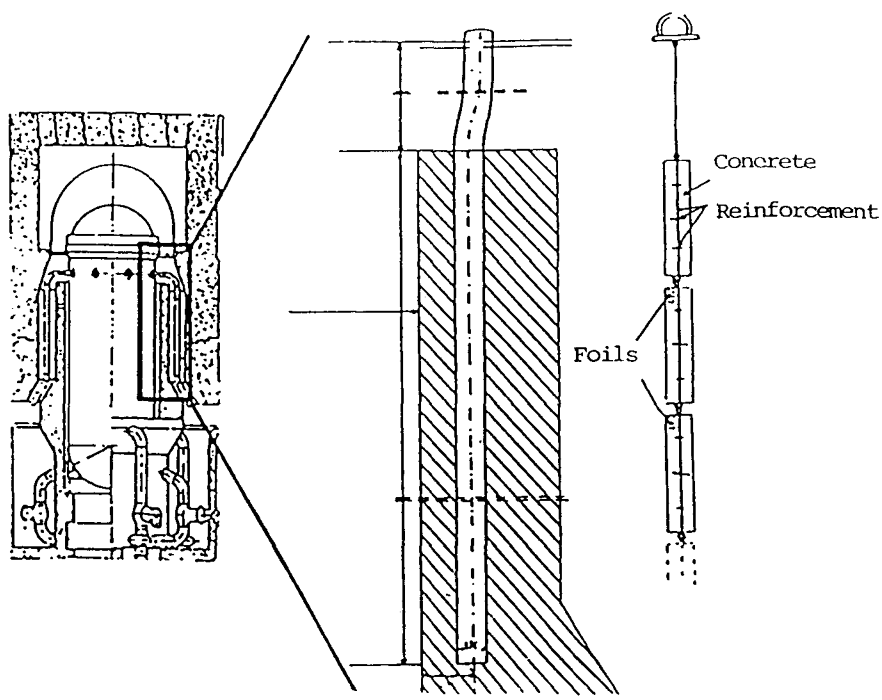
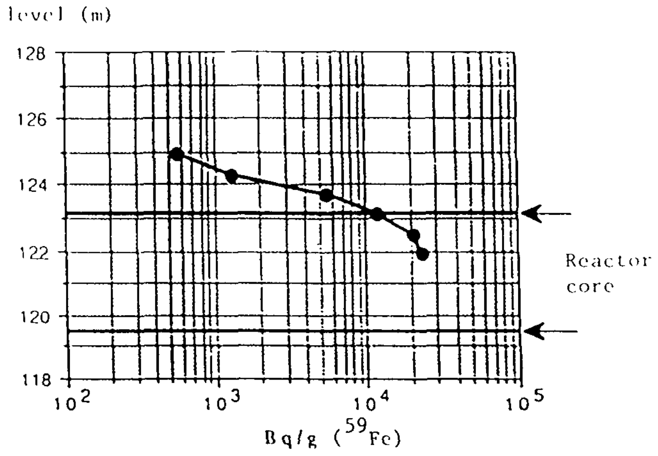
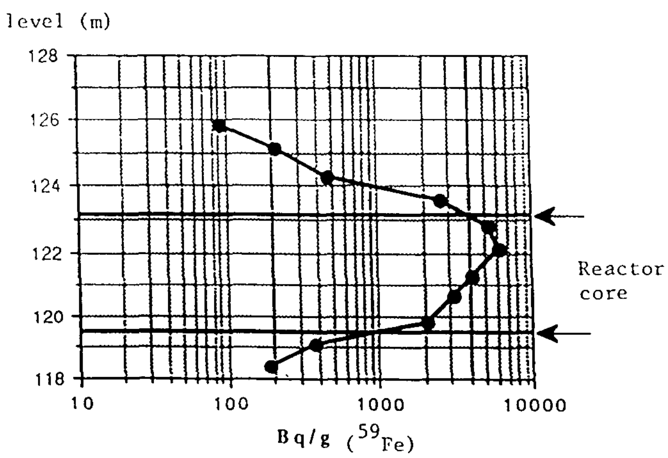


Fig 2. Sketch showing position of test foils and cement specimens.



IRM channel



Annulus between reactor vessel
and biological shield

Fig 3. Example of results from neutron foil measurements in Oskarshamn 1.

The study started at the end of 1989 with preparatory investigations concerning the geometry parameters and other input data to the ANISN-code. It will be reported during 1990.

6. Radioactivity Inventory in Contaminated Systems

The activity induced by neutron flux discussed in the previous chapter is, by the nature of its location, "locked" in place in the matrix of the structural material. Another on-going process during the operation of the reactor is the deposition of radionuclides on surfaces that are exposed to contaminated process media such as reactor cooling water and steam. Information regarding the nature and quantity of these "loose" nuclides is important for the decommissioning of the nuclear station. The highest activities are found in the core components and the reactor pressure vessel due to induced activity in the material.

Since 1977 measurements have been performed on all BWR's constructed by ABB Atom an equipment called "MADAC" (mentioned in 5.1 above). The MADAC-measurements facilitate nuclide-specific activity determination on system surfaces by means of gamma-spectroscopy. A collimated Ge(Li) detector which have been calibrated against surface-sources is used. A special computer code, CYLGAM, is developed by ABB Atom to calculate the surface activity concentrations based on the measured gamma intensity, dimensions of the object, shielding material and other relevant factors.

In a first study in this area (ref 3) the inventory of contamination in main systems in BWRs is estimated based on the MADAC measurements. In a second study (ref 6) the MADAC measurements were extended to comprise also such secondary systems that were not included in the first study because of their low contamination level. In the first study there is also a discussion on how the MADAC measurements used for the BWRs could be utilized for the PWRs.

The studies do not include:

- handling of spent fuel (not included in the decommissioning activities).
- neutron induced activity in the reactor, internals and surrounding structures
- activity accumulated during operation in the filters, ion exchange resin, etc or in the conditioned waste at the station.

6.1 Methods for Estimation of the Inventory.

Rough estimates of the current main radioactive inventories in systems wetted by contaminated process media have been made concerning two BWRs, Oskarshamn 2 in Sweden and TVO 1 in Finland. These estimates are based on:

- Dose rate measurements at a large number of points. These have been made annually since the start of reactor operation, during the shutdown for refuelling, etc.

- Measurements with the MADAC equipment from 1977 onwards. MADAC measures the activity on system surfaces nuclide specifically. It uses a collimated Ge(Li) detector which is calibrated for surface sources. Details of the MADAC measuring system can be found in ref 6.

The measurements were made two to three weeks after the shutdown of the reactor to allow for decay of the most short lived radionuclides.

6.2 Reactor Pressure Vessel

The inventory of surface contamination activity of the reactor pressure vessel has been estimated on the basis of measurements on the shutdown cooling system for the coolant wetted surfaces and on dose rate measurements at the inside of the pressure vessel lid.

Estimated inventory of surface contamination (Bq)

	Oskarshamn 2	TVO 1
Co-60	2.8×10^{11}	2.4×10^{11}
Co-58	3.2×10^{11}	1.7×10^{11}
Mn-54	1.4×10^{11}	8.7×10^{11}

As a matter of interest it can be mentioned that the calculated induced activity in the material of the pressure vessel after 40 years of operation is 3.8×10^{13} Bq (long lived radionuclides) which is significantly more than the activity due to contamination of the tank. Most of this activity is concentrated in the region nearest the core.

6.3 Steam Separator

The internal component with the largest surface area is the steam separator which has an area of 1 500 - 2 000 m^2 and is responsible for about half of the total surface contamination in the power station. The estimate is based on dose rate measurements as well as on a simplified calculation model.

Estimated inventory of surface contamination (Bq)		
	Oskarshamn 2	TVO 1
Co-60	9.9×10^{11}	9.3×10^{11}
Co-58	1.7×10^{12}	6.5×10^{11}
Mn-54	1.9×10^{12}	7.0×10^{11}

6.4 Other systems

The other systems for which quantitative estimates have been made are:

- Control rod drivers/hydraulic scram system
- Steam lines
- Feedwater lines
- Recirculation system
- Relief system
- Shutdown cooling system
- Containment vessel spray system
- Low pressure coolant injection system
- Reactor/Fuel pools and clean-up systems
- Spray for reactor flange
- Auxiliary feedwater system
- Reactor water clean-up system
- Drains from reactor systems

There are a few other systems, like those for waste treatment and systems in the turbine plant, for which no quantitative estimates have been made.

The results of the estimates are summarized in Table 4.

Table 4. Radionuclide Inventory of Systems in Oskarshamn 2 and TVO 1.
(Only surface contamination. Induced activity is not included).

SYSTEMS	OSKARSHAMN 2 Total (Bq)	OSKARSHAMN 2 Co-60 (Bq)	TVO 1 Total (Bq)	TVO 1 Co-60 (Bq)
Reactor Vessel	$7 \cdot 10^{11}$	$3 \cdot 10^{11}$	$5 \cdot 10^{11}$	$2 \cdot 10^{11}$
Steam Seperator	$5 \cdot 10^{12}$	$1 \cdot 10^{12}$	$2 \cdot 10^{12}$	$1 \cdot 10^{12}$
Control Rod Drives	$2 \cdot 10^{10}$	$1 \cdot 10^{10}$	$4 \cdot 10^{9}$	$2 \cdot 10^{9}$
Hydraulic Scram	$5 \cdot 10^{9}$	$3 \cdot 10^{9}$	$5 \cdot 10^{8}$	$2 \cdot 10^{8}$
Steam Lines	$3 \cdot 10^{11}$	$5 \cdot 10^{10}$	$9 \cdot 10^{10}$	$4 \cdot 10^{10}$
Feedwater	$\approx 10^{10}$	$\approx 5 \cdot 10^{9}$	$1 \cdot 10^{10}$	$5 \cdot 10^{9}$
Recirculation	$4 \cdot 10^{11}$	$2 \cdot 10^{11}$	-	-
Relief	$\approx 10^{9}$	$\approx 10^{9}$	$\approx 10^{9}$	$\approx 10^{9}$
Shutdown Cooling	$5 \cdot 10^{11}$	$3 \cdot 10^{11}$	$2 \cdot 10^{11}$	$1 \cdot 10^{11}$
Containment Vessel Spray	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$
Low Pressure Coolant injection	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$
Reactor/Fuel Pools	$9 \cdot 10^{11}$	$7 \cdot 10^{11}$	$1 \cdot 10^{12}$	$8 \cdot 10^{11}$
Fuel Pool Clean-up	$8 \cdot 10^{10}$	$6 \cdot 10^{10}$	$1 \cdot 10^{11}$	$5 \cdot 10^{10}$
Spray for Reactor Flange	$\approx 10^{10}$	$\approx 6 \cdot 10^{9}$	$1 \cdot 10^{10}$	$6 \cdot 10^{9}$
Auxiliary Feedwater	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$	$\approx 10^{8}$
Reactor Water Clean-up	$6 \cdot 10^{10}$	$4 \cdot 10^{10}$	$2 \cdot 10^{11}$	$2 \cdot 10^{10}$
Drains from reactor systems	$2 \cdot 10^{8}$	$3 \cdot 10^{7}$	$2 \cdot 10^{9}$	$2 \cdot 10^{9}$
Total	$8 \cdot 10^{12}$	$3 \cdot 10^{12}$	$5 \cdot 10^{12}$	$2 \cdot 10^{12}$

6.5 Complementary Measurements

Estimates of radionuclide inventory have been based on readings taken on systems and components with high dose rates (ref 3), as these were of most interest for the operating staff. Systems with lower levels of contamination were not covered. Moreover, even in the systems that were included, the number of measurements was limited and thus not always enough for making wide generalizations.

In order to provide a better base for the activity estimates of such lower contaminated systems, a second study was made, based on a series of measurements in the Oskarshamn 2 reactor during its annual shutdown in 1989 (ref 6).

The study comprises both MADAC-measurements and surface sampling to estimate surface contamination. In the surface sampling method, developed by ABB-KWN (Baden), the oxidized surface is ground with a greased grinding rod on which the oxide is collected. By measuring the activity on the grinding rod and the surface area, the contamination level could be established. The method gives a low detection limit but can only be used on open surfaces. Tables 5 and 6 summarizes the measurements. The location of the measuring and sampling positions are shown in Fig. 4.

6.6 Calculation Programme

A calculation programme BKM-CRUD has been developed by ABB Atom for computing and predicting the surface deposition of radionuclides on primary system surfaces. The model calculates the release of inactive and active crud products from fuel element and primary system surfaces and their subsequent deposition. The model takes into account the annual refuelling, when a significant part of the active crud products are removed from the reactor.

It is the intention of the SSI to continue the project and combine the measurements and the calculation program BKM-CRUD to facilitate accurate predictions of surface contamination levels on all major surfaces of contaminated reactor systems prior to decommissioning.

Table 5. Results from MADAC-measurements. The measuring points are shown in Fig. 4.

System nr/ measuring point	Dominating radionuclides	Surface contamination (Bq/m^2)	Dose rate ($\mu Sv/h$)
322/2	^{144}Ce, ^{141}Ce, ^{51}Cr	4.1 E7	6.5
323/1	^{144}Ce, ^{51}Cr, ^{141}Ce	1.1 E7	4.5
327/3	^{60}Co, ^{140}La, ^{58}Co	3.8 E5	0.2
332/5	^{51}Cr, ^{60}Co, ^{141}Ce	3.1 E6	2*
332/6	^{51}Cr, ^{60}Co, ^{131}I, ^{141}Ce	6.7 E5	2*
312/4	^{60}Co, ^{58}Co, ^{54}Mn	1.3 E5	0.6
455/7	^{51}Cr, ^{60}Co, ^{140}La, ^{140}Ba	1.7 E5	25*
341/8	^{140}La, ^{140}Ba		27
742/9	^{60}Co, ^{140}La, ^{152}Eu		0.5

* High background radiation level

Table 6. Results from measurements on surface samples. The measuring points are shown in Fig 4.

System nr/ measuring point	Dominating radionuclides	Surface contamination (Bq/m^2)	Dose rate (µSv/h)
211/17	^{60}Co, ^{58}Co, ^{54}Mn	1.7 E9	5000
326/18	^{60}Co, ^{58}Co	3.3 E9	2800
413/13	^{60}Co, ^{65}Zn, ^{58}Co	6.0 E7	410-640
412/16	^{51}Cr, ^{60}Co	2.3 E7	7
413/12	^{60}Co, ^{65}Zn, ^{58}Co, ^{131}I	1.7 E5	5.5
431/15	^{140}Ba, ^{60}Co, ^{131}I	3.7 E5	75
441/14	^{51}Cr, ^{140}Ba, ^{141}Ce	6.4 E7	15-20
D8.63/19	^{60}Co, ^{58}Co, ^{51}Cr	1.1 E4	0.5
D8.65/20	^{60}Co, ^{51}Cr, ^{58}Co, ^{54}Mn	2.0 E4	0.5

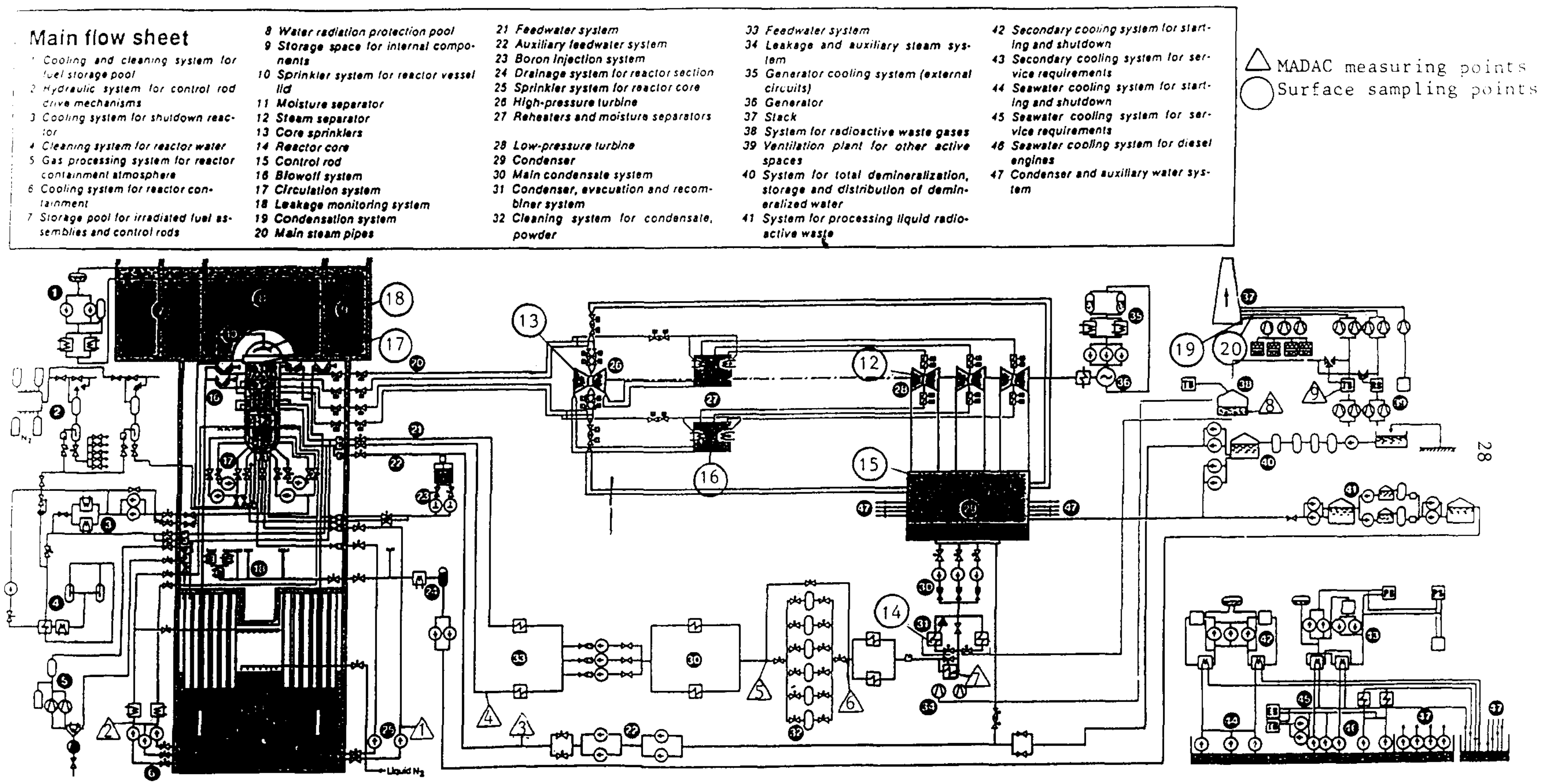


Fig 4. Main flow sheet indicating MADAC measuring positions and surface sampling positions.

6.7 Application to PWRs

In contrast to the direct cycle BWR, a PWR has a primary reactor coolant system and a secondary steam production system. Thus it has fewer systems in contact with the contaminated reactor coolant. The systems of main interest for radioactive contamination are:

- Reactor coolant system
- Steam generators
- Pressurizer
- Chemical and volume control
- Shutdown cooling
- Boron injection
- Waste systems

Some measurements have also been made on PWRs, e g at Ringhals by Studsvik AB for a number of years. These will have to be completed by more measurements to be useful for a comprehensive base for radionuclide inventory estimations. There is an increased need for shielding in PWR's with their compactly arranged primary systems. This places special requirements on the quality of the measuring equipment.

The BKM-CRUD programme cannot be utilized for the theoretical estimation of activity. A new programme would have to be written for this purpose.

7. Transport of Decommissioning Waste

The total volume of radioactive waste from the dismantling of a nuclear power station is of the same order of magnitude, if not larger, than that produced during the operating life time of the reactor. The decommissioning waste is moreover generated during a relatively short period of time. It also contains material with another radionuclide composition.

It is important that such waste is transported rationally and economically and, evidently, with an acceptable degree of radiation protection and safety. A new edition of the internationally accepted IAEA transport regulations has recently been issued (ref 23). Although transport of large volumes of wastes is to some extent included in this edition, the combination of waste packages and carriers, i.e. the transport system as a whole, is not yet included in the considerations.

Preliminary studies indicate that the transport recommendations can impose limitations which can complicate a rational waste management and also result in higher occupational doses for the personnel engaged in decommissioning than should be necessary.

The NKA study on this subject (ref 4) was therefore directed towards:

- making an overview of the nature and volume of decommissioning waste to be transported and expressing this in numbers and types of transport containers necessary. The Swedish nuclear programme has been used as the basis for this overview.

- preparing a basis of judgement for the various safety authorities in the Nordic countries regarding the practical applicability of the new IAEA transport recommendations for the transport of decommissioning waste.

In another NKA-programme, KAV-365, an overview of Nordic transports involving radioactive material is given (ref 24).

7.1 Quantity of Decommissioning Waste

In Table 7 are summarized the results of the overview regarding the quantity of decommissioning waste five years after shut down. The number of transport containers of various types necessary for transportation of the wastes is shown in Table 8. The choice of container types has been influenced by the various types of waste and their predicted activity contents.

Table 7. Overview of waste arising from decommissioning of the Swedish nuclear power plants

Decommissioning Waste from the various reactors						
Active Waste	(In tons)					Inactive waste
Reactor	Reactor Vessel	Other Active Systems	Sand	Concrete	Total	
O1	650	1990	250	615	3505	6135
O2	650	2475	250	900	4275	7480
O3	760	4725	1050	1410	7945	13905
R1 (ref)	650	3325	350	915	5240	9170
R2 (ref)	330	2460	-	975	3765	8135
R3	330	2460	-	975	3765	8135
R4	330	2460	-	975	3765	8135
B1	650	2470	250	900	4275	7480
B2	650	2655	250	990	4550	7965
F1	760	4120	1050	1230	7165	12540
F2	760	3935	1050	1230	6980	12215
F3	760	4720	1050	1440	7975	13960
			Total	(In tons)	63205	115225

Table 8. Transport containers needed to transport the decommissioning waste arising from the Swedish nuclear power reactors

Transport Units for Active Waste

Reactor	ISO 30 m^3	½ISO 15 m^3	ST	ATB 20 m^3 Pipes and Equipment	ATB 20 m^3 Other waste	HKB 20 m^3 Reactor material
01	60	145	29	3	28	40
02	75	198	29	4	28	40
03	143	367	29	6	33	40
R1	100	226	29	4	28	40
R2	50	118	11		57	65
R3	50	118	11		57	65
R4	50	118	11		57	65
B1	75	198	29	4	28	40
B2	80	205	29	4	28	40
F1	124	330	29	5	33	40
F2	119	330	29	5	33	40
F3	143	370	29	6	33	40
Total	1069	2773	294	41	443	555
				484		
Net volume required in SFR3 in m^3:						
	32070	41595	7025	9680		

which gives a total of 90370 m^3 of waste.

In the above table,

ISO	Denotes	Standard ISO-container
ST	"	Special transports of complete pieces of equipment
ATB	"	Shielded waste transport container
HKB	"	Core component container
SFR3	"	Final waste repository for decommissioning waste

The core component container is a modified fuel transport cask. The final conditioning of reactor internals has been assumed to take place in connection with the conditioning of spent fuel.

The total activity per reactor, 5 years after the final shutdown, is estimated to be 9×10^{11} Bq, not including the activity in the reactor vessel and internals. The activity is concentrated to a few systems. About 75 % of all transports can be carried out in ordinary ISO containers. It is estimated that 6 % of the transports can take place as "special arrangements", i.e. without packaging. Some of the components transported in this manner, such as PWR steam generators, will require special care and attention, beacuse of the activity levels in the channel head area. Most of the components for which special arrangements are planned are pieces of equipment with low activity levels, but they would require considerable efforts in segmenting.

7.2 Applicability of IAEA Transport Recommendations

The new IAEA transport recommendations (ref 23) have been studied in order to illustrate possible differences with and problem areas in relation to the existing Swedish transport system. The ISO containers and the core component packages that are proposed for the Swedish systems will be in complete accordance with the requirements for Industrial Package 1 (IP1) and Type B package respectively. The shielded waste transport packaging ATB fulfills the formal requirements in § 134 in the IAEA transport regulation for an IP2 packaging. Some of the verifications, e g drop test from 0.3 m, was done by calculations.

The Swedish system satisfies the requirements such as dose rates, transport index, etc with comfortable margins. Moreover, a detailed radiation protection report will be prepared and will have to be accepted by the authorities before the start of the transport of decommissioning waste. The transport of large pieces of equipment which is included in the above tables, has been proposed in order to limit occupational doses. Such transports may be permitted as "Special arrangements" according to § 141 in the IAEA regulation. In the Swedish case a large number of similar pieces of equipment are expected to be transported in this manner. The NKA study has shown that further guidelines would be useful in this area.

8 Case study

In order to identify areas where further knowledge is needed before dismantling can be performed in a controlled manner and considering established radiation protection practice, a model study of decommissioning of the JEEP II reactor at IFE in Kjeller, Norway, was performed (ref 7). The areas of main interest during the study were:

- Radionuclide inventory
- Waste characterisation
- Waste management
- Transport
- Disposal

while others such as dismantling and decommissioning techniques were left aside. These techniques are expected to be easy to adapt to the specific situation at a later stage. Similar work is also in progress in other countries.

8.1 Radionuclide inventory

The JEEP II reactor is in full operation and has recently been given prolonged operating licence for another 10 years. This limits the possibilities to take samples and make on-site measurements. The only samples which could be used for destructive testing were from a replacement plug, which had been used for several years in the biological shield.

Due to lack of nationally tested computer codes the estimation of the inventory was carried out using manual calculations based on estimated neutron flux, experience from other reactors, and measurements on a few samples.

The dominating radionuclide is Co-60 but also Eu-152 was of great significance. This is in agreement with the study (ref 2) on radionuclides of importance in decommissioning discussed in chapter 4.

8.2 Waste characterisation

From drawings, other documents and previous experience it was clear that the materials of main interest as future decommissioning waste from the JEEP II reactor were:

- Aluminium in the reactor tank and internal components
- Carbon steel in the thermal shield
- Heavy concrete containing iron ore in the biological shield

The inventory of radionuclides is dominated by induced activity in the construction material. Thus the radionuclides are normally, already when they are generated, incorporated and immobilized in a stable non-combustible form which is an important safety asset during waste treatment, temporary storage, transport and disposal.

In an environment containing water, metal corrosion and concrete degradation will gradually cause an increase in the mobility of a small fraction of the nuclides which were originally present in the waste.

In a water filled repository environment the degradation of concrete is likely to be caused by sulphate attack and calcium hydroxide leaching. For a one meter thick slab of concrete, based on ordinary Portland cement, it is estimated that the complete leaching of calcium from the concrete will require a time period of the order of 10 000 years. The corrosion rate of ordinary unalloyed steel is estimated to be in the range from 0.0001 to 0.0003 mm per year, depending on pH, E_h, and other chemical conditions of the ground water.

Metal corrosion produces hydrogen gas, and this gas production is one of the factors which have to be considered when designing a repository for decommissoning waste.

8.3 Waste management

A main objective in waste management is to minimize the volume of radioactive waste by use of efficient decontamination methods and waste segregation. This can eliminate radioactive contamination of waste which otherwise can be treated as non-radioactive material. Efficient decontamination methods can be a means of reducing the radiation dose to personnel during decommissioning. It can also increase the amount of waste which can be treated as non-radioactive.

It is anticipated that the waste treatment plant at Kjeller will be used during decommissioning for concentrating and purifying liquid wastes resulting from decontamination, for solidification of waste concentrates and ion exchange resin, and for waste volume reduction by cutting, pressing and combustion.

8.4 Transport

The waste transport to the repository will be carried out in containers in accordance with national and international regulations for transport of radioactive materials in a similar way as has been discussed in chapter 7. The largest radioactive component, the reactor tank, can be cut into sections and transported in container, or it can be transported as a whole unit, under special authorized transport permission.

In Norway there is more than 30 years of experience with transport of radioactive materials, both national and international transports.

Road transports over a distance of 120 km from Halden to Kjeller of reactor wastes arising from operation and maintenance of the Halden Boiling Water Reactor, have been carried out since 1963. The total activity contained in this waste is up to now approximately 10 TBq, about three times more than the activity estimate for the decommissioning waste from JEEP II. All waste transports up to date have been carried out without any incidents involving members of the general public and without contamination of the environment. The radiation dose to operating personnel has been kept well below the dose limits given by the Norwegian National Institute of Radiation Hygiene.

This study (ref 7) of decommissioning waste transport has not revealed any radiation or safety problems which can not easily be dealt with within the existing transport regulations.

8.5 Waste disposal

Radioactive waste generated in Norway is now processed and stored in waste storage buildings at IFE, Kjeller.

It is anticipated that this waste and future radioactive wastes from decommissioning will be permanently disposed of in a suitable repository. This future repository for the Norwegian low and intermediate level radioactive waste is subject to a study by a governmental committee, planning to present a proposal for its construction principles and siting during 1990. Safety analyses of underground repositories for radioactive waste from reactor operation and decommissioning have been carried out in Finland and Sweden. The results of the analyses show that these repositories can provide very good radiological safety.

Large quantities of wastes from dismantling can be treated as non-radioactive, or can be used for backfilling and landscaping when returning the site to green-field conditions.

8.6 Further work

An important result of the study (ref 7) was the identification of areas which have to be further evaluated in Norway. Some examples of such areas are: estimations of radionuclide inventory including use of computer codes, suitable waste characterisation systems and disposal alternatives. The need for an increased competence in the field of decommissioning is recognised.

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