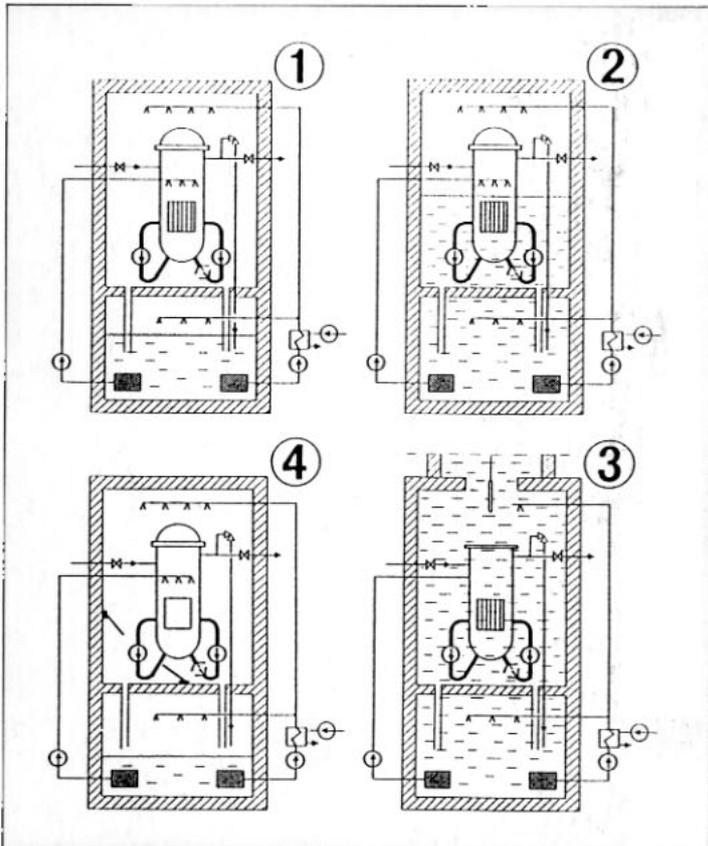


# MANAGEMENT OF RADIOACTIVE WASTE FROM A MAJOR CORE DAMAGE IN A BWR POWER PLANT



nka

Nordic  
liaison committee for  
atomic energy

**MANAGEMENT OF  
RADIOACTIVE WASTE  
FROM A MAJOR CORE DAMAGE  
IN A BWR POWER PLANT**

**Final Report of the NKA Project KAV 390**

**Edited by:**

**Jan Elkert, Hans Christensen and Börje Torstenfelt  
ABB Atom, Sweden**

**MARCH 1990**

This Report is available from:

Swedish Nuclear Power Inspectorate  
Box 271 06  
S-102 52 STOCKHOLM, Sweden

and

ABB Atom AB  
S-721 63 VÄSTERÅS, Sweden

The front cover shows four stages of the restoration work. No. 1 is the situation just after the event. No. 2, containment filled with water, and No. 4 containment decontamination, are covered by the report, while No. 3, fuel removal, is not addressed in the report.

ISBN 87 7303 420 7

Graphic Systems AB, Malmö 1990

NORD 1990:31

Technical Report SKI 90:2

ABB Atom Report RM 89-91

**ABSTRACT**

Large amounts of fission products would be released in case of a major core damage in a nuclear power reactor. In this theoretical study the core damage is caused by a loss of coolant accident followed by a complete loss of all electric power for about 30 minutes resulting in the release of 10 % of the core inventory of noble gases. A second case has also been briefly studied, in which the corresponding core damage is supposed to be created merely by the complete loss of electric power during a limited time period.

It appears from the study that the radioactive waste generated as a consequence of an accident of this extent can be managed in the reference reactor with only minor modifications required in the waste plant. The detailed results of the study are reactor specific, but many of the findings and recommendations are generally applicable.

**INIS Descriptors:**

BWR TYPE REACTORS, DECONTAMINATION, LIQUID WASTES, LOSS OF COOLANT, NUCLEAR CORES, RADIOACTIVE WASTE PROCESSING, RADIOACTIVE WASTES, SOLID WASTES, FINLAND, SWEDEN.

This report is part of the safety programme sponsored by NKA, The Nordic Liaison Committee for Atomic Energy, 1985-1989. The project work has partly been financed by the Nordic Council of Ministers.



## SUMMARY

Fission products released from a damaged reactor core are unavoidably dispersed in the reactor building in various ways depending on the sequence of events leading to the damage and the extent of the damage. For example, the dispersion can be due to the loss of integrity of the reactor pressure vessel or the circulation of cooling water outside the reactor containment. The presence of radioactive substances in large amounts in places where this is not designed for would be of great concern as the resulting ionizing radiation may affect the possibility to maintain the reactor control in the long run, for example if the access to vital parts of the reactor building is restricted or impossible. Consequently, it is important to take care of the fission products released and safely dispose of them as soon as possible.

The objective of this project has been to find out to what extent the waste plants presently existing at the reactor sites can be used to manage a larger, hypothetical reactor accident in a BWR plant and if any additions to these plants are required for this purpose. The study has been administered by the Swedish Nuclear Power Inspectorate (SKI) and carried out by ABB Atom. It has been sponsored by the Inspectorate, the Swedish National Institute of Radiation Protection (SSI), the Swedish nuclear utilities OKG AB and Sydkraft AB, the Finnish nuclear utility TVO, and the Nordic Council of Ministers.

In the main case studied, the starting event is a pipe break causing a loss of coolant accident (LOCA), followed by the complete loss of all electric power. Some time has also been devoted to a second case, where the scenario is merely a complete loss of all electric power, starting with the outer net. In both cases the sequence leading to a core melt-down is assumed to stop after a while, when the electric power has been restored and eventually the core cooling reestablished. The accident scenarios involve a fuel damage resulting in the release of 10% of the core inventory of noble gases. It should be noted, that the scenarios chosen are very unlikely to happen and serve only as a background in order to provide the above mentioned release of fission products. Accident management is only briefly treated and where appropriate.

In the main case the reactor containment is filled with water to secure the long term core cooling. This means that a volume of about 10 000 m<sup>3</sup> of highly radioactive water will have to be treated in the waste plant, and also that accessible surfaces in the containment will be radioactively contaminated, which at the end gives a difficult waste management problem. In the second case the radioactive matter released from the core is mainly confined to the reactor vessel, leading to a much smaller volume of about 250 m<sup>3</sup> of water with a very high radioactive content.

In this theoretical study the Oskarshamn 2 reactor - run by the OKG utility - has been used as reference reactor. The prerequisite has been to restore the reference reactor to such an extent that it can eventually operate, and to have the first reactor unit at the Simpevarp site, sharing the same waste plant, operating during the restoration work.

The first waste problem encountered after an accident is the management of highly radioactive leakage water from the reactor containment. The cleanup systems in the waste plant are not available at the beginning, because of the need for preparations and any necessary modifications before starting the major cleanup. Therefore, the leakage water should be directed to the containment, which under these circumstances is the most suitable storage space. In the reference reactor, this is possible after making certain pipe connections in the waste plant, something which can be completed within a week following the accident.

Other options recommended are to modify the waste plant so that it can receive and treat the assumed amount of leakage water or to make additions in the leakage drain and floor drain systems in the reactor building.

The major cleanup step is supposed to start at the earliest three months after the accident and will be performed as a feed and bleed process. The existing cleanup and solidification systems in the waste plant should be used as much as possible.

The fission products that need to be taken care of are mainly cesium-134 and cesium-137 and to some extent strontium-90. Cesium is the largest radiation source in the containment water and on contaminated surfaces. The total amount of cesium and strontium that needs to be taken up by ion exchangers and solidified in the waste plant is  $2.7 \cdot 10^4$  TBq and  $1.5 \cdot 10^3$  TBq, respectively (three months after the accident).

Due to the high radiation doses expected to the ion exchange resins in the cleanup system, the use of zeolites should seriously be considered for the major cleanup step. Specific zeolites have an excellent selectivity of especially cesium and a profound radiation resistance as compared to organic resins. At present the use of granulated zeolites has not been tested in the cleanup system, but no serious problems are expected. It is recommended, however, that preparatory measures are considered for the cleanup system to enable the connection of special filter vessels for inorganic sorbents such as zeolites in case the ordinary vessels are unsuitable or not available.

Instead of using zeolites in the existing ion exchange vessels, special once-for-all filters with zeolites may perhaps be used, which once loaded are directly solidified in cement.

Cement should be used as the solidification matrix because cement is radiation resistant, zeolites are well integrated into a cement matrix and cement is the solidification matrix normally used in the reference solidification plant, hence the existing equipment can be utilized. However, with the use of special once-for-all zeolite filters the cement casing would have to be modified.

In the early stages of the cleanup process it is the solidification capacity of the waste system that will be limiting, while the uptake of nuclides on the ion exchanger will be limiting at the very end of the process. Thus, during the first months of the cleanup, extra solidification capacity would speed up the restoration, for example if a mobile solidification unit was also used. Moreover, the use of prefabricated lids to the moulds is feasible.

As a conservative estimate the entire water cleanup procedure yields about 800 concrete moulds (cubical with external dimension 1.2 m and wall thickness 0.25 m), assuming that zeolites are used in the major cleanup step and that the surface dose rate is below 300 mSv/h on the moulds. The procedure is completed about two years after the occurrence of the accident. Handling of concrete moulds with such a high surface dose rate is possible in the reference reactor but requires the introduction of remotely controlled equipment and probably additional radiation shielding.

If, for some unforeseen reasons, it is not possible to use zeolites, the alternative is organic resins. In this case the presently allowed surface dose rate on the moulds is only 30 mSv/h, leading to a much lower radioactive load than in the zeolite case. Consequently, the required number of concrete moulds will increase to between 3 500 and 4 000, and the entire water cleanup procedure would be completed three to five years after the accident.

While the decontamination of the waste water seems to be a fairly straightforward process, the surface decontamination of the reactor vessel and containment, and equipment therein, would be much more difficult. Methods for decontaminating nuclear facilities are known, but few of these can be used on, for example, concrete surfaces or external parts of equipment without causing damage or at least requiring major reconstruction work of the reactor after the decontamination.

The restoration of the reactor will be a time consuming and expensive process. It is delayed not only by the surface decontamination and the removal of all remaining mineral wool insulation but also as a consequence of the water filling of the containment. This means that at least all electrical equipment including cables, and probably much of the mechanical equipment, must be either renovated or replaced in order to comply with the quality assurance requirements.

As a main conclusion of the study it appears that a core damage corresponding to a release of 10% of the noble gas inventory in the fuel can be managed in the reference reactor Oskarshamn 2. Only minor modifications are required in the waste plant after the event.

Many of the findings and recommendations of the study are generally applicable. One issue, for example, is the influence of radiation on the tightness of components and the possible leakage due to a break down of sealings. This question has not been fully studied in the project. Another issue is the importance of not operating the ion exchanger in the reactor water cleanup system after the event in order not to expose the ion exchange resin to such a high radiation dose that it will be difficult to handle. Individual characteristics at different nuclear power plants, however, make the detailed results of the study reactor specific.

As regards the second case, this has not been analysed in detail. Here, the contaminated water is confined to the reactor vessel and hence, has a much higher concentration of radionuclides. This can be dealt with either by evaporation or by dilution and subsequent cleaning as in the first case. Therefore, the same conclusion is valid. In the reference reactor it is possible to manage the radioactive waste after the event. Furthermore and in general, the possibility of taking the reactor into operation again seems more likely in the second case, as the contamination of the reactor containment will be much smaller.



**SAMMANFATTNING**

Det är oundvikligt att klyvningsprodukter från en skadad reaktorhård kommer att spridas i reaktorbyggnaden på olika sätt beroende på de händelser som föregår haveriet och haveriets omfattning. Så till exempel kan spridningen ske genom att reaktortanken förlorar sin integritet eller genom att kylvattnet cirkulerar utanför reaktorinneslutningen. Förekomsten av radioaktiva ämnen i stora mängder på ställen där detta inte har förutsatts vid konstruktionen skulle innebära stora bekymmer, eftersom strålningen kan påverka möjligheten att i det långa loppet kontrollera reaktorn, t ex om tillträdet till viktiga delar av reaktorbyggnaden blir begränsat eller omöjligt. Det är därför viktigt att så snart som möjligt ta hand om de frigjorda klyvningsprodukterna och förvara dem på ett säkert sätt.

Avsikten med detta projekt har varit att ta reda på i vilken omfattning de nuvarande avfallsanläggningarna vid kärnkraftverken kan användas för att ta hand om ett större, hypotetiskt reaktorhaveri i en BWR-anläggning och om några kompletteringar av anläggningarna behövs för detta ändamål. Utredningen har administrerats av Statens Kärnkraftinspektion (SKI) och utförts av ABB Atom. Arbetet har finansierats av SKI, Statens Strålskyddsinstitut (SSI), de svenska kärnkraftföretagen OKG AB och Sydkraft AB, det finska kärnkraftföretaget TVO samt av det nordiska ministerrådet.

I huvudfallet är den utlösande händelsen ett rörbrott som orsakar en förlust av kylvattnet och som följs av ett fullständigt bortfall av all elektrisk kraft. En del tid har också ägnats åt ett andra fall, där scenariot endast är ett fullständigt bortfall av elkraften, som börjar med det yttre nätet. I båda fallen antas det att händelsekedjan som leder till en härdsfälta avbryts efter en stund då den elektriska kraften har återfåtts och slutligen kylning åter erhållits. Haveriscenarierna innebär en bränsleskada som resulterar i att 10 % av hårdens innehåll av ädelgaser frigörs. Det bör påpekas att sannolikheten för de valda scenarierna skall inträffa är mycket liten och att de endast utgör en bakgrundsmiljö som skall resultera i den tidigare nämnda frigörelsen av klyvningsprodukter. Haverihanteringen berörs endast kortfattat och där det är på sin plats.

I huvudfallet fylls reaktorinneslutningen med vatten för att säkerställa den långsiktiga kylningen. Detta innebär att en volym av omkring 10 000 m<sup>3</sup> mycket radioaktivt vatten skall behandlas i avfallsanläggningen samt att de fria ytorna i inneslutningen blir radioaktivt kontaminerade, vilket är av stort intresse när det gäller avfallshanteringen. I det andra fallet är de radioaktiva ämnen som frigörs från hårdens huvudsakligen inneslängda i reaktortanken, vilket ger en mycket mindre volym av omkring 250 m<sup>3</sup> vatten men med en mycket högre aktivitetskoncentration.

I denna teoretiska utredning har Oskarshamn 2, som drivs av OKG AB, använts som referensreaktor. Förutsättningen har varit att återställa reaktorn i sådan omfattning att den slutligen kan tas i drift och att reaktorblock 1, som använder samma avfallsanläggning, skall vara i drift under detta arbete.

Det första avfallsproblemet efter ett haveri är omhändertagandet av läckagevatten med hög radioaktivitet från reaktorinneslutningen. Reningsystemen i avfallsanläggningen är inte tillgängliga i början, därför att förberedelser och eventuella nödvändiga modifieringar behöver göras i avfallsanläggningen innan det huvudsakliga reningsarbetet startar. Läckagevattnet bör därför dirigeras till inneslutningen som under dessa förhållanden är en synnerligen lämplig förvaringsplats. I referensreaktorn är det möjligt att göra så senast en vecka efter haveriet, sedan vissa rörförbindelser gjorts i avfallsanläggningen.

Andra rekommenderade alternativ är att modifiera avfallsanläggningen så att den kan ta hand om den förmodade läckagemängden eller att komplettera systemen för golvdränage och systemläckage i reaktorbyggnaden.

Det stora reningssteget antas starta tidigast tre månader efter haveriet och sker genom att det renade vattnet återförs till reaktorinneslutningen från avfallsanläggningen. De system som finns i denna för rening och för överföring till fast form (injutning) bör användas i så stor utsträckning som möjligt.

De klyvningsprodukter som behöver tas om hand är huvudsakligen cesium-134 och cesium-137 och i viss mån strontium-90. De båda cesiumisotoperna utgör den största strålningskällan i det vatten som finns i inneslutningen och på kontaminerade ytor. Den totala mängden cesium och strontium som behöver tas upp av jonbytare och överförs till fast form i avfallsanläggningen är  $2,7 \cdot 10^4$  TBq respektive  $1,5 \cdot 10^3$  TBq (räknat tre månader efter haveriet).

På grund av den förväntade höga stråldosen till jonbytarmassan i reningssystemet bör användningen av zeoliter i det stora reningssteget allvarligt övervägas. Speciella zeoliter har en utmärkt selektivitet, särskilt för cesium, och en hög motståndskraft mot strålning i jämförelse med organiska jonbytarmassor. För närvarande har inte användningen av kornformiga zeoliter provats i reningssystemet men några allvarliga problem kan inte förutses. En rekommendation är dock att förberedande åtgärder övervägs för reningssystemet för att möjliggöra anslutningen av särskilda filterkärn för oorganiska sorbenter såsom zeoliter i fall de existerande kärn är olämpliga eller inte disponibla.

I stället för att använda zeoliter i de nuvarande jonbytarkärn kan eventuellt speciella engångsfilter med zeoliter användas. När de en gång har fyllts kan de direkt gjutas in i cement.

Cement bör användas som matris vid överföringen av jonbytarmassan till fast form, därför att detta material är motståndskraftigt mot strålning och zeoliter förenas väl med cementmatrisen. Vidare är cement den solidifieringsmatris som normalt används i referensreaktorns ingjutningsanläggning, varför den nuvarande utrustningen kan användas. Men används speciella engångsfilter skulle denna anläggning behöva modifieras.

I de tidiga stadierna av reningsprocessen är det ingjutningsanläggningens kapacitet som kommer att bli begränsande, medan upptaget av radioaktiva ämnen på jonbytarna blir begränsande mot slutet av reningen. Under de första månaderna av reningen skulle därför en extra ingjutningskapacitet snabbt på återställandet, t ex genom att en mobil anläggning också används. Dessutom är det möjligt att använda förtillverkade lock till betongkokillerna.

En försiktig uppskattning visar att hela processen att rena vattnet ger omkring 800 betongkokiller (kubiska med yttermått 1,2 m och vägg tjocklek 0,25 m) om man förutsätter att zeoliter används i det stora reningssteget och att ytdosraten på kokillerna är högst 300 mSv/h. Arbetet är avslutat omkring två år efter haveriets inträffande. Det är möjligt att hantera betongkokiller med sådan hög ytdosrat i referensreaktorn, men det kräver att avståndsstyrd utrustning införs och kanske även ytterligare strålskärning.

Om det av oförutsedda skäl inte är möjligt att använda zeoliter är organiska jonbytarmassor alternativet. I detta fall är den högsta tillåtna ytdosraten på kokillerna för närvarande 30 mSv/h, vilket leder till ett mycket lägre aktivitetsinnehåll än i zeolitfallet. Följaktligen ökar behovet av betongkokiller till mellan 3 500 och 4 000 och reningen av vattnet skulle vara avslutad tre till fem år efter haveriet.

Medan reningen av avfallsvattnet tycks vara en ganska okomplicerad process, så skulle ytdekontamineringen av reaktortanken, reaktorinneslutningen och utrustning i denna bli mycket svårare. Metoder för att dekontaminera kärntekniska anläggningar är kända, men få av dessa kan användas på t ex betongytor eller utrustningens yttre delar utan att orsaka skada eller åtminstone kräva större ombyggnadsarbeten av reaktorn efter dekontamineringen.

Återställandet av reaktorn kommer att bli en tidskrävande och dyrbar process. Den försenas inte bara av ytdekontamineringen och borttagandet av all kvarvarande mineralullsisolering utan också som en följd av att inneslutningen fylls med vatten. Detta innebär att all elektrisk utrustning inklusive kablar och troligen mycket av den mekaniska utrustningen måste antingen renoveras eller ersättas för att uppfylla kraven på kvalitetssäkring.

Som en väsentlig slutsats av studien framgår det att en härdskada motsvarande en frigörelse av 10 % av ädelgasinventariet i bränslet kan hanteras i referensreaktorn Oskarshamn 2. Endast mindre förändringar är nödvändiga i avfallsanläggningen efter haveriet.

Många av slutsatserna och rekommendationerna i utredningen är allmänt tillämpbara. En fråga t ex är stråldosens inverkan på komponenters täthet och det eventuella läckaget på grund av att tätningen går sönder. Detta har inte undersökts fullständigt i projektet. En annan fråga är vikten av att inte ha jonbytarna i reningssystemet för reaktorvatten inkopplade efter haveriet, för att inte utsätta jonbytarmassan för så hög stråldos att den blir svår att hantera. Individuella särdrag hos olika kärnkraftanläggningar gör emellertid att studiens resultat i detalj är reaktor-specifik.

Vad beträffar det andra fallet så har detta inte analyserats i detalj. Här är det radioaktiva vattnet inneslutet i reaktortanken och har därför en mycket högre koncentration av radionuklider. Detta kan hanteras antingen genom förångning eller genom utspädning och efterföljande rening som i det första fallet. Därför gäller samma slutsatser. I referensreaktorn är det möjligt att hantera avfallet efter haveriet. Dessutom och i allmänhet syns möjligheten att åter ta reaktorn i drift vara mera rimlig i det andra fallet, då kontamineringen i reaktorinneslutningen kommer att bli mycket mindre.

## LIST OF CONTENTS

	<u>Page</u>
1. INTRODUCTION	1
2. MAIN EVENT AND ASSUMPTIONS	4
2.1 Reactor accidents. Analysis of two types of accident	4
2.2 Waste management aspects of the accidents.	
Selection of main event	5
2.3 Comparison with TMI-2	7
2.4 Reference reactor	8
3. THE SOURCE TERM ESTABLISHMENT	9
3.1 The AB <sub>m</sub> case	9
3.1.1 Release of fission products	
3.1.2 Release of actinides	
3.2 The TB <sub>m</sub> case	16
3.2.1 Release of fission products	
3.2.2 Release of actinides	
3.3 Chemical conditions and expected distribution of released radionuclides between gas phase, liquid phase and solid surfaces	16
3.3.1 Chemical speciation and sorption of fission products	
3.3.2 Chemical speciation and sorption of actinides	
3.3.3 Distribution of radionuclides in the power plant	
4. RADIATION CALCULATION	20
4.1 Introduction	20
4.2 Source terms	20
4.3 Radiation levels and accessibility in the reactor building	21
4.3.1 General calculations for pipes	
4.3.2 Access for measures during the period immediately following an AB <sub>m</sub> event	
4.3.3 Access during the main cleanup procedure	
4.4 Radiation doses to organic materials	23
4.5 Radiation levels and accessibility in the waste plant	24
4.6 Waste container	24
4.6.1 Radioactive load	
4.6.2 Handling and transport	
5. MEASURES TO BE TAKEN IN THE REACTOR BUILDING AND THE WASTE PLANT DURING THE PERIOD IMMEDIATELY FOLLOWING AN AB <sub>m</sub> EVENT	27
5.1 Sequence of events in the reactor containment	27
5.1.1 Spraying	
5.1.2 Filling with water	
5.2 Sampling	28
5.2.1 Water sampling	
5.2.2 Gas sampling	
5.3 Disconnection of the reactor water cleanup system	30

## LIST OF CONTENTS, Cont.

	<u>Page</u>
5.4 Management of airborne radioactivity in the reactor building	30
5.5 Management of leakage water in the reactor building	32
5.5.1 Spreading of radioactivity in the reactor building	
5.5.2 Discharge of leakage water from the reactor building	
5.6 Management of leakage water in the waste plant	34
5.6.1 Introduction	
5.6.2 Return pumping and storage	
5.6.3 Cleaning	
5.6.4 Conclusion	
6. CHOICE OF SORBENTS AND WASTE FORMS FOR THE TREATMENT AND SOLIDIFICATION OF RELEASED RADIONUCLIDES	38
6.1 Sorbents suitable for the cleanup	38
6.2 The solidification matrix	40
7. WATER TREATMENT AND IMMOBILIZATION OF THE GENERATED WASTE	42
7.1 Technical description of the water cleanup method	42
7.2 The various steps of the cleanup process	44
7.3 Water processing	46
7.3.1 Water processing by inorganic sorbents	
7.3.2 Possible microbiological growth	
7.3.3 Processing of the water by evaporation	
7.3.4 The use of organic ion exchange resins	
7.3.5 Special filters	
7.3.6 Doses to the environment from water released to the recipient	
7.4 Immobilization of the generated waste	49
7.4.1 Waste container	
7.4.2 Solidification of inorganic sorbents	
7.4.3 Solidification of organic resins	
7.4.4 Solidification capacity	
7.5 Time required for the cleanup and waste quantities produced	50
7.6 Handling and transport of waste containers	52
7.6.1 Handling in the waste building	
7.6.2 On-site transportation and storage	
8. HANDLING AND TREATMENT OF ADDITIONAL WASTES AFTER AN AB <sub>m</sub> EVENT	53
8.1 Contamination of the reactor containment	53
8.2 Cleanup of surfaces and equipment	54
8.2.1 Decontamination using water	
8.2.2 Chemical decontamination	
8.2.3 Mechanical decontamination	
8.3 Generated waste volumes	58

**LIST OF CONTENTS, Cont.**

	<u>Page</u>
9. THE TB <sub>m</sub> CASE	60
9.1 Measures to be taken in the reactor building and the waste plant immediately following the commencement of the accident sequence	60
9.1.1 Sequence of events in the reactor containment	
9.1.2 Sampling	
9.1.3 Disconnection of the reactor water cleanup system	
9.1.4 Airborne radioactivity	
9.2 Management of leakage water	61
9.2.1 Reactor building	
9.2.2 Waste plant	
9.3 Methods for cleaning the process water	62
9.4 Handling and treatment of additional wastes	64
10. PROPOSALS FOR COMPLEMENTARY ADDITIONS IN THE REACTOR BUILDING AND IN THE WASTE BUILDING	65
10.1 Introduction	65
10.2 Complementary additions in the reactor building	65
10.2.1 Controlled drain systems	
10.2.2 Other measures	
10.3 Complementary additions in the waste plant	66
10.3.1 Liquid waste system	
10.3.2 Solidification system and mould handling	
11. DISCUSSION AND CONCLUSIONS	69
12. ACKNOWLEDGEMENTS	72
13. REFERENCES	73
APPENDICES	
Appendix 1. List of systems mentioned in this report	
Appendix 2. Existing cleanup systems	
Appendix 3. Some building layouts	
Appendix 4. Doses to the environment after releases from the reactor containment	

## 1. INTRODUCTION

The severe reactor accident at the Three Mile Island nuclear power plant in the USA has been the cause of a large number of investigations, research projects and development work in order to understand the course of events related to such accidents and to find measures to mitigate them. Most of the efforts have, for obvious reasons, concerned the securing of the core cooling and the prevention of radioactive emissions to the environment.

Fission products released from the damaged core are unavoidably dispersed in the reactor building due to the loss of integrity of the confining systems, for example the reactor pressure vessel. They can also be found outside the reactor containment due to the circulation of cooling water. The presence of radioactive substances in large amounts in places where this is not designed for is of great concern.

The resulting ionizing radiation affects not only the accessibility of the plant personnel but has also an adverse influence on organic materials used in gaskets and as cable insulation. It will be more difficult to maintain the reactor control in the long run, if the access of plant personnel to vital rooms is restricted or impossible and if component leakage of highly contaminated water increases beyond the acceptable because of the breakdown of gaskets. Consequently, it is very important to take care of the fission products released and safely dispose of them as soon as possible. But this aspect on severe reactor accidents has not attracted large interest so far.

Nuclear reactors have cleanup systems to maintain the specified high water quality required for a safe and reliable operation. The ion exchangers and filters used for this purpose also trap radioactive material, predominantly activated corrosion products released to the cooling water from the fuel rod surfaces. In case the cladding of the fuel rods is damaged, fission products - except for the noble gases krypton and xenon - are also retained in the water cleanup systems.

Each reactor is connected to a waste plant, where spent resins and filter aids from the cleanup systems are taken care of, solidified in cement or bitumen and eventually transported to a final repository for radioactive waste. Radioactively contaminated leakage and drain water is also treated in the waste plant, resulting in the same waste as from the above mentioned cleanup systems.

Since reactor waste plants are designed to annually receive a certain amount of specified radionuclides, it is of great interest to study if they are capable of handling the radioactive waste produced after events, where fuel damages suddenly arise. A study with this purpose has previously been conducted by ABB Atom for the Swedish Nuclear Power Inspectorate and within the Nordic nuclear safety programme (ELK 85). The fuel damage was moderately large, corresponding to a release of about 0.1 % of the core inventory of fission products of most interest in the waste management.

The main conclusion from this study was the following. The waste plants at Nordic nuclear power plants with Boiling Water Reactors (BWRs) are constructed, and have a capacity, such that they can treat the radioactive waste resulting from hypothetical reactor accidents which may give rise to moderately large fuel damage.

The promising result led to a new study aimed at a more severe, hypothetical reactor accident in a BWR plant, where the fuel damage is such that 10 % of the core inventory of noble gases is released. The objective is to find out to what extent the present waste plants can be used to manage a larger, hypothetical reactor accident and if any complementary additions to these plants are required for this purpose.

Like the previous study, this study has been administered by the Swedish Nuclear Power Inspectorate and carried out by ABB Atom. The study is part of the Nordic nuclear safety programme sponsored by NKA, the Nordic Liaison Committee for Atomic Energy.

After a prestudy it was decided to select one accident scenario for the further study, but also to devote some time to a second one. In the first case (denoted AB<sub>m</sub>) the starting event is a pipe break causing a loss of coolant accident (LOCA), followed by the complete loss of all electric power. The second scenario (denoted TB<sub>m</sub>) is merely a complete loss of all electric power, starting with the outer net. The subscript means that the sequence leading to a core melt-down is stopped after a while, when the electric power has been restored - estimated for the AB<sub>m</sub> case to be about half an hour after the starting event - and eventually the core cooling reestablished.

It should be noted here, that the scenarios chosen are very unlikely and serve only as a background in order to provide the above mentioned release of fission products. The project is a study of the waste management after a reactor accident and accident management is treated only briefly and where appropriate.

In the AB<sub>m</sub> case the reactor containment is filled with water to secure the core cooling. This means that a volume of about 10 000 m<sup>3</sup> of highly radioactive water will have to be treated in the waste plant, and also that the free surfaces in the containment will be radioactively contaminated, which is of great interest from the waste management point of view. In the second case the radioactive matter released from the core is principally confined to the reactor vessel, leading to the much smaller volume of about 250 m<sup>3</sup> of water with a very high radioactive content.

The AB<sub>m</sub> case is of a greater interest than the TB<sub>m</sub> case regarding waste management, even though the TB<sub>m</sub> case has a much higher probability to actually occur. However, in absolute figures even the probability of a TB<sub>m</sub> event is very low. Also, although the water concentration of radionuclides will be much higher in the TB<sub>m</sub> case, in principle the water cleanup does not deviate from that in the first case.

In order to make the study more realistic it was decided to have a reference reactor. Due to the characteristics of the AB<sub>m</sub> case it had to be a BWR with external recirculation loops. With permission of the OKG utility the second reactor unit at the Simpevarp site - Oskarshamn 2 - was chosen. It was also decided to add to the study the prerequisite to restore the reference reactor to such an extent that it can eventually operate, and to have the first reactor unit, sharing the same waste plant, operating during the restoration work.

The dominating radionuclides in the contaminated water three months after an  $AB_m$  event - the earliest time when the cleanup is supposed to start - are the two cesium isotopes Cs-134 (half-life 2.06 years) and Cs-137 (half-life 30 years). Their total radioactivity at this moment has been estimated to be  $2.7 \cdot 10^4$  TBq. Compared with the damaged second reactor unit at Three Mile Island (TMI-2) this is about the same amount of cesium isotopes released.

Practically, this is also the only similarity between the real reactor accident and the assumed accident in the reference reactor in this study. TMI-2 is a Pressurized Water Reactor (PWR) with a very short operating life, where a partial melt-down took place. As in all reactors of this type the reactor containment is much larger than in a BWR, and after the accident most of the volume was not filled with water. But there was water in the containment basement and here a substantial difference in contamination was found between concrete surfaces covered or uncovered by the highly contaminated water. The experience from TMI-2, given in a large number of reports but to a large extent summarized in (ANS 88), has been found very useful for this study.

The dominance of cesium isotopes in the water has led to an investigation of using cesium selective inorganic sorbents instead of the organic resins normally used for cleanup purposes (Chapter 6). The best way of using the existing cleanup systems and the need of modifications has been thoroughly studied and so has the subsequent immobilization of the spent sorbents and the handling of the waste containers produced (Chapters 7 and 10). The final disposal of the waste has not been included in the study, but a few matters of special interest have been examined.

Included in the waste management is also the additional waste generated during surface decontamination and replacement of contaminated equipment in the reactor containment (Chapter 8).

Some difficulties have been met in clearly defining the study with respect to accident management. There are some connections with waste management, the most important of which seems to be the handling of contaminated leakage and drain water during the period immediately following an  $AB_m$  event. One minor part of the study has been devoted to this and some other issues of interest in waste management (Chapter 5).

The  $TB_m$  case has only been briefly outlined with emphasis on the differences between the two cases with respect to waste management and the above mentioned issues (Chapter 9).

The introductory Chapters 2, 3 and 4 of the report deal with the selection of main event and prerequisites of the study, the source term establishment, and the radiation calculations required for different parts of the study, respectively. The conclusions of the study are given in Chapter 11.

This final report is essentially based on a number of Technical Reports (TR1-TR9) presented during the course of the project.

## 2. MAIN EVENT AND ASSUMPTIONS

### 2.1 Reactor accidents. Analysis of two types of accident

During normal reactor operation the core is in a state of balance between the heat generated by nuclear fission and the cooling provided by the water in the primary circuit. This balance is maintained by regulating the power output of the reactor by means of the control rods and/or the main recirculation pumps.

A reactor accident where radioactive matter may be released from the fuel could result if the state of balance in the core is disturbed by insufficient cooling. The emergency cooling systems installed for the purpose of hindering any consequences from loss of coolant are discounted in this study or considered to be of insufficient capacity. (The purpose of the emergency cooling systems 322 and 323 are to automatically make good any loss in coolant should the capacity of the main recirculation system either become insufficient or fail completely. See also Figure 2.1.)

Loss of coolant will prevent any further chain reaction in the fuel as the cooling water in a Light Water Reactor (LWR) is also the moderator. This does not mean that heat generation is brought to a complete standstill. The decay of the radioactive fission products will continue to generate heat. This is referred to as residual heat. A total loss of coolant will result in a heat build-up in the core eventually leading the core to melt. As the core heats up the fuel rod cladding and the fuel matrix within the fuel rod will be destroyed. This will result in the fuel rods no longer retaining the fission products and a considerable quantity of radioactive materials will be released.

The RAMA Report (JOH 85) analyzed the course of events where the core is completely melted, melts through the reactor pressure vessel and finally causes the collapse of the reactor containment. This accident analysis provides a suitable basis for the chain of events which would result in damage to the core to the extent considered in this report. The suggestion is that the initial events causing the accident are as those referred to above but that the situation was prevented from developing by the recovery of the core cooling system. This means that the destruction of the fuel rods and subsequent melting of the core were prevented.

Two of the five accident types described in the RAMA Report have been selected for a more detailed analysis. These are:

- Loss of Electric Power (TB)

The main event is failure of the external electricity supply network. Reserve power from either gas turbine or diesel driven equipment has also failed. The power required to close the isolation valves so as to isolate the reactor, and to reduce pressure by means of the pressure relief system, is provided from batteries.

Failure of the electricity supply results in the fast shut down and isolation of the reactor. The absence of power also removes any possibility of cooling the core. The water in the pressure vessel is boiled off due to residual heat and the core is uncovered from water. The steam that is generated is released by the relief (blow down) system. The core begins to be uncovered after approximately 30 minutes, builds up heat and starts to melt after approximately 2-3 hours.

- LOCA and Loss of Electric Power (AB)

The main event is the fracture of a pipe in the primary cooling system within the reactor containment. The situation regarding the supply of electric power is similar to that given above for TB.

Fast shut down occurs and the reactor is isolated. The absence of power means that the core cannot be cooled. The absence of cooling and the pipe fracture result in the reactor being emptied as the water and steam flow out into the drywell (primary containment). The core heats up and eventually begins to melt in a similar manner to the TB case, but in this case the events occur more quickly. The core starts to be uncovered in a matter of seconds and commences to melt after approximately 1 hour.

The selected accident sequence is prevented from progressing before parts of the core have melted. This is a consequence of the basic assumption of the study, that 10 % of the core content of volatile fission products shall be released. When parts of the core have reached melting point, volatile fission products have been released from the fuel in such a quantity that the stated condition has been exceeded.

The TB and AB cases are modified, therefore, to allow for the recommencement of cooling of the core after a period of time which is comparable to the period required for 10 % of the core content of the noble gases krypton and xenon to be released. The modified accident sequences are identified as TB<sub>m</sub> and AB<sub>m</sub>, respectively.

## 2.2 Waste management aspects of the accidents. Selection of main event

The modified cases TB<sub>m</sub> and AB<sub>m</sub> described above, will be discussed here in regard to waste management.

In the TB<sub>m</sub> case the released radioactivity will be circulated in a circuit between the reactor vessel and the condensation pool, and will start shortly after recommencement of core cooling. The assumption can then be made that the shutdown cooling system will be available for cooling of the core. This means that the radioactive matter that has been released will be retained in the reactor and in this system and any further release to the condensation pool will be appreciably restricted.

During the time when the water circulates between the reactor and the condensation pool the gaseous fission products, mainly noble gases and methyl iodide, will be released to the drywell and wetwell in the containment. The greater proportion of the fission products will be found in the aqueous phase from which they will most likely be deposited on the inner surfaces of the condensation pool, reactor vessel, and the emergency cooling systems. The gaseous fission products will eventually fill the containment from which they can be released when convenient.

The situation in the AB<sub>m</sub> case is completely different (Figure 2.1). In this case the fission gases, gasborne radioactive particles (aerosols) and waterborne radioactivity (originally aerosols) leak into and contaminate the drywell. It is anticipated that a considerable amount of the released activity will be deposited on the free surfaces in the drywell. Otherwise, the activity will behave in a similar manner to what happens during the first time period in the TB<sub>m</sub> case.

The  $AB_m$  case is of far greater concern in regard to waste management because of the contamination of the free surfaces in the drywell. In the  $TB_m$  case a part of the radioactive material is absorbed by the water in the condensation pool and the remainder is found in the reactor vessel and the shutdown cooling system, which should simplify waste management.

Spraying of the reactor containment results in a reduction of the activity level of the atmosphere in the reactor containment, especially in regard to aerosol bound iodine. This means a considerable reduction in the amount of radioactive material that is available for release from the containment and that can possibly reach the environment. There is however, a drawback. The water for the containment vessel spray system is drawn from the condensation pool which means that the surfaces in the containment will be contaminated by waterborne radioactive particles. The drywell is already contaminated in the  $AB_m$  case as a result of the pipe fracture. In this case the containment spraying will not cause any further damage which could affect waste management.

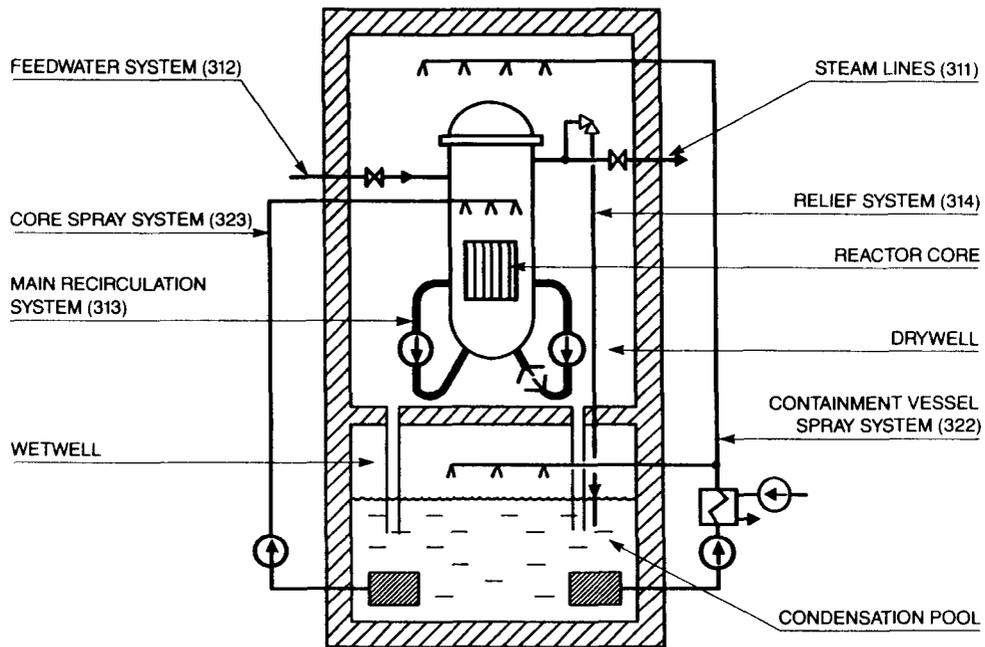


Figure 2.1 The  $AB_m$  case a short time after the event.

The spraying of the containment is an extremely unfortunate measure in regard to waste management in the TB<sub>m</sub> case, as the drywell will now become contaminated. However, one positive measure will most likely be the reduction of aerosols in the containment. Meanwhile, after blow down it is likely that the major part of the aerosols has already been transferred to the aqueous phase in the condensation pool. This means that the importance of containment spraying may be doubtful when considering the effect on the environment.

The foregoing serves to indicate that the most appropriate case to consider is an accident sequence as case AB<sub>m</sub>. This is most interesting from the point of view of waste management, as the free surfaces of the containment will become contaminated whereas this is not expected to happen in the TB<sub>m</sub> case, at least not to the same extent.

Although the TB<sub>m</sub> case has a much higher probability to occur than the AB<sub>m</sub> case, the selection seems to be justified for this study. In absolute figures, even the probability of a TB<sub>m</sub> case is very low.

It can also be noted here, that the AB<sub>m</sub> case covers the case when the core spray system is not available after a LOCA for reasons other than the loss of electric power. The complete loss of all electric power (station blackout), which includes the capacity in reserve from the diesel engines, has occurred for 11 minutes in an US BWR in 1984.

### 2.3 Comparison with TMI-2

The severe accident in the second reactor unit at the Three Mile Island nuclear power plant (TMI-2) in the USA is at present the only available experience of such accidents in a light water power reactor. During the ten years that have elapsed since the occurrence, the understanding of the accident has grown and so has the data from the cleanup procedure. All this knowledge has been reported in technical journals and in separate reports, and has also been summarized at the topical meeting held concurrently with the ANS Winter Meeting 1988 (ANS 88).

TMI-2 is a PWR with a short operating time, about 2 to 3 full power months - the burnup achieved was 3 200 MWd/ton uranium. The removal of the reactor internals and the core debris has revealed that about half of the core (around 70 tonnes) melted during the accident and that some 20 tonnes of the original core material relocated to the bottom of the reactor vessel. In the reference reactor there will be no fuel melting since only 10 % of the core inventory of noble gases is released.

The calculated inventory of Cs-134 and Cs-137 in the TMI-2 core is  $7 \cdot 10^3$  and  $3.3 \cdot 10^4$  TBq, respectively. Due to the very short operating time this is much lower than in the reference reactor, where the corresponding values are  $2.2 \cdot 10^5$  TBq and  $1.8 \cdot 10^5$  TBq, respectively. In TMI-2 about 75 % of the core inventory of cesium was released from the fuel and since the corresponding value for the reference reactor has been estimated to 7 % (cf. Section 3.1), the amount released of these two cesium isotopes is about equal in the two reactors.

PWRs have a much larger reactor containment (often called reactor building) than BWRs. After the accident most of the containment volume in TMI-2 was not filled with water, while this is the case after an  $AB_m$  event (on purpose for other reasons). About 40 % of the cesium inventory in the core of TMI-2 was dispersed in water in the reactor containment basement. Here, a substantial difference in contamination was found between concrete surfaces covered or uncovered by the water and also depending on the coating and condition of the surface. The contamination data has been found to be very useful in this study (cf Chapter 8).

#### **2.4 Reference reactor**

At the start of the study it was decided to select a reference reactor with the intention to make the study more realistic. Due to the characteristics of the  $AB_m$  case it had to be a BWR with external recirculation loops. After permission of the OKG utility the second reactor unit at the Simpevarp site - Oskarshamn 2 - was chosen as the reference reactor. This unit shares the waste plant with the first reactor unit at the site.

A couple of prerequisites were included in the study. First, the goal of the restoration is to make it possible to operate the reference reactor. Secondly, the first reactor unit is operating during the restoration work. The third reactor unit at the site, located at a distance of some hundred metres from the first two units and having a waste plant of its own, is assumed to operate as usual.

As regards the measures taken in the reference reactor after the  $AB_m$  event, it is supposed that no boric acid is fed into the reactor.

Three appendices are included in this report to make it easier for a reader not familiar with BWR technology and the reference reactor in particular. Appendix 1 is a list of systems mentioned in the report. Appendix 2 contains a brief description of the cleanup systems of interest. Appendix 3 includes some selected building layouts to visualize parts of the reactor and waste buildings.

### 3. THE SOURCE TERM ESTABLISHMENT

#### 3.1 The $AB_m$ case

##### 3.1.1 Release of fission products

Five basic mechanisms govern the release of fission products during a reactor accident, namely:

- 1) Leakage due to puncture of the cladding, i.e. burst release,
- 2) Diffusion from the fuel-cladding gap,
- 3) Release from grain boundaries,
- 4) Diffusion from the  $UO_2$ -grains,
- 5) Release from melted fuel.

Each mechanism will successively be dominating as the temperature of the fuel increases.

Burst release is possible when the temperature of the cladding has reached 800-900°C. At the same time as the cladding is heated and becomes soft, the pressure in the reactor vessel is automatically relieved in order to make it possible to use the core spray system. But when the pressure in the reactor vessel drops, the over-pressure in the fuel rod caused by the fission gas forces a pinhead blister to be formed in the heated cladding. The blister is normally less than 0.1 mm in diameter. A fuel damage caused by a LOCA is quite different from the fuel damage encountered during normal operation due to wear and tear where pinholes much larger than formed at a LOCA are expected. When the overheated fuel cladding is punctured the total amount of noble gases present in the gap is expected to be released. There can be quite a large variation of the released amount (from ca 0.1 to 10 % of the total noble gas inventory in the rod), where the amount released mainly is due to the total burn-up of the fuel and to the half-life of released radionuclides. Small portions of other radionuclides - especially cesium and iodine - are also released. Included in burst release are gas atoms present on or close to the surface of the fuel and cladding.

Diffusion from the gap is a slow process in which basically iodine and cesium diffuse from the gap between fuel and cladding through holes in the cladding formed during the accident. The total amount released is substantially smaller than from the burst release.

The release from grain boundaries starts at a fuel temperature of around 1350°C. The main causes of release are formation, swelling and merging of fission gas bubbles. The expanded bubbles physically separate the grains, thus, forming escape channels out of the fuel pellets. Approximately equal amounts of noble gases, iodine and cesium are released, given as fractions of the total inventory. This released fraction can for a high burn-up fuel amount to 20 %.

Diffusion from individual fuel grains is insignificant in comparison to burst release at temperatures around 1000°C and below, but becomes continuously more important as the temperature increases. A temperature increase of 100°C approximately doubles the diffusional release rate of fission products from the grains. For example, at a fuel temperature of 2000°C the release rate of iodine, cesium and the noble gases is about 10 % per minute of the core inventory.

The release of fission products from melted fuel is a complicated process, which so far is incompletely known. The analysis is made more difficult because chemical transitions take place simultaneously with the melting process, e.g. melting and oxidation of Zircaloy occur coincidentally during many hypothetical accident sequences. The melting point of Zircaloy is at 1800°C and that of the oxidation product  $ZrO_2$  is at 2700°C. Thus, the relative rates of melting and oxidation of the cladding material have an utmost impact on the overall release process.

These five release mechanisms are dependent on a number of parameters, e.g. the burn-up, the fuel density, the  $UO_2$  grain size, the half-lives of the radionuclides, temperature and time. Of these parameters temperature and time are most important.

Based on experimental data the rate of the fission product release has been quantitatively determined (NUR 81). The release of different radionuclides is given in Figure 3.1 as  $F$ , the released fraction of total inventory per unit of time ( $\text{min}^{-1}$ ) versus the temperature. Note that burst release is not included in the figure.

The amount of fission products released from various parts of the core can be predicted if  $F$  is known at different locations in the core and then integrated over a certain time.

The total release - the source term - is obtained by adding releases from all parts of the core and for all radionuclides.

Calculated release rates for noble gases are given in Figure 3.2. The figure shows the released fraction of radioactive noble gases with long half-lives as a function of the duration of the accident, i.e. the time from the accident begins until the core is permanently cooled again. After approximately 45 minutes (i.e. before the core starts to melt) 10 % of the long-lived noble gas inventory has been released, 3 % of this due to burst release and 7 % of the release due to other mechanisms.

It is important to remember that 45 minutes is the time until a continuous cooling has once again been established. Electric power has been supplied earlier, thus starting the core spraying. The time needed to cool the core from the start of the core spray is about 10 minutes, which means that electric power is recovered 35 minutes after the cooling has been lost. The times given for the accident propagation are approximate and more recent models, including non-linear heat evolution in the fuel, etc, predict a more rapid process where cooling is necessary within half an hour in order to prevent serious fuel damage.

In Table 3.1 the fractions released for all long-lived radionuclides of interest are given. The fraction released is estimated from Figure 3.1. For elements not represented in Figure 3.1, an estimate has been made based on the literature.

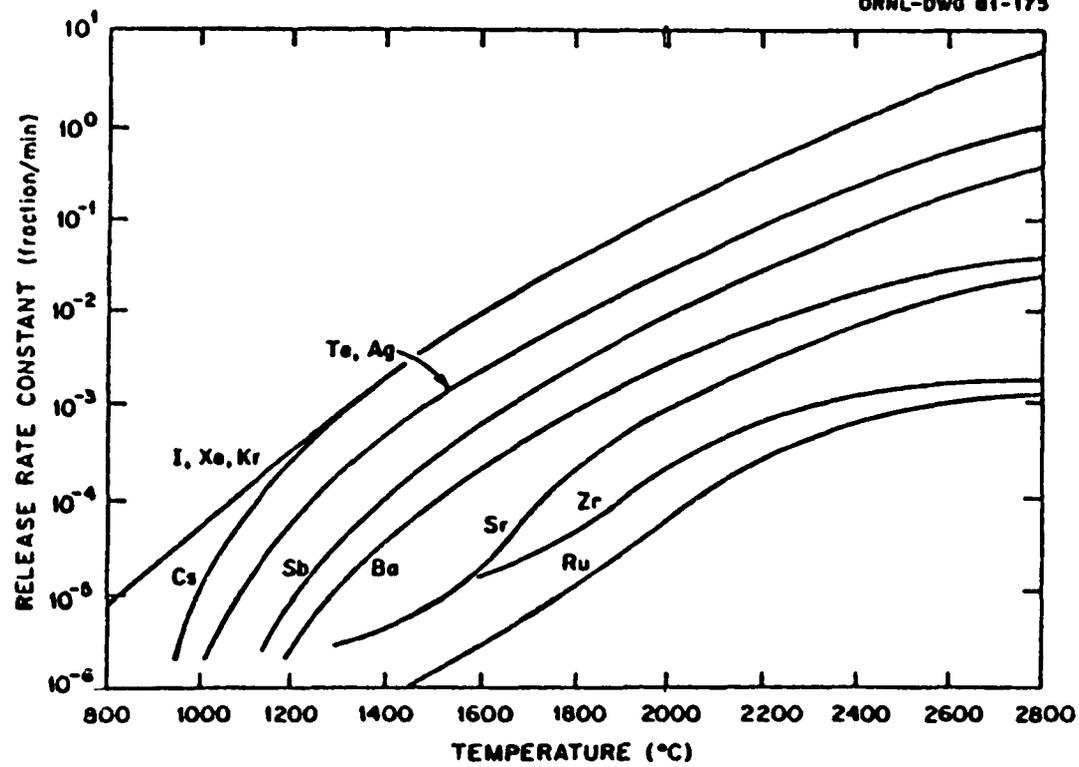


Figure 3.1. Fission product release rates from uranium fuel.

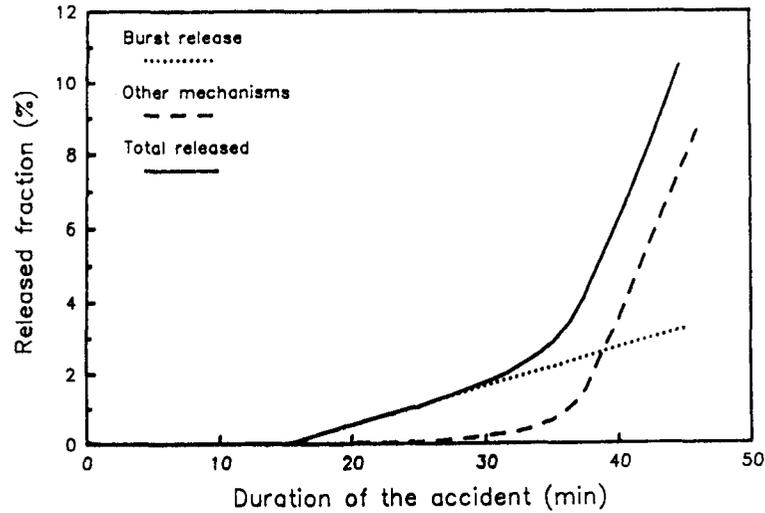


Figure 3.2. The released fraction of noble gases given in % of total core inventory.

The calculated source term for the most important radionuclides is given in Table 3.2. 45 minutes is used as reference time, i.e. the time when cooling of the core is regained. In the table the amount of released radionuclides after one day, one week and three months is given. These times are considered interesting with respect to accessibility of the plant and to waste handling.

As can be seen in Table 3.2 the core source term is initially dominated by iodine, the noble gases and cesium, representing 40%, 38% and 11%, respectively, of the total radioactive content of  $2.4 \cdot 10^{18}$  Bq. After 24 hours of decay the noble gases (55 %) and iodine (35 %) are dominating and after three months the radionuclides Cs-134 and Cs-137 are making up for 85 % of the total radioactivity.

Table 3.1. Released fractions given as % of the total core inventory.

Element	Released fraction (%)
I, Cs	7
Te, Ag	1.5
Sb	0.5
Sr, Ba	0.19
Mo	0.15
Y	0.05
Zr, Nb	0.015
Ru, Rh, Pd, La	0.004
Ce, Pr, Nd, Pm	

Table 3.2. The source term for the AB<sub>m</sub> case and Oskarshamn 2.

Nuclide	Half-life	Source term (Bq)			
		45 min	1 day	1 week	3 months
Kr-83m	1.83 h	1.9·10 <sup>16</sup>			
-85m	4.48 h	4.9·10 <sup>16</sup>	1.3·10 <sup>15</sup>		
-85	10.7 y	1.8·10 <sup>15</sup>	1.8·10 <sup>15</sup>	1.8·10 <sup>15</sup>	1.8·10 <sup>15</sup>
-87	76 min	7.8·10 <sup>16</sup>			
-88	2.84 h	1.2·10 <sup>17</sup>	4.1·10 <sup>14</sup>		
Rb-88	18 min	7.8·10 <sup>16</sup>	4.6·10 <sup>14</sup>		
Sr-89	51 d	3.6·10 <sup>15</sup>	3.6·10 <sup>15</sup>	3.3·10 <sup>15</sup>	1.1·10 <sup>15</sup>
-90	29 y	2.7·10 <sup>14</sup>	2.7·10 <sup>14</sup>	2.7·10 <sup>14</sup>	2.7·10 <sup>14</sup>
Y-90	64 h	3.5·10 <sup>11</sup>	6.2·10 <sup>13</sup>	2.3·10 <sup>14</sup>	2.7·10 <sup>14</sup>
-91	59 d	1.2·10 <sup>15</sup>	1.2·10 <sup>15</sup>	1.1·10 <sup>15</sup>	4.1·10 <sup>14</sup>
Zr-95	64 d	4.5·10 <sup>14</sup>	4.4·10 <sup>14</sup>	4.2·10 <sup>14</sup>	1.7·10 <sup>14</sup>
Nb-95	35 d	4.4·10 <sup>14</sup>	4.4·10 <sup>14</sup>	4.4·10 <sup>14</sup>	2.8·10 <sup>14</sup>
Te-127m	109 d	2.8·10 <sup>14</sup>	2.8·10 <sup>14</sup>	2.7·10 <sup>14</sup>	1.6·10 <sup>14</sup>
-127	9.4 h	1.9·10 <sup>15</sup>	5.7·10 <sup>14</sup>	2.7·10 <sup>14</sup>	1.6·10 <sup>14</sup>
-129m	34 d	1.4·10 <sup>15</sup>	1.4·10 <sup>15</sup>	1.2·10 <sup>15</sup>	2.2·10 <sup>14</sup>
-129	69 min	6.0·10 <sup>15</sup>	8.8·10 <sup>14</sup>	7.8·10 <sup>14</sup>	1.4·10 <sup>14</sup>
-132	78 h	3.7·10 <sup>16</sup>	3.0·10 <sup>16</sup>	8.4·10 <sup>15</sup>	
-134	42 min	3.1·10 <sup>16</sup>			
I-131	8.04 d	1.2·10 <sup>17</sup>	1.1·10 <sup>17</sup>	6.5·10 <sup>16</sup>	4.8·10 <sup>13</sup>
-132	2.3 h	1.6·10 <sup>17</sup>	3.1·10 <sup>16</sup>	8.6·10 <sup>15</sup>	
-133	21 h	2.5·10 <sup>17</sup>	1.1·10 <sup>17</sup>	9.5·10 <sup>14</sup>	
-134	53 min	2.1·10 <sup>17</sup>			
-135	6.6 h	2.3·10 <sup>17</sup>	2.0·10 <sup>16</sup>		
Xe-131m	11.8 d	2.3·10 <sup>15</sup>	2.2·10 <sup>15</sup>	1.7·10 <sup>15</sup>	1.8·10 <sup>13</sup>
-133m	2.2 d	1.1·10 <sup>16</sup>	9.0·10 <sup>15</sup>	1.6·10 <sup>15</sup>	
-133	5.25 d	3.6·10 <sup>17</sup>	3.4·10 <sup>17</sup>	1.6·10 <sup>17</sup>	2.5·10 <sup>12</sup>
-135m	15.6 min	6.6·10 <sup>16</sup>	5.7·10 <sup>15</sup>		
-135	9.1 h	1.1·10 <sup>17</sup>	7.2·10 <sup>16</sup>		
-138	14.1 m	8.4·10 <sup>16</sup>			
Cs-134	2.1 y	1.6·10 <sup>16</sup>	1.6·10 <sup>16</sup>	1.6·10 <sup>16</sup>	1.4·10 <sup>16</sup>
-136	13.1 d	3.6·10 <sup>15</sup>	3.5·10 <sup>15</sup>	2.5·10 <sup>15</sup>	3.1·10 <sup>13</sup>
-137	30 y	1.3·10 <sup>16</sup>	1.3·10 <sup>16</sup>	1.3·10 <sup>16</sup>	1.3·10 <sup>16</sup>
-138	32 min	2.0·10 <sup>17</sup>			
-139	9.5 min	3.0·10 <sup>16</sup>			
Remaining		balance	balance	balance	balance
Total		2.4 ·10 <sup>18</sup>	7.8 ·10 <sup>17</sup>	2.9·10 <sup>17</sup>	3.2·10 <sup>16</sup>

During the cleanup process both cesium and strontium will be leached from the punctured fuel pins. With a leach rate of about  $5 \cdot 10^{-5}$  and  $10^{-6}$  fractions of total inventory per day for cesium and strontium, respectively, a maximum of 500 TBq cesium (as Cs-134/Cs-137) and 0.1 TBq strontium (as Sr-90) would be released in 1 year.

### 3.1.2 Release of actinides

Uranium and transuranic elements (that is, basically neptunium, plutonium, americium and curium) can be released to the reactor water by in particular two mechanisms:

- 1) dissolution of uranium dioxide and consequent leaching from the damaged fuel during shutdown and cleanup after a reactor accident.
- 2) due to particles stripped from the fuel when the fission gas is ejected from the cladding during a burst release.

Table 3.3 gives the expected amounts of some actinides that will be released from the fuel.

The dissolution of  $UO_2$ -fuel in water is temperature dependent. At normal operating temperature ( $286^\circ C$  at 70 bar) in a BWR the  $UO_2$  is oxidized and dissolved in contact with oxygen rich cooling water. During operation a fuel dissolution of about 1 % uranium has been observed after fuel damages after ca 2-3 months of operation. At water temperatures below  $100^\circ C$  - a temperature which is expected in the cooling water as soon as the emergency cooling is in operation - the dissolution of uranium dioxide is considerably less.

R. Forsyth and L.O. Werme (FOR 86) have performed fuel leaching experiments at room temperature using spent fuel, in which they found a maximum uranium concentration in solution of ca 1 ppm (1 mg/l) U. Decreasing leaching rates have been observed after fuel damages. The reason is probably that a formation of an oxide layer enhances the transport resistance.

With a uranium solubility of 1 mg/l a maximum of ca 24 mg uranium can be leached per year from a reactor core with a size corresponding to the core in Oskarshamn 2 if 10 % of the core has been damaged. The total release of 24 mg uranium was estimated using a combined dissolution/diffusion process (TR1). Due to diffusion resistance in the formed oxide layer in the gap between fuel and cladding 24 mg/year uranium is to be considered as a maximum level.

In a burst release, fragments are stripped from the fuel and released together with the gas. The fraction released as particulate matter in a burst release has been reported to be between 0.002 to 0.02%, where by comparing with what was found in TMI-2 the former value seems more valid (MAL 79).

**Table 3.3. Estimated amounts of actinides released from the fuel after an AB<sub>m</sub> event.**

Nuclide	$t_{1/2}$ (y)	Weight (g)	Activity (Bq)
U-235	$7.04 \times 10^8$	2	$1.6 \times 10^5$
U-238	$4.47 \times 10^9$	150	$1.8 \times 10^6$
Np-237	$2.14 \times 10^6$	0.1	$2.6 \times 10^6$
Pu-239	$2.439 \times 10^4$	1	$2.3 \times 10^9$
Am-241	433	0.01	$1.3 \times 10^9$

For Oskarshamn 2 a loss of 0.002 % of the fuel would give a maximum of 0.16 kg uranium. After a burn-up of 28 500 MWd/t approximately 0.15 kg (95 %) would be <sup>238</sup>U; about 1.9 g of the rest would be other uranium isotopes, 1.4 g plutonium isotopes, a little more than 0.1 g other actinides and the rest (ca 4.5 g) is fission products (AKA 76). With a solubility of uranium of 1 mg/l this means that all released uranium could be dissolved in a water volume of 10 000 m<sup>3</sup>.

Fuel leaching studies have shown the total amount of plutonium dissolved from spent fuel to be 1 ppb (1/ug/l) (FOR 86). This means that for Oskarshamn 2 a total of 0.024 mg plutonium would be dissolved from the core during one year (TR1).

But when plutonium is found in deposits on the reactor core or otherwise in the reactor vessel it probably originates from fuel fragments torn from the core in connection with a burst release.

With a maximum solubility of 1 ppb in 10 000 m<sup>3</sup> water all released plutonium would be dissolved in the water, but thermodynamic equilibrium calculations give that approximately 0.08 g plutonium should be soluble in 10 000 m<sup>3</sup> water using the most probable chemical conditions in the containment, i.e. Pu(IV), pH 7-9, <100°C and comparatively low carbonate content in the water. The equilibrium constants for plutonium were chosen from (ALL 84).

One of the best indications of a fuel damage under normal operation is when neptunium can be detected in the cooling water. Neptunium exists in oxidation state +5 in solution and forms the easily soluble NpO<sub>2</sub><sup>+</sup> ion, thus all released neptunium can be expected in the water phase.

Americium and curium are most likely leached congruently with uranium during plant operation, and in the same way as for plutonium they will together with uranium be redistributed in crud predominantly on the fuel cladding.

### 3.2 The TB<sub>m</sub> case

#### 3.2.1 Release of fission products

In the TB<sub>m</sub> case the progress of events like uncovering and heating the core are slower compared to the AB<sub>m</sub> case. This could lead to an accumulation of fission gas at the grain boundaries and an increased release compared to the AB<sub>m</sub> case if the cladding eventually is punctured. But if there is an accumulation of fission gases at the grain boundaries it is probable that the noble gases and other gaseous elements, e.g. iodine and cesium at high temperature, are equally mobile and therefore a noble gas release of 10 % should lead to the same release of other fission products in the TB<sub>m</sub> case, giving a fission gas composition similar to the AB<sub>m</sub> case. Thus, a release of fission products larger than in the AB<sub>m</sub> case is not expected. All other accidental events, giving a lower temperature increase, would lead to releases less than in the AB<sub>m</sub> case and consequently Table 3.2 represents also a worst TB<sub>m</sub> case.

#### 3.2.2 Release of actinides

The maximum released amounts of uranium and transuranic elements in the TB<sub>m</sub> case is the same as in the AB<sub>m</sub>-case (cf. Table 3.3). All other events causing less serious fuel damages lead to smaller releases of actinides than in the AB<sub>m</sub> case.

### 3.3 Chemical conditions and expected distribution of released radionuclides between gas phase, liquid phase and solid surfaces

#### 3.3.1 Chemical speciation and sorption of fission products

Strontium is incorporated in the uranium matrix as oxide and does not exhibit the same transport to grain boundaries and gap as has been observed for the noble gases, cesium and iodine. Hence, there is only a limited release of strontium during burst release and any substantial releases of strontium are expected only when the uranium matrix is leached. Most of the dissolved strontium remains in the water phase and is removed from solution in the waste treatment facility. Due to exchange with calcium a small sorption on concrete walls is expected due to chemical bonding (chemisorption).

It cannot be excluded that a small part of the iodine - <0.1 % of the core inventory as organic iodine - can exist in the gas phase even after the containment has been water filled. When released from the fuel, part of the iodine can be released as elemental iodine, I<sub>2</sub>(g), and methyl iodide, CH<sub>3</sub>I. The elemental iodine will be dissolved in the water when bubbled through the condensation pool but in the organic form as methyl iodide, iodine will stay in the gas phase and eventually together with noble gases be vented to the atmosphere. Iodine is anionic and soluble in oxic water, predominantly as iodide (I<sup>-</sup>). An insignificant surface sorption is expected, hence, all that is released from the fuel (except the gas phase fraction) is removed from the cooling water by anion exchange in the cleanup system.

Initially, cesium is released to the gas phase as CsI, Cs<sub>2</sub>Te or CsOH, but is expected to be instantly condensed on surfaces in the containment (BEA 87, COL 88). After the TMI-2 accident it was found (BAS 86) that when cesium was released and the temperature in the containment was high (ca 300°C) cesium silicates were formed on surfaces inside the reactor vessel. Cesium in solution exists as the monovalent cation Cs<sup>+</sup>. The total surface sorption of cesium, on surfaces above the water level in the containment at TMI-2, was estimated to 0.0016 % of the core inventory (BEA 87). A higher sorption was measured on surfaces that had been below water for a prolonged time. Because a complete cleanup of the lower levels at TMI-2 has not yet been performed, only preliminary figures on the surface sorption are so far available. But as a rough estimate it seems like approximately 1 % of total released cesium is sorbed on painted surfaces having been exposed to contaminated water for a long time.

At TMI it was found (BEA 87) that at high temperatures (>600°C) tellurium formed a tellurate together with the Zircaloy cladding and insignificant amounts of tellurium was released as long as the Zircaloy cladding was not completely oxidized. When the Zircaloy was oxidized tellurium was released to the gas phase predominantly as SnTe, and in the absence of Zircaloy as Te<sub>2</sub>, H<sub>2</sub>Te and Cs<sub>2</sub>Te. In all cases tellurium was immediately condensed on surfaces in the containment. Cesium telluride and hydrogen telluride are decomposed in water and metallic tellurium is formed. In oxalic water tellurium is oxidized to the +4 state and a sparingly soluble tellurium oxide, TeO<sub>2</sub>, is formed. In TMI-2 93 % of the tellurium was found in bottom sludges and on surfaces, while 6.9 % (ca 2.5 mg/l) was in solution.

### 3.3.2 Chemical speciation and sorption of actinides

In spent fuel, the actinides are incorporated as oxides with very low solubility. Uranium, neptunium and plutonium are tetravalent (An(IV)) and form AnO<sub>2</sub>(s), while americium and curium are trivalent (An(III)) and form the oxide An<sub>2</sub>O<sub>3</sub>(s). No actinides are released in the gas phase or initially in the aqueous phase, but in connection with a burst release fuel fragments can be torn out. Once the fuel fragments are in the water, uranium and neptunium in particular will be dissolved due to oxidation to the hexa- and pentavalent states, respectively.

### 3.3.3 Distribution of radionuclides in the power plant

Of the released fission gases, the noble gases and organic iodine are expected to remain in the gas phase and be ventilated to the atmosphere during the filling of the containment with water (Refer to Section 5.1). Other fission products released are distributed in various proportions between the liquid phase and solid surfaces.

For the sake of simplicity, it is assumed that all released actinides are momentarily transferred to the water phase and either sorbed or eventually treated in the waste treatment plant.

A compilation of the expected distribution of released radionuclides between liquid and solid surfaces is given in Table 3.4 and 3.5 (MAL 79, HUE 85, BAS 86, TOT 86, BEA 87, COL 88).

Of the fission products, cesium will be the totally dominating radiation source when access to the reactor containment is possible, and therefore only the distribution of cesium will be discussed here.

A rough estimate of accessible surfaces in the containment has given that there are around 2000 m<sup>2</sup> concrete surfaces in the drywell and another 800 m<sup>2</sup> in the wet well. Furthermore, there are ca 5000 m<sup>2</sup> steel surfaces on the blow-down pipes (system 314, inner and outer), and 2500-3000 m<sup>2</sup> grating. A total of approximately 20 000 m<sup>2</sup> surfaces exist in the containment, including pipes, cable trays, ventilation ducts etc., but excluding the insulating material.

An estimated 100 m<sup>3</sup> of insulating material are present in the containment. In Oskarshamn 2 the reactor vessel is insulated with "Caposeal" an asbestos containing material, and pipes are insulated with mineral wool. On the reactor vessel there are a total of approximately 70 m<sup>3</sup> insulation and on pipes roughly 30 m<sup>3</sup>.

If the surface sorption corresponds to 1 % of the released radioactivity - which at TMI seems to be valid for submerged surfaces, hence similar to the AB<sub>m</sub> case - the total surface sorption of cesium (as Cs-134 and Cs-137) in the AB<sub>m</sub> case will be 270 TBq.

If the sorption of cesium on mineral wool and Caposeal is similar to that on cement paste, the total sorption of cesium in the mineral wool is ca 7.2 TBq and in the Caposeal ca 27 TBq. With a surface area of 20 000 m<sup>2</sup> the surface contamination of cesium is about 12 GBq/m<sup>2</sup>, with the rough assessment that the sorption is independent of material, and excluding the amount sorbed on the insulating material.

A more thorough discussion on cesium contamination and cleanup of the containment is given in Chapter 8.

In Table 3.5 the amount of released actinides are given both in becquerel and in units of the oral ALI, Annual Limits of Intake (ICR 30), thus giving a reference to the hazardousness of released nuclides. If fuel fragments are released according to the scenario given in section 3.1.2 this gives an equally distributed surface sorption of 90 to 120 Bq/m<sup>2</sup> of U-235 and U-238, and Np-237, respectively, in the containment (not counting the sorption in the insulating material). But an accumulation of both uranium and neptunium in only a few places instead of equally distributed can not be excluded.

With a surface sorption of almost 100 % (cf. Table 3.5) and with an equal distribution throughout the containment, this gives a surface sorption of ca  $1.1 \cdot 10^5$  Bq/m<sup>2</sup> Pu-239 and of  $6 \cdot 10^4$  Bq/m<sup>2</sup> Am-241. But for both plutonium and americium a uniform distribution is very unlikely. Most certainly they are present in "hot spots", i.e. those places where the fuel fragments are deposited.

**Table 3.4. Distribution of fission products in solution and on solid surfaces in the AB<sub>m</sub> case. (Decay data 3 months after the initial event).**

Element	In solution		On surfaces		In bottom sludge	
	(%)	(Bq)	(%)	(Bq)	(%)	(Bq)
Sr	85.8	1.2·10 <sup>15</sup>	0.12	1.6·10 <sup>12</sup>	14.1	1.9·10 <sup>14</sup>
Y a)	5	3.4·10 <sup>13</sup>	95	6.5·10 <sup>14</sup>	b)	
Zr a)	5	8.5·10 <sup>12</sup>	95	1.6·10 <sup>14</sup>	b)	
Nb a)	5	1.4·10 <sup>13</sup>	95	2.6·10 <sup>14</sup>	b)	
Te	6.9	4.7·10 <sup>13</sup>	93.1	6.3·10 <sup>14</sup>	b)	
I-131	96.2	4.6·10 <sup>13</sup>	3.7	1.8·10 <sup>12</sup>	b)	
Cs	98.78	2.67·10 <sup>16</sup>	1	2.7·10 <sup>14</sup>	0.22	5.9·10 <sup>13</sup>

a) Estimated distribution

b) Surfaces + bottom sludges

**Table 3.5. Distribution of actinides in solution and on solid surfaces in the AB<sub>m</sub> case.**

Element	In solution			On surfaces a)		
	(%)	(Bq)	(ALI)	(%)	(Bq)	(ALI)
U-235	13.2	2.1·10 <sup>4</sup>	0.04	86.8	1.4·10 <sup>5</sup>	0.3
U-238	13.2	2.4·10 <sup>5</sup>	0.5	86.8	1.6·10 <sup>6</sup>	3
Np-237 b)	13.2	3.4·10 <sup>5</sup>	17	86.8	2.3·10 <sup>6</sup>	120
Pu-239	0.002	4.5·10 <sup>4</sup>	0.02	99.998	2.3·10 <sup>9</sup>	760
Am-241 c)	0.002	2.5·10 <sup>4</sup>	0.8	99.998	1.3·10 <sup>9</sup>	4·10 <sup>4</sup>

a) Surfaces + bottom sludges

b) The same distribution as for U assumed

c) The same distribution as for Pu assumed

## 4. RADIATION CALCULATION

### 4.1 Introduction

The accessibility to certain parts of the reactor building is of vital importance for the possibility to gain control of the reactor after accidental events such as the AB<sub>m</sub> and TB<sub>m</sub> events. In the same way the selection of the best strategy for the management of the radioactive waste after such events is dependent on the accessibility to the same and other parts of the reactor building and to parts of the waste plant.

Fission products released from the damaged core, mainly the iodine and cesium isotopes, are found outside the reactor containment in high concentrations in systems containing water from the containment and the reactor vessel, i.e. primarily the emergency cooling systems 322 and 323 and the drain systems 345 and 352. The isotopes mentioned could also be found in the shutdown cooling system 321 and the reactor water cleanup system 331 (probably only in the TB<sub>m</sub> case).

Radiation levels much higher than normally occurring are therefore expected in many parts of the reactor building and the waste plant, and consequently, the access of importance could be seriously affected.

Radiation levels have therefore been calculated for a number of positions of interest for the assessment of accessibility in the reactor building and the waste plant of the reference reactor Oskarshamn 2. The calculations have been performed for the AB<sub>m</sub> case using the computer codes CYLGAM, CYLGAX and GAMEN (LUN 75).

The TB<sub>m</sub> case is of minor interest in this study and no source term has been established (cf. Section 3.2). Nor have any calculations been performed especially for this case.

The radiation calculations have also comprised the waste solidification and container handling, and dose estimates for some organic materials used as sealing and cable insulation.

This chapter does not present calculated radiation levels. It has been considered more appropriate to give them in their context in other parts of the report.

### 4.2 Source terms

The gamma sources used in the dose rate calculations are based on the fission product release and distribution given in Chapter 3, i.e. the Tables 3.2 and 3.4. It is assumed that the concentration of radionuclides is uniformly distributed in 10 000 m<sup>3</sup> of water. This is not the case during the first two days following the event when water is being filled up in the reactor containment (Refer to Section 5.1). It is possible to correct for a smaller volume of water, but an uncertainty still exists with regard to the source uniformity during this first time period after the event.

Three months after the event is the longest time for which calculations have been performed. At this moment the two long-lived cesium isotopes Cs-134 and Cs-137 account for 99 % of the total gamma source in the contaminated water.

It has not been necessary to consider the decrease of the concentration of radionuclides in the reactor containment water due to the cleanup procedure, since this is not supposed to start until after three months. Because of the dominance of the two cesium isotopes, it is however easy to correct for this, if desirable later.

### 4.3 Radiation levels and accessibility in the reactor building

#### 4.3.1 General calculations for pipes

Pipes belonging to the emergency cooling systems 322 and 323 constitute the main gamma source after an AB<sub>m</sub> event in the reactor building outside the containment. The dose rates around these pipes are of general interest for determining the accessibility to rooms containing the above mentioned systems. The dose rates for two selected pipes are given in Table 4.1 and present a good picture of the very hard radiation environment.

**Table 4.1. Radiation levels from one DN100 pipe and one DN250 pipe of 5 m length.**

Time after the event	Dose rate (mSv/h)					
	DN100 pipe			DN250 pipe		
	Distance (m) from pipe <sup>a</sup>			Distance (m) from pipe <sup>a</sup>		
	Contact <sup>b</sup>	1.0	3.0	Contact <sup>b</sup>	1.0	3.0
3d	1030	45	9.3	1800	190	43
7d	620	27	5.6	1060	115	26
14d	410	18	3.7	720	77	17
30d	290	13	2.6	510	55	12
90d	240	11	2.1	430	46	10

a) Measured from the centerline and at right angle from the midpoint of pipe.

b) On pipe surface.

#### 4.3.2 Access for measures during the period immediately following an AB<sub>m</sub> event

Following measures are assumed to be necessary after an AB<sub>m</sub> event. For each of the measures the accessibility to the rooms of interest has been judged by calculated or estimated radiation levels.

A. Water and gas sampling (Refer to Figure A3.2, point A)

Initially, sampling should be performed three times per day, but the frequency can be reduced at a later stage. Because of the shielding walls in between and the distance to the systems which have high concentrations of radionuclides after the accident, the radiation levels are very low in the sampling room, the analysis laboratory and along the transportation way between these rooms. Furthermore, the sampling lines in the sampling room are shielded.

B. Manual valve control to prevent leakage water from being pumped to the waste plant. (Refer to Figure A3.1, points A and B)

The systems involved are the controlled area floor drain system (System 345) and the controlled leakage drain system (System 352). This measure is applicable to No. 2 of the three alternatives of accepting leakage water in the waste plant. Alternative No. 2 does not apply to the reference reactor (Refer to Sections 5.5 and 5.6).

The entrance must take place within a short time after the restoration of electric power. This should be possible on the assumption that the fission products released only to a very small extent have reached the piping of Systems 322 and 323, something which seems likely.

C. Inspection and cleaning of sieves in System 345 (Refer to Figure A3.1, point A)

These two measures require access to the area immediately outside the reactor containment where the sump containing the sieves is located. The access way goes either through one of the two pump rooms belonging to Systems 322 and 323 or via a ladder in the room where the sump is located. In both cases the way is close to the piping of Systems 322 and 323. The radiation levels in the pump room, for example, are significantly higher than 100 mSv/h one day after the event, which is the desirable time of the first entrance. Taking into account also the potential for a high radiation level close to the sump, it is found that access is probably not possible. It is not possible, at least, for a longer stay time required for cleaning. After about two weeks the access way is free to enter, but if the sump is accessible also for the two stated purposes is more difficult to decide.

D. Inspection and check of pumps in the emergency cooling systems 322 and 323. (Refer to Figure A3.1, points C and D)

The first entrance is after one week and then once a week. The radiation levels around these pumps one week after the event are 150 to 200 mSv/h, which will give an unacceptable dose of 30 mSv for a 10 min visit. One month after the event the radiation levels have decreased with about 50 %. If, however, the job can be performed from the entrance of the two rooms of interest, the dose rates after one week are 60 and 30 mSv/h, respectively, which could be considered as acceptable.

- E. Inspection and check of pumps in System 352 (Refer to Figure A3.1, point B)

The first entrance is after one week and then once a week. The access way goes through one of the pump rooms of Systems 322 and 323. The radiation levels along the access way and around the pumps are between 10 and 30 mSv/h after one week, which is acceptable.

To sum up, the desirable access is satisfied except for the inspection and cleaning of the sieves in the sump of System 345.

#### 4.3.3 Access during the main cleanup procedure

Dose rates from components in the reactor water cleanup system (System 331) have been calculated, starting with the ion exchangers (Refer to Figure A3.2, point B), to assess if this system can be used for the main cleanup procedure (Refer to Chapter 7) without disturbing the normal accessibility in the reactor building. The running time of the operating ion exchanger has been chosen to give a radionuclide concentration in the resin corresponding to a surface dose rate of 30 mSv/h for the concrete mould used in the solidification procedure (Refer to Section 4.6).

In no case have radiation levels above 10  $\mu$ Sv/h been obtained 3 months after the event when the cleanup is supposed to start. The maximum dose rate on the ion exchanger will be 1.2 Sv/h, which is higher than normally obtained but still possible to manage.

In one corridor (Refer to Figure A3.2, point C) shielded heat exchangers in System 331 account for most of the radiation giving a dose rate of 0.2  $\mu$ Sv/h. Unshielded pipes in Systems 321 and 331 will, however, give high dose rates as indicated by Table 4.1. The conclusion is that the use of System 331 will not affect the normal communication routes in the reactor building, while the entrance to certain rooms will be more difficult than normal. If, however, System 331 will not be used until at the end of the main cleanup procedure or in the final cleanup procedure, the radionuclide concentration in the water has been reduced to such a level that no accessibility problems exist.

In the cases where the source is radionuclides contained in the reactor water, it is possible to transform the result to the  $TB_m$  case. The volume of the reactor water is about 40 times smaller than the water volume in the  $AB_m$  case and hence, the source density is about 40 times greater. For example, the resulting dose rate from the shielded heat exchangers in System 331 will be 6  $\mu$ Sv/h, small enough not to influence the accessibility to the corridor. But unshielded pipes will give rise to adversely high radiation levels. It is of no importance that the same conditions as regards radionuclide concentration are valid for the operating ion exchanger as in the  $AB_m$  case, the use of System 331 for the cleanup procedure is not possible until a substantial decontamination of the radioactive water has been achieved.

#### 4.4 Radiation doses to organic materials

Organic materials exposed to ionizing radiation decompose and could, for example, lose their ability to insulate or seal. A minor study of this problem has been undertaken involving the materials Teflon (PTFE), PVC and Nitrile Rubber, for which the maximum allowable absorbed dose can be set to 300,  $10^5$  and  $2 \cdot 10^5$  Gy, respectively (VOO 72).

PVC is used as cable insulation. The absorbed dose has been estimated for positions close to the electric motors driving the pumps in Systems 322 and 323, and for a time period of up to three months after an  $AB_m$  event. The doses obtained are about 100 Gy, well below the limit given above.

Teflon and Nitrile Rubber are used as sealing in components belonging to Systems 322 and 323. The materials are in close contact with the radioactive water and the estimated doses up to three months after the event are from 200 to  $3 \cdot 10^3$  Gy. Doses of this magnitude pose no problem to Nitrile Rubber, but there is an obvious threat to the durability of Teflon.

This study has, however, not been carried further and no investigation has been made of the risk for break down of any sealings in the systems of interest and the amount of leakage resulting from this (cf. Section 5.5.1). The consequences, for example, could be difficulties to accept the additional leakage in the waste plant, and access restrictions due to a contaminated floor or component.

It is recommended that the influence of radiation doses on the tightness of components and the possible leakage due to a break down of sealings are investigated more in detail in a separate study.

#### **4.5 Radiation levels and accessibility in the waste plant**

The conservatively calculated radiation levels in the control rooms K5.09 (Refer to Figure A3.6, point A) and K6.91 (Refer to Figure A3.7, point A) are low, well below 10  $\mu\text{Sv/h}$ , during the first week following the event. This is also true later on, when after three months the main cleanup procedure starts, and the ion exchanger vessel and the spent resin tank contain zeolites instead of organic resin.

One essential contribution to the radiation level in the main control room K5.09 is obtained from the proposed new pipe installation in the adjacent tank room K4.20 (Refer to Chapter 10 and to Figure A3.5, point A), containing the highly radioactive water from the reactor containment. It is suggested that the installation work is performed during the first week after the event. The radiation level at the working place is then lower than 10  $\mu\text{Sv/h}$ , coming from one of the receiving tanks for leakage water.

With one exception no access problems have been encountered in the waste plant. Outside the ion exchanger room there are valves identified (Refer to Figure A3.5, point B) which have to be manually controlled in connection with the cleanup process. Here the dose rate is about 15  $\text{mSv/h}$ . Furthermore, access to various rooms for component service and in the event of repairs has not been considered. In such cases, the situation should be managed by draining the component and the system parts of interest and by taking the usual radiation protection measures such as decontamination and shielding.

#### **4.6 Waste container**

##### **4.6.1 Radioactive load**

In the reference plant used in this study, the low and medium level radioactive waste generated during normal operation is solidified with cement into prefabricated cubical concrete moulds with an external dimension of 1.2 m. Depending on the specific activity of the waste to be solidified, the wall thickness of the moulds may be chosen to be 0.10 m or 0.25 m, and thus the internal volume will be either 1.0  $\text{m}^3$  or 0.34  $\text{m}^3$ . The

waste load into the moulds may vary somewhat, but as a rough estimate it will be  $0.50 \text{ m}^3$  and  $0.17 \text{ m}^3$  in the two different types of moulds.

The Swedish authorities, the National Institute of Radiation Protection (SSI) and the Swedish Nuclear Power Inspectorate (SKI), have stated a dose rate limit of  $30 \text{ mSv/h}$  on the moulds for organic resins solidified with cement into concrete moulds. This is based on what is practical to handle in the existing solidification plants and on the risk of decomposing the resin.

The absorbed doses to organic resins should be limited to less than  $1 \text{ MGy}$  (PIL 82), while inorganic ion exchangers, for example zeolites which are recommended in this study (Refer to Chapter 6), are much more stable to ionizing radiation. Since zeolites have not been used in Swedish nuclear power plants, there is no surface dose rate limit stated by SSI for concrete moulds in this case. After discussions with the staff at the reference waste plant in this study and with SSI and SKI it has been decided to use a contact dose rate of  $300 \text{ mSv/h}$  for the moulds in the event that zeolites are solidified with cement.

For a unit content of Cs-134 and Cs-137 the surface dose rate has been calculated and is given in Table 4.2.

**Table 4.2. Specific surface dose rate (mSv/h per TBq).**

Nuclide	Mould wall 0.25 m		Mould wall 0.10 m
	Zeolite	Organic resin	Organic resin
Cs-134	12.0	8.2	36
Cs-137	3.9	2.6	12.5

During the major part of the cleanup the concentrations of Cs-134 and Cs-137 in the water are approximately equal. For equal amounts of the two cesium isotopes it is found from Tabel 4.2 that the possible content of cesium solidified in a concrete mould with  $0.25 \text{ m}$  thick walls is 35 to 40 TBq for zeolite ( $300 \text{ mSv/h}$  on the surface), while the corresponding value for organic resin is about 5.5 TBq ( $30 \text{ mSv/h}$  on the surface). Using the thin-walled mould for organic resins, an option suitable for the final cleanup, means a radioactive load of only about 1.2 TBq cesium.

The absorbed dose at the mould centre has been checked for the above mentioned radioactive loads. In the zeolite case it will be about  $9 \text{ MGy}$ , while the organic resin will receive about  $1 \text{ MGy}$  and less than  $0.1 \text{ MGy}$  for the larger and smaller wall thickness, respectively.

At other Nordic nuclear power plants a standard steel cask (volume 200 litres) is used for solidification with bitumen. The allowed contact dose rate is 500 mSv/h and the possible cesium load is less than 1 TBq. Consequently, the standard cask is not a better alternative than the cement solidification of zeolites into the thick-walled concrete moulds.

#### **4.6.2 Handling and transport**

The handling of the waste container during and after the solidification process has been analysed with respect to the radiation exposure of the personnel.

The analysis of the container handling in the waste plant and the on-site storage has not revealed any significant radiological problems, but a few parts of these operations have been identified as possible to improve. This will be further discussed in Chapter 10.

The on-site storage facility is located underground and some hundred metres from the waste building. The transportation to the storage is carried out by means of a specially designed transport vehicle and a trailer, which can take three concrete moulds and is provided with a radiation shield. With a dose rate on the moulds of 300 mSv/h the radiation level at a distance of 2 m from this shield will be 0.25 mSv/h, and so does not meet the IAEA transport requirement of 0.1 mSv/h.

It should, however, be possible to manage this situation by closing the route, since the transport is entirely on the site ground. Otherwise, an additional radiation shield is required. This could also be necessary for the driver's cab where the dose rate is 50  $\mu$ Sv/h.

The on-site storage facility is used as an intermediate storage before the transportation to the assumed final disposal in the SFR facility (Final Repository for Reactor Waste) at the Forsmark site. This operation will be carried out by means of large, specially designed containers equipped with thick shielding walls. These containers are intended for the transportation of waste containers of the same type as those discussed in the preceding Subsection and filled with normal medium and low level reactor waste. It has been shown that two of these special containers are well fitted for the concrete moulds produced in this study. The IAEA transport regulations are met with respect both to the radiation level outside the container and to the radioactive content of the solidified waste.

## **5. MEASURES TO BE TAKEN IN THE REACTOR BUILDING AND THE WASTE PLANT DURING THE PERIOD IMMEDIATELY FOLLOWING AN AB<sub>m</sub> EVENT**

### **5.1 Sequence of events in the reactor containment**

#### **5.1.1 Spraying**

When the electric power is reconnected after approximately 35 minutes, the core spray system (System 323) comes into operation and the core will be sufficiently cooled after approximately 10 minutes. The assumption is made that the core has been damaged to the extent that approximately 10 % of the core content of noble gases has been released. These gases and other fission products are transported to the pipe break location in the main recirculation system (System 313) by the core spray water and to the condensation pool by the blow down pipework. Water for System 323 is drawn from the condensation pool.

The high pressure and temperature in the containment will cause the containment vessel spray system (System 322) to start automatically as soon as electric power is restored. The water in the condensation pool is cooled by this system and returned to the reactor containment, where it is sprayed into the drywell. Spraying cleans the atmosphere in the containment from the condensable fission products and the majority of the released fission products settle in the condensation pool.

#### **5.1.2 Filling with water**

The reactor containment must be filled with water up to a level that is higher than the top edge of the core. In the long term this will, in combination with some pressure relief capacity, facilitate the cooling of the core and further, the removal of the core. There is a total of 2 200 m<sup>3</sup> of water in the condensation pool in the containment and the reactor vessel, including the main recirculation system. An additional quantity of 7 600 m<sup>3</sup> of water is required to fill the containment to a level that is above the top edge of the core. The water is pumped into the containment through the spray nozzles by diesel driven pumps having a capacity of 75-100 kg/s and belonging to the fire water system (System 861). The water is drawn from this system, which has a normal capacity of 4 000 m<sup>3</sup>. Additional supplies of water are accordingly necessary to fill the containment.

OKG suggests that the water is taken from Götmarén, a nearby lake which is used as a water magazine. This raw water needs to be filtered in the sand filter of the local water works so as to reduce the content of humus materials. The water can be pumped to the containment at a rate of 150 m<sup>3</sup>/h except during the time required for filter cleaning after every 1 000 m<sup>3</sup>.

When approximately 6 000 m<sup>3</sup> of water have been pumped into the containment the pressure will have risen to 0.4 MPa. The containment will have to be ventilated to the atmosphere to enable the additional quantity of 1 600 m<sup>3</sup> to be pumped in. A pressure relieving system for the containment was installed during 1988 and the containment atmosphere is pressure relieved by means of a Multi Venturi Scrubber System (MVSS), which retains practically all radioactive material with the exception of the noble gases and the organic iodine.

The manual pressure relieving of the containment atmosphere is preceded by sampling of the containment atmosphere and consultation with the authorities and the meteorologists, to ensure that the release to atmosphere is done under the most favourable weather conditions. The necessity of this procedure is confirmed by an estimate of the doses to the environment (Refer to Appendix 4). The effective dose equivalent to the most exposed individual members of the public has been calculated for releases to the atmosphere starting one day and seven days after an AB<sub>m</sub> event. In the worst case the dose will be 10 and 2 mSv, respectively. For more common weather conditions, however, the doses will be a factor 300 lower.

The assumption seems to be reasonable that the pumping in of water should start approximately 8 hours after the occurrence of an AB<sub>m</sub> event. This time is required to enable an analysis of the situation, arranging for assistance and planning the required activities. The time required for filling the containment will be from 40 to 45 hours based on the pumping capacities given above. Waiting time and the time required for pressure relieving must also be considered. These additional time periods may be eliminated if pressure relieving is possible during the water filling.

## 5.2 Sampling

In order to take correct actions after an AB<sub>m</sub> event, the conditions of the reactor must first be analysed. For this purpose samples of the water in the condensation pool, the water in the reactor vessel and the atmosphere in the containment, must be taken.

The system for sampling is to provide the possibility for determining:

- the quantity of the fission products in the water and the atmosphere in the containment
- the chemistry of the water and atmosphere in the containment

The system for sampling is to be designed and installed in such a manner that the personnel involved are not subjected to higher than acceptable doses of radiation, both during the taking of samples and when passing to and from the sampling area.

Refer to Lahti et al (LAH 80) for the guiding principles regarding the design of an accident sampling system. The system installed in Oskarshamn 2 is designed in accordance with the principles given in that reference.

### 5.2.1 Water sampling

A sample of the water in the containment must be taken as soon as conditions have become stabilized following an accident. This will give an indication of the extent of the damage to the core. Samples should be taken as frequently as three times per day so as to follow the development of the radioactivity. Obviously the frequency of sampling can be reduced at a later stage.

The accident sampling system (System 337) provides three points from which samples of water can be taken from the condensation pool. These are located on the discharge side of two pumps in System 322, and on the discharge side of one pump in System 323. The existing nozzle for an additional power range monitor (PRM) probe on the lower reactor vessel head can also be used for taking a sample of the water in the reactor vessel. In this case a pump is not necessary as gravity will provide the flow for the purpose of sampling.

The water sample is fed through a cooler and into a fume hood located in the Turbine Building (Refer to Figure A3.2, point A). The sample is emptied into the lead shielded sample container, which is then transported to the Analysis Laboratory in the adjacent Electrical Building. The accessibility for sampling is excellent (Refer to Section 4.3.2).

### 5.2.2 Gas sampling

A sample for measuring the level of radioactivity in the atmosphere of the reactor containment must be taken before the atmosphere is ventilated when filling the containment with water.

The sample is obtained by means of special equipment which is a part of the containment gas treatment system (System 741).

The system draws gas from the containment. The sample is fed through a cooler and a moisture separator and is further dried in a refrigerated drier. The gas is then either fed into a vial placed in a fume hood located in the same room as that for the water sampling or returned to the containment through an in-line meter which measures the hydrogen and oxygen contents. When the vial is full, it is disconnected and taken to the Analysis Laboratory for the contents to be analyzed.

The atmosphere of the containment is passed through a MVSS filter, where the moisture borne radioactive material is washed from the gas, before entering the exhaust stack. As a consequence of the moisture separation the sampling procedure can be considered as providing a sample which is comparable to the gas exhausted to the stack during the ventilation of the containment atmosphere.

### 5.3 Disconnection of the reactor water cleanup system

The high temperature and pressure in the reactor containment resulting from an  $AB_m$  event, will result in the closing of the isolation valves in the shutdown cooling system (System 321), at the same time as the pumps in the system are stopped. A quarter of the water flow in the system is fed into the reactor water cleanup system (System 331) during normal operation of Oskarshamn 2.

Should System 321 be restarted after such a series of events and the ion exchangers in System 331 have not been disconnected, then the organic ion exchange resin will begin to absorb an amount of radioactive material far in excess of that intended. The ion exchange resin in a similar clean-up system in TMI-2 was operated for too long a period under similar conditions, with the result that the resin was baked solid and could not be removed from the containers. There is uncertainty as to whether this occurred as a result of radiation, high temperature or both.

The extent of damage to the core in an  $AB_m$  case, or as occurred in TMI-2, does not need to be particularly serious for a similar condition to result. Fuel damage equivalent to the release of approximately 0.1 % of the core content of iodine and cesium will provide this condition as has been shown by Elkert and Ullberg (ELK 85). In order to avoid overloading the ion exchanger, radiation measuring instruments should be installed, see Section 10.2.2.

The retention of the function of System 331 can provide technical benefits to the process as the system will be of assistance in the final stages of the cleanup process (cf. Chapter 7). The system will help to keep the reactor water free from contamination when the reactor vessel has been opened.

One of the assumptions for using both systems 321 and 331 in an  $AB_m$  case is that the inner isolation valves in System 321 are opened before the water level in the reactor containment has reached the level of the valves. In order to reduce the total leakage this procedure should not be carried out, if there was any appreciable leakage from these two systems before the accident occurred. On the other hand it is doubtful whether the isolation valves will stay open, as the valve actuators are not designed for operating submerged in water.

Both the core and the suction inlets of System 321 must be beneath the water level before the pumps in System 321 are started. There will be a delay of at least 48 hours before the required water level is attained which means that there is time for the events to be analyzed and the decision made to disconnect System 331 before restarting System 321. A good reason for restarting System 321 is that the cooling of the core is a high priority and the heat exchangers in the system are an alternative to those in System 322.

### 5.4 Management of airborne radioactivity in the reactor building

System 741 is open to the containment following the events of an  $AB_m$  case and because of the risk of leakage, provides a potential for the spreading of the airborne radioactivity from within the containment. The system is restarted manually when the electric power is reconnected so as

to recombine the gases in the containment and for the purpose of obtaining samples for analysis. A final sample is taken towards the end of the process of filling the containment with water, before stopping the fans and manually closing the isolation valves.

The input and output fans to the ordinary ventilation system in the reactor building, which is a part of the ventilation system for restricted areas (System 742), are stopped following an AB<sub>m</sub> event. There is a change in the connections in System 742 when electric power is restored, so that the air extracted from the reactor building is fed through the emergency filters in the offgas system (System 341), and out to the atmosphere.

Airborne radioactivity from the reactor containment will be found in different areas of the reactor building, especially in areas adjacent to the containment. This form of radioactivity is referred to as diffused leakage and can be the result of leakage in System 741 and in the penetrations in the containment wall.

There is a lesser possibility that leakage comes from the closed isolation valves and the associated systems. Leakage of airborne radioactivity can also come from systems which are open to the reactor containment. In both of these cases the leakage is associated with the leakage of water (cf. below).

The airborne radioactivity may require that radiation protection measures become necessary. In the meantime System 341 will restart when electric power is restored after approximately 35 minutes, and begin to exhaust any radioactivity that has collected. The possibility remains, however, that extraction from certain areas may be ineffective or insufficient, with the result that the concentration of radioactivity in the air is maintained and radiation protection becomes necessary.

The leakage from System 741 and the penetrations in the containment wall will not build up to a large quantity during the period that the ventilation system is out of operation. This should mean that radiation protection problems in the areas concerned are not likely.

The concentration of radioactivity in the air in connection with water leakage is very difficult to establish. This type of leakage will not really be apparent for some length of time.

Suitable radiation protection measures under these circumstances are regular measurement of the air concentration of radioactivity and personnel protection such as face masks or complete body protection overalls.

## 5.5 Management of leakage water in the reactor building

### 5.5.1 Spreading of radioactivity in the reactor building

The spreading of radioactivity from the reactor containment and the reactor vessel into the reactor building, outside of the containment, is the result mainly of leakage from System 322, 323 and possibly the sampling system (System 821) and leakage through closed isolation valves belonging to the following systems:

Steam lines	(System 311)
Feed water system	(System 312)
Shutdown cooling system	(System 321)
Auxiliary feed water system	(System 327)

All the systems are connected to the reactor vessel and have parts located outside the containment. It is supposed here that system 321 has not been started.

In the System 821 it is sampling connections in System 322 that could leak. Should an  $AB_m$  event occur during sampling, the hand operated valve in the sampling line can not be closed for a period of one to two weeks following the event as a result of the high radiation level (cf. Chapter 4). During this period the water would run from the connection to the controlled area floor drain system (System 345).

The isolation valves in the controlled leakage drain system (System 352), will close following an  $AB_m$  event, after which drainage from the reactor containment, such as the intermediate valve stem drainage from the inner isolation valves, is directed to the condensation pool by means of one of the safety valves in System 352. Intermediate valve stem drainage from the external isolation valves is directed to a tank (T1) in System 352.

Any leakage from System 352 is handled in a similar manner to that from Systems 322 and 323. The leakage is collected in the floor drainage gutters in the respective rooms and led to the sump (T40) in System 345 (Refer to Figure A3.1, point A).

The total leakage, which has to be handled following a LOCA, has been estimated as 25 m<sup>3</sup>/day for one Swedish BWR. Approximately 1 m<sup>3</sup>/day of this amount is considered as leakage from the reactor containment (see ELK 85). The leakage in an  $AB_m$  case in Oskarshamn 2 is considered in the greater part to be the same as for this case.

The estimated amount of leakage from the reactor containment does not contain any margin for the leakage that would result in the event of the breakdown of a mechanical shaft seal on any of the pumps in System 322 or 323 (cf. Section 4.4). Excessive leakage from a pump will only be observed during a tour of inspection of the pump areas, but this cannot occur earlier than 1-2 weeks following an accident because of the high radiation level. Leakage from a pump can be stopped by shutting down and isolating the pump providing that the plant can accept a reduction in the cooling capacity.

The pump can probably be repaired after approximately 1 month providing that part of the system can be drained and if necessary, provided with a radiation shield. Clean water should be connected to flush out the pump prior to dismantling.

In the event that sampling of the water in System 322 is being undertaken when the accident occurs, then System 345 will receive a quantity of 2 m<sup>3</sup>/day for a period of 1 to 2 weeks. This quantity must be added to the leakage from the reactor containment.

Consequently the leakage from the reactor containment may exceed 1 m<sup>3</sup>/day. Normally it seems reasonable to assume that the leakage, which will originally be in the region of 1 m<sup>3</sup>/day, will reduce with time, but with the reservation that additional leakage will not result from a damaged mechanical seal on a pump, or from any other unexpected source.

The duration of the leakage of a quantity of 24 m<sup>3</sup>/day from systems which are isolated from the containment is unknown. The leakage should stop relatively quickly as the volume of the systems is limited, always providing that they are not being continually fed with water which will replace the leakage. The dilution of the highly radioactive leakage from the reactor containment, when mixed with other leakage water in 345 T40 and 352 T1, will consequently reduce with time.

#### **5.5.2 Discharge of leakage water from the reactor building**

When the electric power is restored and with the existing layout of Systems 345 and 352, the leakage water will automatically be pumped out to the waste plant when the water levels are high in 345 T40 and 352 T1. There are three possible alternatives concerning the ability of the waste plant to accept this quantity of highly radioactive leakage water:

##### **Alternative 1.**

The waste plant can accept the leakage water immediately. As soon as 345 T40 and 352 T1 are filled up, the leakage water is discharged to the waste plant. The existing design of Systems 345 and 352 is not affected.

**Alternative 2.**

The waste plant can accept a limited amount of the leakage water, an amount equal to that discharged by Systems 345 and 352 during a 30 minute period after restoration of the electric power. During this time manual control can be taken to prevent any further pumping of leakage water (Refer to Section 4.3.2). Systems 345 and 352 will need to be modified so that the leakage water can be stored in the reactor building until the systems can discharge a further quantity to the waste plant.

**Alternative 3.**

The waste plant has to be modified in order to accept the leakage water. During this period the water must be stored in the reactor building, which means that Systems 345 and 352 need to be modified. Manual control is not possible under these circumstances and automatic valve control, etc. is required to retain the leakage water in the reactor building.

Alternative 1 is valid if the waste plant has the capacity for receiving all the leakage water that is produced during the period when the plant is being modified and prepared for the main cleanup procedure of the water in the reactor containment. Here, receiving capacity includes the possibility to pump back leakage water to the containment, either immediately or after a supplementary pipe installation (Refer to Par. 5.6.2). Otherwise Alternative 2 or 3 will apply, in which case comparatively large modifications will be necessary to Systems 345 and 352. This is dealt with in Chapter 10.

In an AB<sub>m</sub> case a number of valves in System 322 are activated to the closed position. These valves are normally activated to open at high water level in the condensation pool so allowing water from the containment to be discharged to the waste plant. The closure of these valves must not be overridden when the reactor containment is being filled with water. The valves must, however, be able to open at a later stage when the containment is full of water as the discharge to the waste plant during the main cleanup procedure is controlled by these valves.

**5.6 Management of leakage water in the waste plant****5.6.1 Introduction**

Under normal conditions water from 345 T40 and 352 T1 is collected in separate receiving tanks in the liquid waste system (System 342). After a decontamination step, normally performed by means of filters, the floor drain (System 345) is released to the recipient. After a cleanup step by means of filters and organic ion exchangers the leakage drain (System 352) is re-used within the power plant.

These cleanup steps can be initiated automatically when the water level in the receiving tanks has reached a certain level, but normally there is always a circulation of water between these tanks and the cleanup filters. After an AB<sub>m</sub> event it is very important that highly radioactive water will not be circulated in large parts of System 342 until any required modifications of the installation have been performed.

The treatment of the leakage water in the waste plant immediately following an accident, can be done by one or more of the following procedures.

1. Pumping the water back to the reactor containment.
2. Storage in the waste plant.
3. Cleaning by means of organic ion exchangers.
4. Cleaning by means of zeolites.

The containment must be fed with the same amount of water as that which leaks out to maintain the water level in the reactor containment and the reactor vessel. This is not an urgent need as the containment can be filled with water to a level above the top of the core so as to provide a margin for leakage. Leakage from other parts of the plant, e.g. the turbine section, is mixed with the leakage from the containment before being discharged to the waste plant. This is the reason why a part of the leakage water is retained in the waste plant as all of the water cannot be returned to the containment.

#### **5.6.2 Return pumping and storage**

The reactor containment provides a very suitable storage facility for the highly radioactive leakage water. Despite being diluted by mixing with the water, which was originally found in the receiving tank and the leakage water from sources other than the containment, the radioactivity concentration of the water is much higher than normal. Therefore, the most convenient method for handling the leakage water is to pump as much as possible back into the containment, meanwhile, trying to minimize the leakage from sources other than the containment. Thus, reducing the loading on the receiving tanks.

The return pumping is dependent on the existence of the necessary conduits, or that suitable conduits can be made available. The latter is possible in Oskarshamn 2, as new pipe connections with shut-off valves can be installed in the waste plant. Such an installation, which is discussed in Chapter 10, could be completed within a week following the accident.

Should there not be the possibility to pump the water back to the containment then the total quantity of leakage water must be handled in the waste plant. Furthermore, the water level in the reactor containment must be maintained in another way, possibly by a supply of water from System 861.

The receiving tanks, designed for Oskarshamn 2, in the reference waste plant are large, (total 250 m<sup>3</sup>) and can accept the anticipated flow of leakage water for a period of one or two weeks. This period will depend on the quantity of water in the tanks when the accident occurs and the rate at which the leakage reduces during that time. The leakage from the containment is small in the beginning and the leakage from other sources could probably be minimized within a week, which should mean an even longer period during which the tanks can be fed with leakage water.

The storage of the water is beneficial as the radioactivity will be reduced with time and any required modifications to System 342 can be completed. There is also time in which to discuss further handling of the water and to make the necessary preparations.

### 5.6.3 Cleaning

During normal operation the leakage water is cleaned by the use of organic ion exchange resins, which absorb the radioactive ions and also the dissolved salts. The operating time is restricted from the radioactive point of view by either the recommended maximum radiation dose 1 MGy to the compound or the allowed surface dose rate 30 mSv/h on the concrete moulds used for the cement solidifying of the resins in Oskarshamn 1 and 2 (Refer to Section 4.6).

An alternative to organic ion exchanger resins is zeolite, refer to Chapter 6, which will absorb a larger quantity of radioactivity in the form of cesium and strontium, but on the other hand will not absorb other radionuclides and dissolved salts. Furthermore, in this case higher surface dose rates are permissible on the concrete moulds used for solidification. As zeolite is not at the present used in Sweden, there is the need for preparatory work including testing and the obtaining of the necessary material. This is the reason why it is imperative to use an organic ion exchanger compound for cleaning purposes during the first period following an accident.

In the event that the waste plant serves more than one reactor, as in the case of Oskarshamn 2, then a part of the capacity of the waste plant must be reserved for the other reactor. Consideration must also be given to the capacity of the solidification plant and the facilities for the storage of the spent radioactive resins. The capacity of the solidification plant is currently 0.8 mould per shift.

If necessary, the cleaning of leakage water should be carried out with bead resins in one of the two deep bed ion exchangers in system 342. During normal cleaning of the leakage drain water the ion exchanger only needs to be back flushed at intervals of a few months. The tank T13, which is used for the storage of spent resins, has a capacity of 19 m<sup>3</sup> but is normally only half full. There is therefore every opportunity for the storage of a number of charges of ion exchange resins during the cleaning of the leakage water, without the need for immediately commencing the solidification process.

Once more, it should be observed that cleaning of leakage water should be avoided during the first time following an AB<sub>m</sub> event in order not to contaminate System 342 before any necessary modifications of the installation have been undertaken.

Based on the previously stated amount of leakage and the concentration of radionuclides in the water in the reactor containment, the input flow of radioactivity to the waste plant during the first week after the accident can be estimated as being between 150 and 600 TBq with a composition as Table 3.2, of which Cs-134 and Cs-137 represent 20 and 80 TBq, respectively. A quantity of from 1 to 3 charges, each 1.5 m<sup>3</sup> in capacity, of ion exchange resins is required for the cleaning of water having such a level of radioactivity and assuming that solidification can be left for some time. Between 10 and 30 concrete moulds with a wall thickness of 0.25 m will be required to solidify that amount of radioactive waste. In the event that the solidification process must be commenced earlier and the only available concrete moulds have a wall thickness of 0.10 m then the number of moulds required will be a magnitude greater.

A flow of 2.5 kg/s is required to ensure that the ion exchangers operate satisfactorily and provide the required cleaning efficiency of 99 %. This means that the operating period will be in the order of 5 to 10 hours so as to achieve the optimum concentration of radioactivity in the resin. The clean water is discharged to the outlet tank but can also be stored in Tank T43 outside the waste plant. Allowing for the stated grade of cleanliness and the expected concentration of radioactivity in the water before cleaning, the dose rates outside the tank three months following the accident will be 50 and 1 µSv/h at distances of 10 m and 100 m, respectively.

There are no problems concerning access to the control rooms in the waste plant during this first period following an AB<sub>m</sub> event (Refer to Section 4.5).

#### **5.6.4 Conclusion**

To sum up, the observation can be made that leakage from the reactor containment and other areas of the power plant, following an AB<sub>m</sub> event, can be managed in a reassuring manner without any major problem being evident. These conclusions are in the first place valid for the reference reactor Oskarshamn 2. Certain additional investigations must be carried out before similar conclusions can be made for other BWR's in the Nordic Area.

## 6. CHOICE OF SORBENTS AND WASTE FORMS FOR THE TREATMENT AND SOLIDIFICATION OF RELEASED RADIONUCLIDES

Two major factors have to be considered when deciding which way to clean up radioactively contaminated waste waters. First of all, a load as high as can be handled should be used in order to reduce the total number of waste containers produced. Secondly, the material used for treatment and solidification of the waste must be sufficiently radiation and temperature resistant to stay intact for a prolonged period of time.

### 6.1 Sorbents suitable for the cleanup

Normally organic bead or powdered ion exchange resins are used as sorption material for the purification of process water in a nuclear power plant. Compared to the use of inorganic ion exchangers, the use of organic resins for the cleanup of waste waters after an  $AB_m$  case accident would simplify the handling and solidification of spent resins in the waste plant as the procedure used in the normal operation of the solidification system can be used. Precautions must be taken though, due to the higher external radiation from pipes and tanks, etc., and especially to the rather limited radiation resistance of organic resins. In the  $TB_m$  case, where the cesium concentration in the water is much higher, organic resins are probably less suitable due to the high risk of over-loading the resin.

Most organic ion exchangers available for commercial use are synthetic resins made of hydrocarbon chains (typically linear polystyrene) cross-linked with divinylbenzene (DVB). The degree of cross-linking determines the lattice width, swelling characteristics, ease of ion movement, hardness and resistance to mechanical breakdown. Due to the ionic groups fixed on the resin framework, the ion exchange capability is imparted to the material.

When using organic resins the dose rate limit stated by Swedish authorities on the final solidified waste product is 30 mSv/h. With this surface dose rate the concrete mould with 0.25 m thick walls may be loaded with about 5.5 TBq cesium, assuming equal amounts of Cs-134 and Cs-137 (Reference to Section 4.6). This gives an absorbed dose of about 1 MGy. But during accident circumstances the limit of 5.5 TBq cesium per waste container is too low as this would lead to a too large number of waste packages, therefore inorganic adsorbers are recommended.

Inorganic ion exchangers are much more stable to ionizing radiation than are organic ion exchangers (about 1000 times more resistant), and would be preferred for the purification of highly radioactive waste waters. Of inorganic ion exchangers, zeolites have a profound ability to adsorb cesium and are widely used for cleanup purposes. Furthermore, zeolites have a good stability against radiation. No chemical or physical effects have been observed in zeolites after a gamma irradiation giving a dose of less than 1 GGy.

With a maximum surface dose rate of 300 mSv/h, a cement mould with 0.25 m thick walls can be filled with 35 to 40 TBq Cs-134/Cs-137 (50/50) using zeolite. This will give a total accumulated dose to the waste of about 9 MGy.

Zeolites are crystalline, hydrated aluminosilicates containing alkali and/or alkaline earths, in particular: sodium, potassium, magnesium, calcium, strontium and barium. They can be prepared as high purity crystalline powders or as preformed pellets from reactive aluminosilicate gels or hydrogels. Because of their three-dimensional framework structure, most zeolites do not undergo any appreciable dimensional change such as swelling or shrinking (BRE 74).

Organic resins are temperature sensitive. The temperature should be kept below 60°C. Elevated temperature leads to loss of active groups with a decrease in capacity as a consequence. No particular structural damage occurs to zeolites at temperatures up to about 700°C. At about 900°C and above, the crystalline structure collapses causing an apparent shrinkage of about 50 percent.

Not only high radiation and possibly a high temperature, but also low selectivity for cesium make organic resins less suitable for the cleanup.

Since normal tap water and sand filtered raw water will be used for filling-up the containment vessel, the content of ionic species not usually present in the containment water, e.g. sodium and calcium, will compete for the active sites on the ion exchange resins. Furthermore, as the accident-generated water will remain in the containment vessel for a long period of time, the minerals dissolved by leaching of the concrete material will further contribute to the mineralization of the water and, thus, to the loading of the resins.

The tap water calcium content at the Oskarshamn site is as an average 25 g/m<sup>3</sup>. The concentration of Cs-137 in the containment water will be approximately 1.3 TBq/m<sup>3</sup> three months after the accident. An activity of 1 Bq Cs-137 corresponds to  $3 \cdot 10^{-13}$  g of cesium, i.e. the content of Cs-137 in the containment water will be about 0.4 g/m<sup>3</sup>. Thus, organic ion exchangers will be loaded with non-radioactive ions long before any significant amounts of cesium have been taken up. Of zeolites, chabazite has a superior selectivity for cesium and should be selected for the cleanup, hence, selectively removing cesium from the highly contaminated water.

A chabazite zeolite, in the sodium form (Linde Ionsiv IE-96), was used with good results in the Submerged Demineralizer System (SDS) used at Three Mile Island (TMI) (TOT 86). In fact, the zeolite used could probably take a 100 times higher load of Cs-134/Cs-137 than can be accepted in the present study considering acceptable surface dose rates on the final solidified waste cubicals.

But IE-96 is not particularly effective in adsorption of strontium; therefore, for the removal of strontium e.g. the Linde A-51 zeolite could be used. The Linde A-51 was also used at TMI, and the best decontamination result was achieved when the zeolites were kept in mixed beds with the IE-96/A-51 zeolite ratio at approximately 1/1. If the strontium content in the waste water is low, a ratio closer to 2/1 might be feasible.

Zeolites have not been used in the liquid waste system of Swedish nuclear power plants. Thus, it is important to confirm that this is possible by performing non-radioactive tests of the system before the actual cleanup starts. Even if no obvious problems are envisaged; for example, an increased tear of pump seals caused by the zeolites could increase the overall leakage of the plant.

Furthermore, with a high radioactive load of the zeolites, a radiolysis of the water cannot be excluded. At the very high concentration of cesium adsorbed on the Submerged Demineralizer System cartridges used at TMI-2, water was split into hydrogen and oxygen by radiolysis. The gases were recombined into water using a catalyst mixed with the zeolite. An addition of a catalyst to the zeolite bed would be quite possible in both the AB<sub>m</sub> and the TB<sub>m</sub> case, but is due to the much lower cesium concentration as compared to the TMI-2 case probably not necessary. The proper action can be taken after a check whether or not there is a gas production in the zeolite bed.

The zeolite columns will significantly reduce the over-all radioactivity of the waste water, but before the water can be released to a recipient, a final polishing is needed, where residual cationic and anionic radioactive species are removed. At this time, after the waste water has been processed through the zeolite columns, the cationic and anionic organic resins used in the normal operation of the power plant are adequate (cf. Chapter 7).

## 6.2 The solidification matrix

Both bitumen and cement are commonly used as solidification matrices for spent ion exchange resins. Of these two, cement is preferred due to the following reasons.

The waste generated from the cleaning of the contaminated water could be stored in the storage facility for reactor waste, SFR, at the Forsmark power plant. The silo in SFR used for medium radioactive waste is cast in sulphate resistant concrete. Thus, considering the preferred final storage of the waste, cement is the most appropriate solidification matrix for the used ion exchangers.

The addition of zeolites to the cement is not expected to significantly degrade the properties of the cement matrix. Experiments have shown that mixtures of cement and zeolite (a zeolite type showing high adsorption of cesium) have high compressive strength, similar to the compressive strength of non-mixed cement formulas and that incorporating the Cs-zeolite into concrete has no adverse effects on mechanical or chemical properties of the waste forms. Also, the addition of zeolite loaded with Cs-137 to the cement markedly reduced Cs-137 leachabilities as compared to the leachability of Cs-137 directly incorporated in the cement/sludge mixture (STO 79).

Cement and mortar have been observed to change the composition of the zeolites clinoptilolite and mordenite at the interface to form Al-substituted tobermonite. But the formation of Al-substituted tobermonite led to good bonding at the interface between the aluminosilicate ion exchanger and cement or mortar, and, furthermore, the Al-substituted tobermonite had good cesium sorptive properties (KOM 83).

Thus, it is expected that an incorporation of the IE-96 Chabazite Zeolite in the cement matrix will give a waste form with high mechanical and chemical stability. Although a transformation of the chabazite to some other aluminosilicate form at the interface between the cement matrix and the zeolite is quite probable, a low cesium leachability is expected.

Cement has a good resistance to ionizing radiation and is accordingly used as shielding and construction material in e.g. nuclear reactors and gamma irradiation facilities for sterilization.

Stone (STO 79) reports of experiments where the long-term radiation stability of concrete was studied by gamma irradiating specimens to 100 MGy, which simulated a 100-year integrated dose of their actual waste. No adverse effects on compressive strength or Sr-leachability could be ascribed to radiation. It has also been reported in the literature that radiation damage on concrete due to gamma radiation is not observed until an accumulated dose of 0.1-1 GGy. Considering the doses that concrete constructions receive in nuclear facilities, the cement moulds used for storage of ion exchangers will probably stay intact at doses higher than 1 GGy.

The most serious limitation of cement as a host material is that it dissolves in an acid environment (i.e. below ca pH 5). But the presence of an acid environment is very unlikely if, for example, the cement moulds will be stored in the SFR concrete silo, where the pore water in the concrete/-cement will be buffered at a pH above 11 for thousands of years.

The cement quality used for the immobilization should be a sulphate resistance type, i.e. a cement quality with a low content of tricalcium - aluminat ( $C_3A$ ). The reason hereto is to prevent - or at least to minimize - the formation of ettringite, which will be the result of the reaction between the sulphate ions in the waste and the  $C_3A$  of the cement.

## **7. WATER TREATMENT AND IMMOBILIZATION OF THE GENERATED WASTE**

### **7.1 Technical description of the water cleanup method**

As the intention is to take the nuclear power plant into operation again, care must be taken to minimize any further damage to the reactor. In order to keep the costs for treatment, storage and transportation at a reasonable level, the quantities of the generated waste volumes should be limited to the greatest extent possible without jeopardizing the safety. Therefore, the existing cleanup and treatment systems should be used whenever possible in order to reduce the time as well as the cost expenditure for the cleanup process.

There exist two major systems that could be utilized for the cleanup: "the reactor water cleanup system" (System 331) and "the liquid waste system" (System 342). Of these two systems, System 342 is favoured due to the following reasons.

First of all, a prerequisite for using System 331 is that the inner isolation valves in the shutdown cooling system (System 321), which feeds System 331, are opened before the water level has reached the level of the valves during the filling of the reactor containment. It is, however, uncertain whether the isolation valves will stay open (Refer to section 5.3).

Secondly, if System 331, which is located in the reactor building, is to be used in connection with the cleanup process, the spent sorbents must be transported hydraulically to storage tanks located in the radwaste building. As the transportation distance is about 200 m, there will be a risk for settling of sorbents, partly damaged by radiation, in the pipelines. This would result in clogging of parts of the transportation pipes. As the specific activity of the spent sorbents will be high, a clogging of the transportation pipes may endanger the progressive cleanup process.

Due to the high radioactivity content of the water the running time of the filters in System 331 will be very short (about 15 to 20 minutes) until the ion exchanger is loaded. Hence, there is a considerable risk of radiation damage of the resins leading to clogged vessels and pipes.

Furthermore, zeolites are recommended as sorbents because of their high radiation resistance. However, a drawback could be that the filter vessels in System 331 contain inlet and outlet strainers but no filters to trap "fines", which may be released from inorganic sorbents. From considerations of space and under the existing circumstances such filters can be difficult to install in System 331.

As a consequence of the above-mentioned disadvantages, it is not recommended - at least not during the major part of the cleanup process - to use the filter vessels in System 331 in the cleanup process of the highly radioactive water which is stored in the reactor vessel and in the reactor containment. Closer to the end of the cleanup, it could possibly be considered to operate the filters in System 331 in parallel together with the filters in System 342 in order to speed up the final cleaning of the water. This would be meaningful only after the water has been decontaminated to more than 95 %, because before that the solidification capacity is the limiting factor.

The cleanup procedure is started by pumping water from the reactor containment via the containment vessel spray system (System 322) to receiving tanks installed in the waste building.

The water needs to be prefiltered in order to remove possible mechanical particulates, as these could rapidly clog the ion exchange vessels. System 342 is equipped with disc filters which can be precoated with inert inorganic filter aids and can thus be used as pre-filters. If necessary, the discs can be coated with a mixture of inert inorganic and organic filter aids and powdered ion exchange resins. The major cleanup of the ionic contaminants would be in the ion exchange vessels filled with inorganic cesium- and strontium-selective sorbents, i.e. with zeolites. After the decontamination step obtained by the sorbents, a final mechanical filtration should be performed in order to trap "fines" which may have been released from the sorbents. Spent filter aids and spent sorbents are flushed to separate receiving tanks installed in the waste plant. As the backflushing pipes from the filters and ion exchangers are very short and straight, the risk for clogging is negligible.

The decontaminated water is pumped back to the containment via System 322 - the spray nozzles in the drywell of the containment and the sieves in the condensation pool - as it is of utmost importance to maintain the water balance within the containment and thus keeping the water level in the reactor vessel above the reactor core.

A recycling of the water between the containment and the waste plant can be performed simultaneously as different pipe lines will be used for each transport direction. Only minor alterations, e.g. the addition of some extra pipes, which can be performed after the incident has occurred, are required in the waste plant (cf. Chapter 10).

The pumps in System 322 are taking the water merely from one point, the bottom of the condensation pool, and water can be returned above the water level in the containment. A further recirculation flow may be needed between the condensation pool and the other parts of the containment and the reactor vessel to enable that the whole water volume within the containment will be treated. A mixing of the water can be achieved by means of the Systems 322 and 323 (The core spray system). It is not necessary to maintain a continuous recirculation flow, but only when samples are showing unacceptable differences in concentration of the contaminants within the different parts of the containment and the reactor vessel.

To avoid a major contamination of a substantial part of System 342 in the waste plant, some parts of the system should if possible be isolated and used only for handling of the highly contaminated water. This can be done in case of the reference reactor, because the waste plant has a flexible construction allowing for many handling options.

Due to the possibility of a separation of different treatment lines in the liquid waste system, also low level waste water generated from the second, intact reactor can be treated in the same system without any risks for being mixed with highly contaminated water. Thus, the first reactor unit can be operated at full power during the shut down and cleanup period of the damaged reactor.

The cleanup of the water and the solidification of the generated waste will in principle be performed in the same way as during the normal operation of the nuclear power plant.

It should be pointed out that the cleanup system, including performed modifications, must be tested especially concerning the handling of the inorganic sorbents which are intended to be used for the cleanup purposes after an AB<sub>m</sub> event.

It is presupposed that the cleanup of the water from the damaged reactor will start about three months after the incident has occurred, partly to enable shortlived nuclides to decay and partly to be able to perform modifications and tests in the waste plant. Time will also be needed to train personnel for the forthcoming tasks, and for discussions with authorities about the planning of the cleanup, acceptable dose rates on waste containers, etc.

## 7.2 The various steps of the cleanup process

The water cleanup procedure can be subdivided into the following main parts.

The first step in the cleanup procedure is to decrease the cesium concentration in the water within the containment (including the water within the reactor vessel) to an acceptable low degree of contamination. At a cesium concentration of about  $7 \cdot 10^8$  Bq/m<sup>3</sup>, the reactor vessel head may be disconnected and a mixing of the water within the containment with the water in the reactor pool and the fuel storage pools can be accepted considering radiation doses.

With the assumption that the total water volume within the containment including the water in the reactor vessel is thoroughly mixed with the water volume in the pools, the cesium concentration in the pool water will amount to about  $5 \cdot 10^8$  Bq/m<sup>3</sup>, which will result in an acceptable dose rate of 0.1 mSv/h when working on the reactor service bridge.

The cleanup procedure will be performed as a feed and bleed process. The water is circulated between the containment and the cleanup equipment in the waste treatment plant. This means that the decontamination rate will decrease markedly as the cleanup proceeds. Without special arrangements a 99.97 % purification is needed in order to get the above mentioned cesium concentration and thus permitting a disconnecting of the reactor

pressure vessel head. This work itself requires only a 98.5% purification of the water in the containment, assuming an acceptable dose rate of 0.1 mSv/h at the work place on the bottom of the reactor pool. But a prerequisite for the lower overall purification factor must be that the water in the reactor vessel has a better purification factor to allow the mixing of this water with the water in the reactor pool and the fuel storage pools. This should be possible to accomplish by, for example, supplying the reactor vessel with demineralized water from System 733 or some other water with very low concentration of radioactivity.

It has not been investigated in detail if the suggested procedure, which significantly shortens the required cleanup time, is possible to carry out. Also, it has not been studied whether or not there will be a rapid mixing between the low contaminated water magazine in the reactor vessel and pools and the higher contaminated water in the containment. With only a minor mixing of the water in the reactor vessel and the containment, the above described method could be feasible. A further study is recommended.

Prior to disconnecting the reactor vessel head, the undamaged fuel stored in the fuel pools should be removed in order to prevent contamination by the damaged fuel. In principle, undamaged fuel should be stored separately from damaged fuel.

The next phase of the cleanup procedure will be focused on keeping the cesium concentration in the pool water as low as reasonably achievable while internal parts and the fuel are unloaded from the reactor pressure vessel.

During the unloading stage both the pool water cooling and cleanup system (System 324) and System 342 in the waste plant, which is connected to the containment, will be in full operation. If System 331 is available for cleanup purposes, also this system may take part at this stage of the decontamination process. As the pool water is mixed with the water in the reactor vessel and the containment a high concentration (as compared to the concentration of the radioactive species) of non-radioactive species (e.g.  $\text{Ca}^{2+}$ ) will be treated in System 324. Thus, in order to prevent a rapid loading of the filters in System 324, cesium and strontium selective sorbents (i.e. zeolites) should be used as a precoating material for the filter cartridges. If using only organic ion exchange resins, these will be rapidly exhausted by the non-radioactive ions.

The final step of the water cleanup process will start after the fuel has been unloaded and the reactor vessel head is reinstalled.

Before discharging the large volume of water (10 000 m<sup>3</sup>), it must be purified to an acceptable degree of low contamination. A fine-polishing of the water is performed by means of conventional organic ion exchangers. By this means the small amount of the remaining radioactive nuclides as well as the non-radioactive ions will be removed.

The major part of the water within the reactor containment can be emptied by the System 322. For emptying the last amounts of water, submersible pumps may be usable.

### 7.3 Water processing

#### 7.3.1 Water processing by inorganic sorbents

Due to high radiation resistance and selectivity, inorganic sorbents (i.e. zeolites) will be used for a selective removal of cesium and strontium ions from the water phase, whilst other ions will remain in the liquid phase (Refer to Chapter 6).

In order to prevent a fast build-up of pressure drop in the inorganic sorbents or to avoid a clogging of the bed volumes, the water that shall be decontaminated should, as a first step in the process, pass a mechanical filter. By inserting a prefilter in the cleanup line mechanical scrap and fines will be trapped, thus preventing at least to some extent a rapid clogging of the ion exchange vessels.

In the reference plant used in this study, there are installed disc filters in the liquid waste system. These filters can be precoated with either powdered organic resins or with some kind of inert organic or inorganic filter aids. A mixture of both can also be used. Precoating and backwashing of the filters are fast and easy operations and the spent filter aids are collected separately from spent granular sorbents.

The detailed cleanup procedure by means of the prefilters must be adjusted to the prevailing circumstances. Via the sampling system (System 337 or System 821) there will be given an indication about the kind and the concentration of the impurities which must be taken care of, i.e. the existing types of solid particles in the water phase. This will indicate the type of precoating material that should be used in the prefilters.

The ion exchange vessels belonging to System 342 are manufactured for using granular organic resins with a certain mesh size. Therefore, when employing other types of sorbents for this special cleanup purposes, the grain size of the sorbent must be large enough to avoid a release of the granules through the outlet screens of the ion exchange vessels. Furthermore, the sorbents should be sintered i.e. have a certain mechanical strength to prevent, or at least to minimize a release of "fines". The sorbents intended to be used for this special type of processing water, should also have a certain large grain size in order to prevent a fast build-up of the pressure drop during the filtration.

The amount of radioactive matter that is allowed to be adsorbed by the sorbents depends on the maximum allowable surface dose rate on the containers in which the solidification will take place. When solidifying radioactive waste in a concrete mould with a wall thickness of 0.25 m the radioactive load into the mould amounts to 35-40 TBq cesium ( $^{134}\text{Cs}$  and  $^{137}\text{Cs}$ ) for the chosen limit of the surface dose rate, i.e. 300 mSv/h for zeolites (Refer to Section 4.6). As the waste load into the mould will be about  $0.17 \text{ m}^3$  the specific activity of the sorbents will amount to 205 - 235 TBq/ $\text{m}^3$ .

After the ion exchange vessel a mechanical filter needs to be installed with the purpose of trapping fines which may have been released from the inorganic sorbents. Hence, mostly cesium, strontium and solid particles will be separated from the water phase during this stage of the cleanup process.

The water which is pumped back to the reactor containment will contain only minor quantities of dissolved radioactive species and the major part of

the dissolved non-radioactive salts. The major part of these dissolved substances will be separated from the water in the final stage of the purification, i.e. after the fuel has been unloaded from the core.

### 7.3.2 Possible microbiological growth

The water stored within the containment may contain organic materials such as humus and oils. This will cause a bacterial growth in the water. If any compounds or bacteria are present in the water, the pre-filtration step can hardly be performed as a continuous operation, because the pressure drop over the filters may increase rapidly. Frequent backwashing operations will thus be needed. The oil can, at least to some extent be separated from the water phase by a pretreatment step. Organics (humus, bacteria, etc) can as shown in TMI-2, to an acceptable degree be controlled by a strong oxidizing agent such as hydrogen peroxide (KAT 87). However, as the reference reactor used in this study is intended to be restored and put into operation again, the use of a strong oxidizing agent should possibly be prevented as there may be risk of attacks on some of the construction materials. This must be investigated further. In this study no investigations have been performed concerning the destruction of the organics.

### 7.3.3 Processing of the water by evaporation

In the AB<sub>m</sub> case the use of the evaporator is not recommended because this will actually expand the area which needs to be decontaminated and elongate the overall cleanup time.

First of all, using a feed and bleed process when treating the 10 000 m<sup>3</sup> of contaminated water in the containment will take approximately six years of continuous operation with an evaporator capacity of 20 m<sup>3</sup>/day.

Secondly, as the water used for filling up the containment was ordinary tap water and raw water, the concentration of non-radioactive salts may amount to 25 g/m<sup>3</sup> concerning calcium. After the concentration step by evaporation, the calcium concentration in the evaporator bottoms that shall be solidified will amount to 2.0 kg/m<sup>3</sup>. At this concentration the solubility product for some inorganic salts will have been reached or exceeded, e.g. calcium sulphate will precipitate. A precipitation of salts in the evaporator should be avoided as this makes cleaning the evaporator very difficult.

Finally there will be no time-saving or no gain in volume reduction of the waste by treating the water by evaporation, as the immobilization of the evaporator bottoms will result in the same production of concrete moulds as in the zeolite case.

### 7.3.4 The use of organic ion exchange resins

Prior to being released to the recipient, a remaining "fine-polishing" of the low level waste water stored in the containment may be necessary as it is expected that the water will contain small amounts of radioactive species. The final cleanup step should therefore be performed by means of organic ion exchange resins. During this final decontamination procedure, it should be kept in mind, that the presence of non-radioactive bivalent ions - especially calcium ions - may have an adverse influence on the decontamination process, i.e. increasing the waste volume and cleanup time.

To prevent a long-drawn-out procedure of this final cleanup step, which will be the case if using a feed and bleed procedure, a once-through decontamination through the resins should be aimed at. This can be obtained by connecting two ion exchange vessels in series of which the first one will trap the calcium ions and the second one is used for trapping the radioactive species. As mainly cation exchange resins will be used for this final cleanup step the pH of the effluent from the ion exchange vessels may drop somewhat, and thus the water may need to be neutralized prior to the release to the recipient.

### 7.3.5 Special filters

In System 342, a use of special filter cartridges could be feasible. This would be small (probably radiation shielded) filters filled with zeolites and placed in series. The series could consist of three or more filters and once a break-through of cesium has been reached after the first filter, this filter could be disconnected and directly solidified in cement and a fresh filter is connected to the end of the series. This method has not been studied in detail and must be investigated separately but is possibly the best solution considering that in this case study more than one reactor is connected to the waste plant. A similar system was the Submerged Demineralizer System (SDS) used at TMI-2 (TOT 86), in which cartridges containing zeolite were used for treating the highly contaminated water.

### 7.3.6 Doses to the environment from water released to the recipient

The release of a large volume of water with a small concentration of radionuclides exposes the public to a radiation dose. The effective dose equivalent to the most exposed individual members of the public from normal operation is limited to 0.1 mSv per year in regulations issued by the National Institute of Radiation Protection (SSI). The dose shall include the contributions from all releases, to the air as well as to the water, from all nuclear power reactors on the site. It seems appropriate in this case to have the ambition to limit the release to 10 % of the reference value, i.e. 0.01 mSv. It is stated in the same regulations that the global collective effective dose equivalent commitment from releases to the environment during normal operation should be less than 5 manSv per year and GW of installed electric capacity, which means less than about 11 manSv for the Oskarshamn site.

Before the final cleanup step starts the dominating radionuclides are Cs-134 and Cs-137. If it is assumed that the total cesium activity at this time is  $7 \cdot 10^{12}$  Bq (cf Section 7.2), the maximum dose to the most exposed individuals will be approximately 0.1 mSv and the collective dose approximately 2 manSv. The dose factors (respectively mSv and manSv per Bq emitted) have been taken from the Final Safety Analysis Report for Oskarshamn 3.

A purification factor of 10 is required for the final cleanup step in order to meet the objective set above for the individual dose and as a consequence the collective dose will be only 2 % of the limit value. In deciding what the degree of purification should be - after sampling the water - other factors may be considered. It should, however, be no problem to obtain the desired cleanup of the water before releasing it to the recipient.

## 7.4 Immobilization of the generated waste

### 7.4.1 Waste container

The low and medium level radioactive waste generated during normal operation is immobilized in the solid waste system (System 343). In the reference plant the waste is solidified with cement into prefabricated cubical concrete moulds with an external dimension of 1.2 m. In Section 4.6 the relation between radioactive load and surface dose rate has been calculated for the two types of mould most frequently used and having a wall thickness of 0.10 m and 0.25 m, respectively.

The internal volume is either 1.0 m<sup>3</sup> or 0.34 m<sup>3</sup>, of which roughly one half is the waste load. The remaining volume will be needed for additives and cement, in order to obtain a solid waste product with an acceptable crushing strength and an acceptably low leaching rate concerning cesium.

Owing to the high specific activity of the waste generated in the AB<sub>m</sub> case (and also in the TB<sub>m</sub> case) moulds with the stronger radiation shielding must be chosen for the major part of the waste to be solidified. However, during the final "fine-polishing" stage of the cleanup procedure, the generated waste may be characterized as low or medium level, and thus a less strong radiation shielding may be sufficient.

### 7.4.2 Solidification of inorganic sorbents

The solid waste system (System 343) is designed for the solidification of organic ion exchange resins.

Cement is a good solidification matrix for zeolites (cf. Chapter 6). However, preparatory solidification test should be performed by using mixtures of inorganic sorbents intended for the cleanup purposes and water with a chemical composition representative of the water expected in the containment at the time of the cleanup. As zeolites can be used to improve the leaching properties of cesium in cement (TOR 89), the solidified product is expected to be well suited for final storage, e.g. in a rock vault.

It should be mentioned though, that cement solidification of mixtures containing zeolites or titanates has been performed in a full scale operation (0.2 m<sup>3</sup> drums) at an earlier stage by ABB Atom and no mixing and no curing problems were registered when solidifying the waste to a water-to-cement ratio of about 0.50.

As the mechanical properties of zeolites and organic resins are different, the crushing strength of the solidified waste product will vary depending on the sorbent and must accordingly be measured for any particular waste mixture.

With the use of special once-for-all zeolite filters the cement casting has to be modified. In this case a stirrer cannot be used and accordingly the final mixing of the cement cannot be done in the mould. As discussed above, the use of special filters has not been studied in any detail, but one way of solidifying the filters might be to let a prepared cement slurry pour on top of the filter cartridge and have the whole mould vibrated giving a uniform spread of the cement throughout the mould.

### 7.4.3 Solidification of organic resins

The maximum permissible dose rate on moulds containing organic ion exchange resins is in Sweden at present 30 mSv/h. This dose limit is regardless of whether the waste originates from normal operation or from an accident cleanup. Hence, both bead and powdered resins are solidified in the same way as under the normal operation of the station.

This implies that the properties of this kind of waste generated during the accident cleanup will be very similar to the properties of the waste generated during the normal operation.

### 7.4.4 Solidification capacity

During the water cleanup process, the generated waste must be solidified gradually as the storage capacity for the spent sorbents will be limited. As the solidification capacity is not dimensioned for taking care of the large amounts of waste produced after an accident as described above, the capacity of the immobilization system will govern how fast the initial cleanup procedure will progress.

The most time-consuming step of the solidification process is the curing time of about 48 hours prior to casting a non-radioactive concrete lid on top of the curing waste mass. In order to speed up this step of the solidification process, a prefabricated concrete lid has been developed. By using this kind of sealing method of the moulds, the capacity of the immobilization system can be increased by a factor of two to three.

With this improvement and work in three shifts each day the solidification capacity increases to 42 moulds per week. Taking into account the necessary time for maintenance, interruptions of the operation, the capacity required for unit 1 at the site etc, the annual capacity is conservatively assumed to be 960 moulds - corresponding to 160 full working days per year.

The solidification capacity can be substantially increased if a mobile solidification facility is connected to the existing permanent system. In this way the total annual capacity can be increased to 1600 moulds.

## 7.5 Time required for the cleanup and waste quantities produced

The time required for cleanup of the water in the reactor containment, including the reactor vessel, depends on the capacity of the cleanup and solidification systems and on the possible radioactive load in the concrete moulds. The normal time for unloading the fuel is also included in the total time as well as the time to empty the water from the containment and to make the final cleanup step - the "fine-polishing".

The estimated time is based on the possibility to remove the reactor vessel head before the desirable average cesium concentration of  $7 \cdot 10^8$  Bq/m<sup>3</sup> has been reached in the water within the containment (Refer to Section 7.2). Here, an overall purification factor of 99.8 % has been assumed.

The cleanup rate is initially governed by the solidification capacity since no spent sorbents are assumed to be stored. At the end, however, the purification capacity is the governing factor. This is a consequence of the feed and bleed procedure used for the major part of the cleanup.

The radioactive load in a concrete mould with 0.25 m thick walls is 35 to 40 TBq cesium when the sorbent is zeolite (Refer to Section 4.6). The total amount of cesium radioactivity to be taken care of is about  $2.7 \cdot 10^4$  TBq (as calculated three months after the event) in the first and major cleanup step. This corresponds to a production of about 700 moulds.

During the unloading and storage of the damaged fuel System 324 is in full operation. As a rough estimate the waste production from this system results in about 5 moulds.

The final cleanup step, the purification of the water before releasing it to the recipient, generates spent organic resins. About 75 thin-walled moulds are required in the solidification of this waste.

As a conservative estimate the complete water cleanup procedure produces about 800 concrete moulds, including any small contribution from the cleaning of leakage water (Refer to Section 5.6). With the existing solidification system, including the use of prefabricated concrete lids, this procedure is completed about two years after the occurrence of an  $AB_m$  event. If also a mobile solidification system is used, the necessary time will be reduced with about four months.

It is recommended to use inorganic sorbents for the major cleanup. But if for some unforeseen reasons this is not possible, the alternative is organic resins. The possible radioactive load in a concrete mould with 0.25 m thick walls is about 5.5 TBq cesium. Consequently, the required number of moulds will be much higher than in the case of zeolites and so the necessary time increases considerably. Furthermore, the impact of radioactive decay and an additional solidification system is significant.

In the event of organic resins as the only sorbent the complete water cleanup procedure produces about 3 300 concrete moulds and is completed after about 4.5 years, using the lower solidification capacity. With the higher capacity the production is about 3 800 moulds, but the time is reduced to about 3.5 years.

The number of concrete moulds and the necessary time in this case demonstrates well the obvious advantage of using inorganic sorbents during the major part of the cleanup.

The content of alpha radioactive nuclides in the waste is of special concern with regard to the assumed final disposal in the SFR (Final Repository for Reactor Waste) facility at the Forsmark site. This facility is approved for the reception of, for example, 0.4 TBq Pu-239 and 1 TBq Am-241, while no figures are given for uranium and neptunium isotopes. According to Table 3.5 there is 45 kBq Pu-239 and 25 kBq Am-241 in solution and hence available for incorporation in the solidified waste. Apparently, the waste produced during the water cleanup can be accepted in SFR with respect to the amount of alpha radioactive nuclides.

The situation with respect to the amount of cesium isotopes in the waste generated during the cleanup process is not so favourable. In the safety analysis performed for SFR it has been assumed that the radioactive inventory is maximum 810 TBq and 4 900 TBq for Cs-134 and Cs-137, respectively. These limits are exceeded in the study. However, a decay time of about 9 years before disposal will decrease the inventory of Cs-134

to below the limit. For Sr-90, the third radionuclide of interest, the situation is about the same as for Cs-137.

Consequently, an additional safety analysis and a new licensing procedure will be required before transportation to SFR is permitted. In practice, however, there should be no real problem to accept the waste in question and only administrative measures are expected to be necessary.

## **7.6 Handling and transport of waste containers**

### **7.6.1 Handling in the waste building**

Prior to the transportation of the filled and sealed concrete moulds from the solidification cell, which is located below ground level in the waste building (Refer to Figure A3.4), the moulds will be subject to an automatic measurement of the surface dose rate. Then, the moulds are transported up to the ground level and placed in a loading position (Refer to Figure A3.5, point C) by means of a traverse crane, which is operated from a radiation shielded place equipped with steel walls and a lead window. Due to the steel walls the operator will have a limitation in visibility and thus parts of the operation of the traverse crane must be performed radio-controlled from a less shielded place on the ground floor. Prior to loading the filled waste containers on the transport trailer, the doors in the waste building have to be opened manually.

It is possible to obtain a temporary storage in a room adjacent to the loading position (Refer to Figure A3.5, point D). By piling up a couple of hundred of empty moulds with 0.25 m walls a storage is created for about a hundred filled moulds. The pile should be two layers thick, equivalent to shields of 1 m concrete.

To avoid some of the manual handling in the waste building, complementary equipment is proposed to be installed. This will be discussed in Chapter 10.

### **7.6.2 On-site transportation and storage**

The on-site transportation and related radiological problems have been discussed in Section 4.6. Although the transport trailer does not meet the IAEA transport regulations with respect to the radiation level outside the trailer, the transportation can be carried out by either closing the route or providing the trailer with an additional shield.

On arrival at the on-site storage each waste container is subject to a *gamma*spectroscopic examination. The waste container is then lifted into a position in the storage using a traverse crane.

The on-site storage has room for approximately 3 000 concrete moulds and is used as an intermediate storage before the transportation to the final disposal in the SFR facility at the Forsmark site. There will probably be no problems to take care of the waste containers produced in connection with the water purification after an AB<sub>m</sub> event. In case organic resins are used even for the major water cleanup, the concrete moulds now stored have to be moved to SFR and so will the concrete moulds initially produced during the water processing.

## 8. HANDLING AND TREATMENT OF ADDITIONAL WASTES AFTER AN AB<sub>m</sub> EVENT

### 8.1 Contamination of the reactor containment

Due to the high temperature of the fuel at the time when cooling of the core can be reestablished, the release of fission products will be in a mixture of gas and steam. Even with maximum capacity on the containment vessel spray system (System 322) this will not be sufficient to condense all of the steam that is formed when cooling the core with the core spray system (System 323). Consequently, the steam will be relieved to the condensation pool. The water in the condensation pool is used as spray water both for System 322 and 323, thus, a uniform surface sorption of fission products can be expected both in the containment and the reactor vessel.

Of the released fission products and actinides the noble gases krypton and xenon and possibly a small fraction of the iodine content (<0.1%) will remain in the gas phase, while the rest will either be sorbed on surfaces in the containment, be dissolved in the aqueous phase or be precipitated in the bottom sediments (cf. Chapter 3). A compilation of the distribution of fission products and actinides between the aqueous phase and solid surfaces are given in Table 3.4 and 3.5, respectively. Of the radionuclides in the tables it is mainly cesium (Cs-134 and Cs-137) that contributes to the external radiation level.

In the following no attempt in estimating the distribution of the actinides between different waste categories has been made. This is because the total amount of actinides will be small and it will be almost impossible to trace them in the waste. If assuming a total surface area of 20 000 m<sup>2</sup> in the containment, this will give a uniform distribution of about 90 Bq/m<sup>2</sup> U-235 and U-238, 120 Bq/m<sup>2</sup> Np-237, 1.1·10<sup>5</sup> Bq/m<sup>2</sup> Pu-239 and 6·10<sup>4</sup> Bq/m<sup>2</sup> Am-241, which should be compared with a cesium surface contamination of about 10<sup>10</sup> Bq/m<sup>2</sup>.

Relatively little is known about the behaviour of radionuclides in a water-filled containment, although the TMI-2 accident has given some guidelines. In Table 8.1 the sorption of cesium and strontium on different surfaces is given. The data are mainly a compilation of measurements done in the TMI-2 containment (TOT 86). Observe that the data given in Table 8.1 are measured mean-values; large variations were found, e.g. due to "hot spots".

A rough estimate of the sorption of cesium on various components, pipes and surfaces in the Oskarshamn 2 containment is given in Table 8.2, using data from Table 8.1. Data given in Table 8.2 are approximate as there are uncertainties in estimated surface areas and because TMI-2 is a PWR with only a couple of months operation time, thus, having a completely different history than Oskarshamn 2. Nevertheless, the values give a picture of the possible contamination of the containment.

**Table 8.1. Strontium and cesium sorption in connection with the TMI-2 accident. (Values represent measured mean values in TMI-2).**

Nuclide	Sorbed on inner parts of the reactor vessel (GBq/m <sup>2</sup> ) a)	Sorbed on pipes in the containment (GBq/m <sup>2</sup> )	Sorbed on concrete surfaces in the containment (GBq/m <sup>2</sup> )
<sup>90</sup> Sr	1.3	0.02	0.02
<sup>134</sup> Cs b)	67	6	6
<sup>137</sup> Cs	67	6	6

a) These values are rough estimates from the TMI-2 accident performed prior to an access to the reactor vessel was possible and are thus to be considered only as guidelines.

b) Same values assessed as for Cs-137 because 3 months after the assumed accident the Cs-134/Cs-137 ratio is approximately 1 in Oskarshamn 2, while in TMI-2 the core inventory of Cs-134 was comparatively much less.

As can be seen in Table 8.2 there is a large sorption in the insulating material and due to large surface areas also on grating, pipes, electrical cables and concrete floors and walls. Typical gamma dose rates expected in the containment vessel two years after the accident are 25 to 40 mSv/h from concrete walls and 8 to 10 mSv/h from a pipe (length 5-10 m, diam. 0.21 m) with 140 mm thick mineral wool insulation, both at a distance of 1 metre.

Furthermore, the cesium isotopes will contribute to a beta dose rate to the skin of about 7 mSv/h at a distance of 1 m from an unshielded concrete surface. The high beta radiation could cause special problems, e.g. the dose monitoring. In TMI-2 a thermoluminescence dosimeter (TLD) was developed especially for this purpose.

## 8.2 Cleanup of surfaces and equipment

A number of techniques, like water spraying, chemical decontamination etc., have been tested experimentally and/or during decontamination and decommissioning of nuclear plants (DEC 85, TAR 87).

In Oskarshamn 2, due to the gamma radiation from Cs-134 and Cs-137 the initial cleanup must be performed using remote controlled equipment. Examples on how a versatile remotely controlled vehicle can be equipped are given in the literature (TAR 87, FOL 86) where especially the experience gained during the TMI-2 cleanup can be utilized. In the following a few decontamination techniques are briefly described in the order they might be used during a cleanup.

**Table 8.2. Estimated surface sorption of cesium in the containment two years after an AB<sub>m</sub> event. Values given for Oskarshamn 2 and using data from Table 8.1.**

Material	Surface area (m <sup>2</sup> )	Surface sorption (TBq)
Cables	900	8
Components	6	0.05
Concrete surfaces	2000	18
Grating	2500	23
Insulation a)	100 b)	26
Insulated pipes c)	1000	9
Non-insulated pipes d)	3000	27
Motors	50	0.5
Pumps	13	0.1
Vent. system	100	0.9

a) Values evaluated using the equation  $K_d=q/c \cdot V/m$ , where  $q$  = sorbed on the solid (Bq),  $C$  = in solution (Bq),  $V$  = volume of the solution (m<sup>3</sup>) and  $m$  = weight of the solid (kg). A  $K_d$  of 0.0006 m<sup>3</sup>/kg is used, which is the sorption of cesium in cement paste (ATK 88).

b) Given as volume in m<sup>3</sup>.

c) Mainly the dry well. The surface represents pipes stripped on insulation. The surface sorption is calculated using the assumption that the concentration of cesium is the same in the insulation and the liquid.

d) Mainly the wet well.

### 8.2.1 Decontamination using water

The first step when decontaminating the reactor containment would be water spraying. As the initial step accessible surfaces are rinsed using water with low pressure. By thus removing any loose contaminated dust and debris, a re-contamination of clean areas is minimized during the continuing decontamination. In Table 8.3 the expected radioactive contamination on various surfaces after the initial containment water cleanup is given.

After rinsing with water, ultra-high-pressure (UHP) water jetting can be used to clean concrete surfaces and other painted surfaces. At TMI-2 UHP water spraying removed 50% of the contamination on concrete surfaces (the outer loosely bound contamination layers). Furthermore, by removing paint using UHP the dose rate on some equipment was decreased by 50% and the beta radiation by 90%. But at TMI-2 certain types of paint could not be removed using UHP water. The dose rates from concrete surfaces remaining after UHP water jetting would be around 15 mSv/h.

**Table 8.3. Additional wastes a) two years after an AB<sub>m</sub> case in Oskarshamn 2.**

Type of waste	Mass (kg)	Volume (m <sup>3</sup> )	Total surface area (m <sup>2</sup> )	Surface activity (GBq/m <sup>2</sup> )	Total (GBq)
Ventilation system (fans)	700	1	25	7.2	180
Cables	11000	1.9	900	7.2	6500
Motors	3800	2.4	50	7.2	360
Components, Solenoid valves etc)	2000	1	6	7.2	43
Mineral wool insulation	4500	30 b)	-	-	5500 c)
Caposeal (reactor vessel insul.)	17000	70 d)	-	-	20000c)
Concrete surfaces	9200 e)	4	2000	7.2	14000

a) Additional wastes include everything except ordinary operational wastes (overalls, shoe protection etc) and spent ion exchange resins etc. from the cleanup of the containment water. The surface activity after the containment water cleanup is expected to be 80% of what was originally sorbed.

b) Given as undamaged mineral wool. Density = 150 kg/m<sup>3</sup>. Porosity 0.90 gives a rest of 3 m<sup>3</sup> waste.

c) Calculated using a cesium sorption on cement of:  $K_d = 0.0006 \text{ m}^3/\text{kg}$ .

d) Given as undamaged insulation. Density = 240 kg/m<sup>3</sup>.

e) A uniform layer of 2 mm assumed. Density (concrete) = 2300 kg/m<sup>3</sup>.

UHP water jetting can also be used to remove mineral wool insulation on pipes etc, and to open up the protective metal sheets covering the insulation, thus, making it possible to take off some or all of the underlying insulation. A stripping of the insulation would decrease the radiation level from the above mentioned pipe at a distance of 1 metre to about 2 to 5 mSv/h depending on how much of the insulation that can be removed.

Water spraying can however, cause problems in reactors where the reactor vessel is insulated with "Caposeal" as in Oskarshamn 2. Caposeal, which is an asbestos gypsum-like material softens and decomposes in contact with water. Thus, those parts of the reactor vessel which are submerged in water will be stripped off the insulation. Not only will this contribute to a significant waste volume, but it would not be possible to restart the reactor until the reactor vessel is reinsulated. Hence, a postaccident

handling where the outside of the reactor vessel could be kept dry is preferable. With the reactor vessel submerged in water, this would produce sludge consisting of 70 m<sup>3</sup> insulating material with a radionuclide content of about 0.4 TBq/m<sup>3</sup> cesium.

### 8.2.2 Chemical decontamination

A number of chemical methods are available for various applications. In order to make a restart of the reactor possible within a reasonable time and cost, only weak decontamination processes should be used. The weak methods (low concentration processes), i.e. methods resulting in insignificant or only minor chemical attack on the underlying material, often consist of treatment with an oxidizing agent (e.g. potassium permanganate), thereafter with a weak organic acid (e.g. oxalic or citric acid) and finally rinsing with water. Frequently a complex forming agent (like EDTA) is included in the process, but this is not necessary when removing cesium; as in the AB<sub>m</sub> case. On some nuclear power plants chemical decontamination is used as part of the station maintenance, hence, reducing the overall plant radiation level.

Possible problems when using chemical decontaminants could be corrosion attacks and a waste that is more difficult to treat and solidify.

After water spraying the dose rate inside the containment vessel is probably still several mSv/h. Thus, to allow for dismantling of equipment a remotely controlled chemical decontamination using a low-concentration process is probably necessary. In particular the inner surfaces of the pipeworks in Systems 321, 322 and 323 should be washed. Depending on the degree of the contamination of the inner parts of these systems, three different levels of decontamination of the systems can be envisaged:

- 1) Cleaning the systems outside the containment vessel, thus reducing the radiation level in the reactor building but at the same time minimizing the use of decontamination solution to some tens of m<sup>3</sup>.
- 2) Decontaminating parts of systems 322 and 323 by pumping through the condensation pool and through pipes outside the containment.
- 3) Pumping with all three systems and also spraying with 322 and 323, hence, cleaning also the inner parts of the reactor vessel and drywell.

In cases 2) and 3) a total of around 1 000 m<sup>3</sup> decontamination solution will be needed. A chemical decontamination could reduce the contamination level by 50 to 90%.

High-concentration processes can be used on dismantled parts in a special decontamination bath. This would reduce the volume of the liquid radioactive waste and decontaminated parts could possibly be treated as non-radioactive material.

### 8.2.3 Mechanical decontamination

As a final decontamination of the containment it is probable that remaining paint and possibly a layer of the underlying concrete need to be removed. A further use of UHP water jetting or a chemical decontaminant can probably be used to remove the paint, while layers of the concrete can be removed by numerous methods, e.g. scabbling, where carbide tipped pistons rapidly impacts the surface; diamond grinding, where a large number of closely spaced blades cuts away the concrete surface; and chipping, where tools like e.g. jackhammers are used to remove the concrete. As all mechanical methods normally are quite dusty, water misting can be used to minimize dust.

If the complete decontamination could reduce the over-all dose rates in the containment to below 0.3 mSv/h, it will thereafter be possible to enter the containment.

### 8.3 Generated waste volumes

As a precondition used when estimating the generated waste volume it is supposed that the containment is filled with water for two years. The estimated waste volume and the amount of cesium sorbed on this waste are given in Table 8.3. Note that the levels of cesium sorbed are calculated using sorption data measured on surfaces prior to the decontamination of the containment, but with the assumption that 20% of the originally sorbed cesium has desorbed during the cleanup of the contaminated water used in the containment to ensure the continuous cooling of the damaged core. As the sorption of cesium is an equilibrium process the gradual cleanup of the containment water will decrease the surface sorption. Thus, desorption of cesium has at the time when access to the containment is possible reduced the contamination level as compared to the values given in Tables 8.1 and 8.2. A qualitative value is difficult to give as desorption is a much slower process than sorption, but in Table 8.3 the values represent an expected desorption of 20% of what was initially sorbed. Cesium in the insulating material has probably to a large extent remained sorbed, due to not only a slow desorption process but also due to slow diffusive transport.

All electrical equipment submerged in water for two years is destroyed (GOE 85). In TMI the most common cause for malfunctioning electrical equipment was moisture. An AB<sub>m</sub> case accident in Oskarshamn 2 would lead to 11000 kg (1.9 m<sup>3</sup>) electrical cables wasted, 3800 kg (2.4 m<sup>3</sup>) electrical motors (basically from control rod drives), about 2000 kg (1 m<sup>3</sup>) electrical components, and 700 kg (1 m<sup>3</sup>) fans from the ventilation system that need to be exchanged due to water damages. (Note that the given volumes are the physical volumes without taking the shape of different objects into consideration. Thus, without compactation actual waste container volumes would be much larger).

Furthermore, all mineral wool insulation is expected to be ruined. In Oskarshamn 2, all pipe insulation in the containment is made of mineral wool, giving a volume of 30 m<sup>3</sup> (undamaged) mineral wool. With a porosity of 0.9 this gives 3 m<sup>3</sup> waste having a cesium content of ca 1.8 TBq/m<sup>3</sup> in the compacted waste. With the reactor vessel submerged in water an additional 70 m<sup>3</sup> of Caposeal insulation as sludge has to be taken care of. The cesium content in the Caposeal sludge will be around 0.3 TBq/m<sup>3</sup>.

In TMI-2 it was found that on concrete surfaces covered with epoxy paint radionuclides had sorbed, but there was no penetration into underlying surfaces. Only where the paint was damaged or on uncovered surfaces a significant penetration was observed (up to around 2 cm). On the epoxy paint the radionuclides were sorbed only in a thin layer with virtually no further penetration.

A removal of the paint and a thin layer of the underlying concrete surfaces would result in around 9000 kg concrete detritus (4 m<sup>3</sup> undamaged concrete) with a total cesium content of ca 3.6 TBq/m<sup>3</sup> as intact concrete. With a porosity of the concrete detritus of around 0.3 this gives roughly 6 m<sup>3</sup> having a cesium content of 2.4 TBq/m<sup>3</sup>.

There will also be an additional amount of normal operational wastes, like overalls and shoe protection etc.

Other pieces of equipment could probably be renovated (e.g. exchange of rubber washers, bearings, etc.) due to damages caused by water. Due to the quality assurance requirements all movable parts in the containment need to be either exchanged or taken to service.

Large volumes of water with a low radioactive content will be produced during cleanup of the containment vessel. How many m<sup>3</sup> of water that are needed is dependent on the contamination and the desirable degree of decontamination. Figures from similar events giving estimates on how much decontamination water that is needed have not been found in the literature, but most likely tens of thousands m<sup>3</sup> will be needed, but with recirculation of the spraying water, the actual physical volume of the water could possibly be limited to a few thousands m<sup>3</sup>.

An estimate of the number of waste moulds produced due to treatment of the spraying water has been done in the following way. It is assumed that 1% cesium of the source term is sorbed and 15% of this is sorbed in the insulating material. If 20% is removed during cleanup of the contaminated water and 50% of the remaining is decontaminated using water spraying this corresponds to 0.3% of the source term or about 65 TBq cesium about two years after the accident. Using a mould surface dose rate of 30 mSv/h, this could theoretically be solidified in about ten moulds using organic ion exchange resins as adsorbent. But considering the salt content of the contaminated water it is likely that up to 16 m<sup>3</sup> spent resins will be produced which, depending on whether the wall thickness of the mould is 0.1 or 0.25 m, corresponds to 50 or 100 moulds, respectively.

The number of moulds produced due to the chemical decontamination would be minor, presumably less than five.

## **9. THE TB<sub>m</sub> CASE**

### **9.1 Measures to be taken in the reactor building and the waste plant immediately following the commencement of the accident sequence**

#### **9.1.1 Sequence of events in the reactor containment**

In the event of a fast shut down of the turbine and steam dumping is not permissible, a fast shut down (scram) of the reactor will occur which will also initiate blow down from the reactor to the condensation pool through the relief system (System 314). System 322 will attempt to start cooling the condensation pool but this will not be possible because of the complete loss of electric power. Normally the speed of the main circulation pumps (System 313) is reduced to 20 % of full load speed, but in this case the pumps will come to a standstill because of the electric power failure.

The final stage of pressure relief of the reactor vessel is controlled by two pressure relief valves in System 314, which normally continues until System 321 takes over the cooling of the reactor. System 321 will fail to commence to cool the reactor due to the failure of electric power and so the water level in the reactor will be reduced as the water continues to boil off to the condensation pool.

When the water level in the reactor vessel has reached an extremely low level the blow down valves in System 314 will open so as to reduce the pressure in the reactor vessel and so allow the core spray system (System 323) to come into effect. System 323 will fail to operate because of the electric power failure and so the water level in the reactor vessel will continue to fall. The fuel will be damaged when the level of the water has fallen to the extent that the core is uncovered. The fission products that are released firstly contaminate the water in the reactor but are also transferred to the condensation pool by System 314.

The complete absence of electric power will mean that the condensation pool will not be cooled and the temperature of the water in the pool will rise. This implies that the temperature of the atmosphere in the containment will also increase and may cause the automatic spraying of the drywell in the containment when System 322 starts after reconnection of electric power. The possible contamination of the drywell will certainly involve a major decontamination work, but this is considered to be appreciably less than in the event of an AB<sub>m</sub> case.

#### **9.1.2 Sampling**

In principle there is no difference between the sampling methods used for cases TB<sub>m</sub> and AB<sub>m</sub>. On the other hand the level of radioactivity in a TB<sub>m</sub> case is approximately 40 times greater, which increases the requirement regarding the radiation protection of the personnel involved. Gas samples can be taken continually after a TB<sub>m</sub> case but only for some 48 hours following an AB<sub>m</sub> case.

### 9.1.3 Disconnection of the reactor water cleanup system

The disconnection of the reactor water cleanup system (System 331) is of extreme importance in the event that System 321 is started. A comparison with (ELK 85) shows that during an operating time of 30 seconds to 1 minute, the ion exchange resin will have absorbed sufficient fission products that the radiation dose to the resin will reach approximately 1 MGy within a matter of weeks. The ion exchange resin will be difficult to handle with that radiation dose as mentioned in Chapter 6.

In the event that System 331 is not disconnected, then the control room personnel must be informed that the ion exchanger in the system is being overloaded. The installation of equipment for radiation measurement is discussed in Chapter 10. This equipment is to provide an alarm if the radioactivity concentration of the reactor water would be too high.

### 9.1.4 Airborne radioactivity

The isolation valves in System 741 are not affected in a TB<sub>m</sub> case and will remain open against the reactor containment. As soon as electric power is reconnected the fans will commence to operate and the system will function again.

The ordinary ventilation system (System 742) in the reactor building will function normally as soon as the electric power is reconnected. Any diffused leakage will consequently be released to the environment by means of this ventilation system. This release can be avoided by the use of the emergency filters in System 341.

The TB<sub>m</sub> and AB<sub>m</sub> cases do not differ in any other respect in regard to airborne radioactivity.

## 9.2 Management of leakage water

### 9.2.1 Reactor building

The spreading of water borne radioactivity from the reactor containment and the reactor vessel in a TB<sub>m</sub> case results from leakage from the same systems as those concerned in an AB<sub>m</sub> case. This is valid for both those systems which are open to the reactor containment and those which are connected to the reactor vessel but can be isolated from the reactor vessel by means of the isolation valves.

However, the situation in System 321 can be different following the events of a TB<sub>m</sub> case. In preparation for a cold shut down, System 321 will normally come into operation as soon as the pressure in the reactor is reduced to 10 bar. System 321 can be brought into operation, once electric power has been restored after the events of a TB<sub>m</sub> case, when there is a need to reduce the temperature of the reactor to below 100°C. This is with the provision that System 331 is disconnected as the case stated in 9.1.3 above.

Cooling will continue even in the event that the decision is taken not to connect System 321. It is done through pumping of water from the condensation pool by System 323 and the blow down of steam by System 314. The method for cooling down in preparation for a cold shut down without the use of System 321 must be established by means of further investigation.

An estimate has not been made of the volume of the leakage from the containment following the events of a  $TB_m$  case, but this should be somewhere near the same amount as in an  $AB_m$  case. A good proportion of the leakage water in this case would come from the reactor vessel and would have a much higher concentration of radioactivity, approximately 40 times greater than that of the water in the containment in an  $AB_m$  case. Part of the leakage will come from the condensation pool and will have a lower concentration of radioactivity.

Additional leakage with an extremely high concentration of radioactivity may result in the event that System 321 is started. Another disadvantage of starting System 321 is that the radiation level will increase in many areas in the reactor building.

Those valves in System 322, which are normally opened when the water level in the condensation pool is high, are not closed in the event of a  $TB_m$  case (cf. Par. 5.5.2). The prevention of uncontrolled release of water to the waste plant must be provided by manual operation, i.e. by closing the valves from the control room.

### 9.2.2 Waste plant

The management of the leakage water in the waste plant following the events of a  $TB_m$  case are principally the same as that following an  $AB_m$  case and therefore a further analysis has not been carried out. One difference is that the leakage water from the reactor containment has both high and low radioactive components as described above, but these become mixed together in the receiving tanks. This set of conditions will also become balanced in time if the water is returned to the condensation pool, which is a convenient action for reasons given in Section 5.6.

### 9.3 Methods for cleaning the process water

The steam released to the condensation pool through System 314 transfers noble gases and to some extent iodine and cesium to the water in the condensation pool. The major part of the released radionuclides will however remain in ca 250 m<sup>3</sup> water in the reactor vessel. In this water the concentration of radioactivity is roughly 40 times higher than in the water to be processed in the  $AB_m$  case.

Without dilution it will take the ion exchangers in System 331 only about 30 seconds to adsorb enough radionuclides to reach the dose rate limit 30 mSv/h, accepted on the final solidified waste container (cf. Chapter 7). This is of course an impossibly short operational time to be feasible. Therefore it is most inexpedient to use System 331 for the major cleanup. Instead the water should be transferred to the waste plant, and preferably be diluted before being processed in the liquid waste system (System 342).

Due to the intense radiation, the best procedure for pumping this highly contaminated water from the reactor vessel to the waste plant is to use the drain pipes for spent ion exchange resins in System 331. This can either be achieved by pumping via heat exchanger 331 E3 to the feed tank in the evaporator line (Line 5), or by adding new pipes in the waste plant making it possible to pump the water to the receiving tank in the system drain line (Line 1). Of course, the ion exchanger used in System 331 must be emptied of spent resin, before the cleanup of the reactor cooling water starts.

As the contaminated water largely is limited to the reactor vessel and the condensation pool, it is easier to keep the water balance as compared to the AB<sub>m</sub> case. The decontaminated water is pumped back from the waste plant to the condensation pool and from there through the 323 spray nozzles into the reactor vessel.

After dilution the water could, as in the AB<sub>m</sub> case, be cleaned either by use of zeolites or organic resins. A dilution of the waste water can either be done by diluting the water in the reactor vessel, thus, both increasing the operation time of the ion exchangers and decreasing the dose rate from pipes, or it can be done by recirculating processed water in the waste plant either to one of the receiving tanks or directly to the waste stream going into the ion exchangers. The first method will decrease the overall radiation level in the plant but will take longer time, as it is a feed-and-bleed process. The second method will keep a high radioactivity concentration in the pipes but will take less time. Dilution of the waste water prior to being treated in the ion exchangers in System 342 can be carried out utilizing the demineralized water distribution system (System 733).

In the TB<sub>m</sub> case it may be feasible to concentrate the radioactivity by evaporation and then solidify the evaporator bottoms directly in cement. Only a moderate concentration by evaporation will be necessary as the specific cesium activity of the water stored into the reactor pressure vessel should be increased from about 110 TBq/m<sup>3</sup> to about 220 TBq/m<sup>3</sup>. With an evaporation capacity of 20 m<sup>3</sup>/day a specific activity of 220 TBq/m<sup>3</sup> can be obtained during approximately nine days of operation. It should be kept in mind that the water which has to be treated in the TB<sub>m</sub> case was de-ionized before being highly contaminated and thus there will be little risks for precipitation of salts that can contaminate the evaporator and can disturb the solidification of the evaporator bottoms.

As the chemical composition of evaporator bottoms will vary over broad outlines, only empirically developed recipes will be applicable, when this waste category has to be immobilized. Therefore, pre-testings should always be performed in order to prevent failures.

In connection with the solidification of the evaporator bottoms zeolites could be mixed thoroughly into the grout in order to reduce the leaching rate of cesium and strontium .

It should be observed that the evaporator bottoms may contain antifoaming additives which can cause a retardation of the cement hydration and thus to some degree inhibit the solidification.

The decontamination of the water in the condensation pool will be performed basically in the same way as in the AB<sub>m</sub> case (cf. Section 7.1). System 322 can be used to pump the water to the waste plant, where the water will be processed through an ion exchanger. It is easier to clean the water in the TB<sub>m</sub> case, since the volume of contaminated water is significantly smaller than in the AB<sub>m</sub> case.

As in the AB<sub>m</sub> case the cleanup can start about 3 months after the initial accident. The handling and treatment in the waste plant and the number of waste containers produced will be roughly the same as in the AB<sub>m</sub> case.

#### 9.4 Handling and treatment of additional wastes

A precondition when discussing contamination and cleanup in the TB<sub>m</sub> case is that System 322 has not been spraying the primary area of the containment (cf. Section 9.1). This means that there will be almost no contamination of the drywell as only noncondensable gases (the noble gases and possibly minute amounts of organic iodine) will be vented from the condensation pool through drywell. The adsorption of, and radiation from the noncondensable gases one year after the accident will be insignificant, thus, no external contamination of the drywell is expected.

In particular Systems 321, 322 and 323 will contain contaminated water with a much higher content of long-lived radionuclides than during normal operation. Furthermore, surfaces in the condensation pool and the reactor vessel that have been in contact with the water will be highly contaminated.

Part of the cesium sorbed in Systems 321, 322 and 323 and possibly also in the reactor vessel and the condensation pool will be desorbed during the successive cleanup of the water. But a final decontamination of the systems using a chemical decontamination agent are likely necessary due to high residual radiation. This results in spent radioactive decontamination agents of a similar volume as produced when cleaning the same systems in the AB<sub>m</sub> case (cf. Chapter 8).

As a substantial contamination of drywell is not expected (cf. Section 2.2), the generation of additional wastes will be insignificant in the TB<sub>m</sub> case as compared to the waste volumes generated when processing the contaminated water and to the amount of additional wastes produced in the AB<sub>m</sub> case.

## 10. PROPOSALS FOR COMPLEMENTARY ADDITIONS IN THE REACTOR BUILDING AND IN THE WASTE BUILDING

### 10.1 Introduction

After the occurrence of an AB<sub>m</sub> event or a TB<sub>m</sub> event it is of utmost importance that prompt and correct actions are taken, partially to moderate the consequences and partially to set about the restoring work as soon as possible in order to bring the power plant into operation again.

It has been found during the course of the study, that this heavy task can be facilitated, if some preparations have been carried out prior to the occurrence of such an event. These measures are connected with actions that must or should be taken during the period immediately following the event. In other cases, related to later works, such as the cleanup of the highly radioactive water in the reactor containment, planning or preparatory measures are adequate.

Measures discussed in this chapter are mainly related to the AB<sub>m</sub> case, but can also have a bearing on the much less penetrated TB<sub>m</sub> case.

### 10.2 Complementary additions in the reactor building

#### 10.2.1 Controlled drain systems

The handling of the highly contaminated leakage water after an AB<sub>m</sub> event has been thoroughly investigated in Sections 5.5 and 5.6. It was found that this issue can be managed in a reassuring manner without any major problem being evident in the reference reactor Oskarshamn 2. But it would be desirable to have possibilities to take samples both from the controlled area floor drain system (System 345) and the controlled leakage drain system (System 352) in order to pre-check the radioactivity and the composition of the waste prior to directing the water to the liquid waste system (System 342).

If in other reactor plants System 342 cannot receive the leakage water from the reactor building after the accident (cf. Alternatives 2 and 3 in Section 5.5.2), there must be arranged for some opportunity to store the leakage in a safe manner within the reactor building. If the storage capacity of the collecting systems 345 and 352 is unsatisfactory, it should be arranged for a possibility to pump the collected leakage from these systems to the reactor containment. This is the most outstanding and safe space to store this highly contaminated water until it, from the safety point of view, can be pumped to System 342 where decontamination will occur. To make it feasible to transfer the water from Systems 345 and 352 to the reactor containment, the following measures should be taken:

- A complementary addition of a conduit that permits the pumping of collected water from System 345 to the storage tank in System 352.
- A complementary addition of equipment in System 352 that permits the pumping of collected leakage from the storage tank to the reactor containment.

- It should be prepared for the installation of complementary equipment that will collect the controlled floor and liquid drain produced within areas outside the reactor building. By this means no additional water will be conveyed to the reactor building and thus the water balance within the reactor containment will be maintained.

One alternative to the above recommended measures in the reactor building is to introduce supplementary equipment in System 342 or other design modifications in the waste plant (e.g. radiation shields), that permit without any restrictions the acceptance of all leakage water after an accident.

### **10.2.2 Other measures**

Dependent on where the highly contaminated water will be stored and treated, the material in gaskets installed in pumps, valves, motors etc. and the material in cables should be checked with regard to the radiation resistance (Refer to Section 4.4).

To facilitate service on pumps, valves etc. preparatory connections should be installed in order to make it possible to flush conduits and equipment prior to dismantling.

Equipment for radiation measurements should be installed in order to avoid that the ion exchangers in the reactor water cleanup system (System 331) are being overloaded with radionuclides, as an overload could lead to problems handling the spent organic resins. This is especially important in case of a TB<sub>m</sub> event, when the probability of starting the shutdown cooling system (System 321) again is not negligible. The equipment should preferably be installed close to a System 321 pipe having uncleaned reactor water. In this way the control room personnel will get an early warning that there is a high concentration of radioactivity in the reactor water and so a risk for problems as regards to the operating ion exchanger.

## **10.3 Complementary additions in the waste plant**

### **10.3.1 Liquid waste system**

Normally the cleanup steps by means of filtration and ion exchange of the floor drain and the leakage drain are initiated automatically, when the water level in the collection (receiving) tanks have reached a certain level, or there is always a circulation of water between these tanks and the cleanup filters (Refer to Section 5.6.1). If larger parts of System 342 will be contaminated by the highly contaminated leakage water after an accident, it will to an important extent aggravate the installation of supplementary equipment. To avoid this inconvenience the following preparatory measure is proposed for System 342:

- Connections are installed on the collection tanks in System 342 in order to facilitate the installation of radiation measuring instruments, which will be used to shut off the above mentioned cleanup functions in System 342.

As the most part of the highly contaminated water will be decontaminated by means of inorganic sorbents, it is recommended that the installation of the following supplementary equipment is prepared for in System 342:

- Preparatory measures ment for the installation of one ion exchange vessel containing the inorganic sorbents, if it will not be possible to use the existing vessel in Line 1 (cf. Section 7.1). It should be possible to install this vessel after the existing disc filter which will be used to trap mechanical scrap from the incoming water.
- Preparatory measures should be taken that will facilitate the installation of a mechanical filter with the task of trapping "fines" that may have been released from the inorganic sorbents. It should, however, first be investigated if it is possible to use one of the existing filters in Line 3 (cf. Section 7.1).
- Preparatory measures should be taken to make it possible to install radiation measuring equipment before and after the ion exchange vessel containing the inorganic sorbents.
- Preparatory measures should be taken that will allow the installation of conduits which make it possible to return both the highly contaminated water stored in the collecting tanks in System 342 as well as the decontaminated water to the reactor containment.

The aim of this installation is to take care of the leakage water - the reactor containment is the most suitable storage space as mentioned above and as discussed in Section 5.6.2 - and to keep the water balance within the containment. The proposed installation - Line 1 and Line 3 are both connected with Line 7 - could be completed within a week following an AB<sub>m</sub> event and should be carefully planned in order to avoid unnecessary radiation exposure.

- Preparatory connections should be installed close to pumps and valves in order to facilitate the connection of flushing water, which will be needed prior to dismantling or service of the components.

It has not seemed to be realistic to propose preparations that will make it possible to increase the cleanup capacity in System 342 as the capacity of the subsequent solidification in System 343 will decide the time table for the first part of the cleanup procedure of the power plant.

### **10.3.2 Solidification system and mould handling**

In the existing solidification system belonging to the reference reactor the different waste categories are solidified with cement into prefabricated concrete moulds. Owing to the large amount of waste that will be generated and thus has to be solidified in connection with the cleanup procedure of the water, the following preparatory measures are proposed for System 343 in order to speed up the solidification capacity:

- To work out the basis for handling of prefabricated concrete lids which shall be placed on top of moulds with a surface dose rate of 300 mSv/h.
- One hole should be made in the ceiling above the solidification cell to facilitate the transportation of empty moulds into the cell.

A further increase of the solidification capacity in addition to what will be available when the proposed measures will have been accomplished, can be obtained by connecting a remotely operated mobile solidification system to the existing permanent system. Concrete blocks have to be placed around the mobile facility in order to obtain the required radiation shielding.

As the immobilization of the waste will start not earlier than 3 to 6 months after the accident occurred, there should be a fair chance to obtain and install this additional equipment. By using also a mobile system, the solidification capacity could be increased by about 65 %.

Concerning the handling of moulds with a surface dose rate of 300 mSv/h in the existing solidification system an exhaustive study has been performed. The examination of the handling, the transportation and the on-site storage has resulted in the following proposals:

- The electric lift in the solidification cell should be supplemented with a hydraulic one. A winch for moving the moulds to the lifting position would also be desirable.
- The overhead crane used for lifting moulds should be provided with a remotely operated closed-circuit television camera in order to safely transfer the filled moulds from the solidification cell to the unloading ramp.
- The doors at the unloading gate should be remotely operated and thus be provided with adequate equipments.
- The nuclide specific gamma spectroscopic measurement should be performed on a place where the back-scattering will be sufficiently low. This can probably not be arranged for in the on-site storage, where moulds with high surface dose rates will be stored. A better location seems to be the lifting position in the solidification cell.
- An existing hole for the overhead crane leg in a radiation shield within the on-site storage should probably be closed in order to reduce the dose to the operator of the overhead crane. Additional radiation shielding may probably be necessary in order to protect the operator.

## 11. DISCUSSION AND CONCLUSIONS

This investigation has shown that an accident where 10 % of the core inventory of noble gases is released, but no serious damages are caused to the core structure or fuel, is able to stop the operation of the reactor for several years. If re-insulation of the reactor pressure vessel is impossible, the consequence will even be a close down of the reactor. Processing of the contaminated water seems to be fairly straightforward with the existing liquid waste system - after what is estimated to be minor reconstruction work - and the major cleanup of contaminated water could be finished about two years after an  $AB_m$  event in the reference reactor Oskarshamn 2. However, the damage to particularly the electrical equipment by water corrosion when the reactor containment is filled with cooling water, and the probable damage to both mechanical and electrical equipment during surface decontamination of the reactor vessel and containment will prolong the shutdown of the reactor.

The first waste problem encountered after an accident is the management of highly radioactive leakage water (leakage drain or floor drain) from the reactor containment. The cleanup systems in the waste plant are not available at the beginning, because of the need for preparations and any necessary modifications in the waste plant before starting the major cleanup. Therefore, the leakage water should be directed to the containment, which is a most suitable and safe storage space. In the reference reactor, this is possible after making certain pipe connections in the waste plant, something which can be completed within a week following the accident. It is also possible in the reference reactor to store the leakage water in the waste plant and clean it after a couple of weeks using organic resins as usual.

These favourable conditions are, however, reactor specific and investigations should be performed for each individual reactor. Other options are to modify the waste plant so that it can receive and treat the assumed leakage water or to make additions in the leakage drain and floor drain systems in the reactor building. Either option should be accomplished as soon as possible.

The fission products that need to be taken care of are mainly cesium-134 and cesium-137 and to some extent strontium-90. Cesium is the largest radiation source in the containment water and on contaminated surfaces. The total amount of cesium and strontium that needs to be taken up by ion exchangers and solidified in the waste plant is  $2.7 \cdot 10^4$  TBq and  $1.5 \cdot 10^3$  TBq, respectively. These values are significantly higher than the radioactive inventories used in the safety analysis for the Final Repository for Reactor Waste (SFR) at the Forsmark site, to which the waste most probably will be transported. Consequently, an additional safety analysis and a new licensing procedure will be required before transportation is permitted. In practice, however, there should be no problem to accept the waste and only administrative measures are expected to be necessary.

Due to the high radiation doses expected to the ion exchange resins in the cleanup system, the use of zeolites should seriously be considered. Specific zeolites have an excellent selectivity of especially cesium and a profound radiation resistance as compared to organic resins. At present the use of granulated zeolites has not been tested in the cleanup system, but no serious problems are foreseen. Instead of using zeolites in the existing ion exchange vessels, special once-for-all filters with zeolites could be used, which once loaded are directly solidified in cement. This method has not been studied in any detail and should be investigated separately.

The major cleanup step is supposed to start at the earliest three months after the accident. The existing cleanup and solidification systems should be used as much as possible. In the reference reactor, the cleanup system should be the liquid waste system, mainly Line 1, designed to treat leakage drain water during normal reactor operation. Preparatory measures should be considered to enable the connection of special filter vessels for inorganic sorbents such as zeolites in case the ordinary vessels are unsuitable or not available for some reason. Other preparatory measures should also be considered such as to install radiation measuring equipment and to make connections for flushing water close to pumps and valves.

After cleanup the treated water is returned to the reactor containment in order to keep the water balance and to minimize the increase of the total contaminated water volume. For this purpose a minor pipe connection has to be made in the waste plant.

Cement should be used as the solidification matrix because cement is radiation resistant, zeolites are well integrated into a cement matrix and cement is the solidification matrix normally used in the solidification plant, hence the existing equipment can be utilized. However, with the use of special once-for-all zeolite filters the cement casting has to be modified.

In the early stages of the cleanup process it is the solidification capacity of the solid waste system that will be limiting, while the uptake of nuclides on the ion exchanger will be the limiting step at the very end of the decontamination of the containment water. Thus, at the first months of the cleanup, extra solidification capacity would speed up the restoration, for example if a mobile solidification unit is also used. Moreover, the use of prefabricated lids to the moulds is feasible and should be realized.

The dose rate in the local control room of the waste plant will be well below 10  $\mu\text{Sv/h}$ . Thus, although highly contaminated water and spent sorbents are temporarily stored in the receiving tanks, the thick concrete walls and floor, and the distances in the waste plant, keep the radiation level low enough to give complete access to the local control room.

The handling of concrete moulds with a high surface dose rate - 300 mSv/h has been assumed in this study - has been verified to be feasible in the reference reactor. However, the introduction of equipment that allows remote control would facilitate many of the operations and also reduce the occupational radiation exposure. Furthermore, additional radiation shielding is probably necessary.

While the decontamination of the waste water seems to be a fairly straightforward process, the surface decontamination of the reactor vessel and containment, and equipment therein, would be much more difficult. Methods for decontaminating nuclear facilities are known, but few of these can be used on, for example, concrete surfaces or external parts of equipment without causing damage or at least requiring major reconstruction work of the reactor after the decontamination. A possible sequence for the surface decontamination could be 1) water spraying (using low-pressure water), 2) water jetting (hydro-lasing), 3) a weak decontamination process and finally if necessary (e.g. on concrete surfaces) some kind of grinding technique. Internal parts of pipes, pumps, etc. can be cleaned by closed-circuit pumping of either a weak or a strong decontamination solution.

The contamination of concrete surfaces in the containment has been estimated to 7 GBq/m<sup>2</sup> of cesium two years after the accident when the first entry to the containment is supposed to take place. This corresponds to a general dose rate in the containment of around 30 mSv/h. The dose rate very close to surfaces will be higher. Because of the rather high dose rate the initial decontamination steps need to be done using remotely controlled vehicles and equipment.

The restoration of the reactor will not only be delayed by the surface decontamination and the removal of all remaining mineral wool insulation. The water filling of the containment means that at least all electrical equipment including cables, and probably a lot of the mechanical equipment, must be either renovated or replaced in order to comply with the quality assurance requirements. This would be a time consuming and expensive process. Using decontamination methods causing as little damage as possible to the buildings and equipment will result in a longer cleanup time, probably years longer.

As a main conclusion of this study it appears that a core damage corresponding to a release of 10 % of the noble gas inventory in the fuel, as described in the AB<sub>m</sub> case, can be managed in the reference reactor Oskarshamn 2. Only minor modifications are required in the waste plant after the event.

The aim of the study has been to be as general as possible. Many of the findings and recommendations are generally applicable. One issue, for example, is the influence of radiation doses on the tightness of components and the possible leakage due to a break down of sealings. This question has not been fully studied in the project. Another issue is the importance of not operating the ion exchanger in the reactor water cleanup system after the event in order not to expose the ion exchange resin to such a high radiation dose that it will be difficult to handle. Individual characteristics at different nuclear power plants, however, make the detailed results of the study reactor specific.

As regards the TB<sub>m</sub> case, this has not been analysed in detail. The contaminated water is confined to the reactor vessel and hence, has a much higher concentration of radionuclides. This can be dealt with either by evaporation or by dilution and subsequent cleaning as in the AB<sub>m</sub> case. Therefore, the same conclusion is valid. In the reference reactor it is possible to manage the radioactive waste after a TB<sub>m</sub> event. Furthermore and in general, the possibility of taking the reactor into operation again seems more likely in the TB<sub>m</sub> case, as the contamination of the containment will be much smaller.

**12. ACKNOWLEDGEMENTS**

The valuable assistance of Stefan Danielsson and his staff at the Oskarshamn site and the skilful typing of Ms Ulla Hammar is gratefully acknowledged.

## 13. REFERENCES

- AKA 76. Spent Nuclear Fuel and Radioactive Waste, Report from the AKA Study, SOU 1976:30 and SOU 1976:31, Stockholm 1976 (In Swedish).
- ALL 84. Allard B., Olofsson U. and Torstenfelt B., "Environmental Actinide Chemistry", Inorg. Chim. Acta, 94, (1984) 205-221.
- ANS 88. ANS Winter Meeting, Washington, D.C., Oct 30 - Nov 4, 1988. Topical Meeting on the TMI-2 Accident: Materials Behavior and Recovery Technology, Trans ANS, Vol. 57, 1988.
- ATK 88. Atkinson A. and Nickerson A.K., "Diffusion and Sorption of Cesium, Strontium, and Iodine in Water-Saturated Cement", Nucl. Techn. 81 (1988) 100-113.
- BAS 86. Baston V.F. and Hofstetter K.J., "Adherent Activity on Internal Surfaces " in Toth, Malinauskas, Eidam, and Burton (Eds.), The Three Mile Island Accident, Diagnosis and Prognosis, ACS Symp. Ser. 293. Washington, D.C. 1986, 124-145.
- BEA 87. Beahm E.C., "Tellurium Behavior in Containment Under Light Water Reactor Accident Conditions", Nuclear Technology, 78, (1987) 295-302.
- BRE 79. Breck D.W., Zeolite, Molecular Sieve-Structure, Chemistry and Use, Union Carbide Corp., New York 1974.
- COL 88. Collins J.L., Osborne M.F., Lorenz R.A. and A.P. Malinauskas, "Fission Product Iodine and Cesium Release Behaviour Under Light Water Reactor Accident Conditions", Nuclear Technology, 81 (1988) 78-94.
- DEC 85. "Decontamination of Nuclear Facilities to Permit Operation, Inspection, Maintenance, Modification or Plant Decommissioning", Technical Reports Series No. 249, IAEA, Vienna 1985.
- ELK 85. Elkert J. and Ullberg M., Management of Radioactive Waste Resulting from Nuclear Fuel Damage, Nordic Liaison Committee for Atomic Energy, Report SKI 85:2, Swedish Nuclear Power Inspectorate, Stockholm, Dec. 1985.
- FOL 86. Foltman A.J. (Ed.), Proceedings of the Workshop on Requirements of Mobile Teleoperators For Radiological Emergency Response and Recovery, ANL/EES-TM-291, DE87 013859, Argonne Nat. Lab., Argonne, Feb. 1986.
- FOR 86. Forsyth R.S. and Werme L.O., "The Corrosion of Spent Fuel in Synthetic Groundwater", in L.O. Werme (Ed.), Scientific Basis for Nuclear Waste Management IX, MRS, Pittsburg 1986, pp. 327-336.
- HUE 86. In Huebner M.F. (Ed.), Proc. of ANS Meeting on Fission Product Behavior and Source Term Research, EPRI-NP-4113-SR, ANS, July 1985.
- ICR 30. Annals of the ICRP, "Limits for Intakes of Radionuclides by Workers", ICRP Publication 30, Pergamon Press, Oxford.

- JOH 85. Johansson K. (Ed.), RAMA (Reactor Accident Mitigation Analysis), Final Report, RAMA-85/2, Jan. 1985.
- GOE 85. Goel V.S. (Ed.), Proceedings of the International Conference on Nuclear Power Plant Aging, Availability Factor and Reliability Analysis, Am. Soc. of Metals, Metals Park 1985.
- KAT 87. Katonak L.E., Hofstetter K.J., "Microorganisms, Filters and Waterclarity at TMI-2", Waste Management -87, Volume 2, ed. Post, Wacks, Proceedings of the Symposium on Waste Management at Tucson, Arizona, March 1-5, 1987, Page 445, Part 2.
- KOM 83. Komarneni S. and Roy D.M., "Hydrothermal Inter-actions of Cement or Mortar with Zeolites or Montmorillonites", in D.G. Brookins (Ed.), Scientific Basis for Nuclear Waste Managements VI, Vol. 15, Elsevier, New York 1983, pp. 55-62.
- LAH 80. Lahti G.P. et al, Designing for Postaccident Radiological Conditions, NSAC-17, EPRI, Palo Alto, Dec. 1980.
- LUN 75. Lundgren K., "CYLGAX, CYLGAM and GAMEN - Computer codes for gamma transport calculations around cylindrical source distributions", ABB Atom Technical Report, RF 75-194, Västerås, May 1975 (In Swedish).
- MAL 79. Malinauskas A.P., Lorenz R.A. and Collins J.L., "Fission Product Release During LWR Loss-of-Coolant Accidents" in "Radiation Source Terms and Fission Product Release", Trans. Am. Nucl. Soc., 32, June 1979.
- NUR 81. NUREG-0772, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, June 1981.
- PIL 82. Pillary K.K.S. and Palau G.L., "Radiolytic Effects on Ion Exchangers during the Storage of Radioactive Waste", International Conference on Radioactive Waste Management, 12-15 Sept., Winnipeg 1982.
- STO 79. Stone J.A., "Studies of Concrete as a Host for Savannah River Plant Radioactive Waste", in G.J. McCarthy (Ed.), Scientific Basis for Nuclear Waste Management, Vol 1, Plenum Press, New York 1979, pp. 443-452.
- TAR 87. Tarcza G.A. (Ed.) Proceedings of the 1987 International Decommissioning Symposium, Westinghouse Hanford Co., Richland 1987.
- TOR 89. Torstenfelt B. and Hedin G., "Leaching of Cesium from a Cement Matrix" in W. Lutze and R.C. Ewing (Eds), Scientific Basis for Nuclear Waste Management XII, Vol. 127, MRS, Pittsburgh 1989, pp. 495-500.
- TOT 86. Toth L.M., Malinauskas A.P., Eidam G.R., and Burton H.M. (Eds.), The Three Mile Island Accident, Diagnosis and Prognosis, ACS Symp. Ser. 293, Am. Chem. Soc., Washington, D.C. 1986 (and references therein).
- VOO 72. Van de Voorde M.H. and Restat C., "Selection guide to organic materials for nuclear engineering", CERN 72-7, Genève, May 1972.

**TECHNICAL REPORTS (Project Reports)**

- TR1. Torstenfelt B.  
SKI - Determination of the source term related to an assumed, larger core damage ( $AB_m$  or  $TB_m$  event) in a BWR power plant (In Swedish).  
Report RVC 88-12, Rev 1, February 1990.
- TR2. Jansson R. and Elkert J.  
SKI-project - Radiation levels in the reactor building at different times after an assumed, larger core damage in a BWR power plant (In Swedish).  
Report RM 88-1010, September 1988.
- TR3. Pettersson C.-B.  
SKI-project - Radiation levels in the waste plant and the reactor building at different times after an assumed, larger core damage in a BWR power plant (In Swedish).  
Report RM 88-1129, May 1989.
- TR4. Dolck P.-A. and Eriksen L.  
SKI-project - Measures in the reactor building during the period immediately following an assumed  $AB_m$  or  $TB_m$  event. Filling of the reactor containment, sampling, management of leakage water and influence on the systems (In Swedish).  
Report RVB 88-298, August 1988.
- TR5. Lennartsson R. and Christensen H.  
Processing of released activity. Treatment of the aqueous phase by means of organic and inorganic sorbents.  
Report RVC 88-111, December 1988.
- TR6. Åkerblom B. and Christensen H.  
SKI-project - Waste management related to an assumed, larger core damage in a BWR power plant. Cleanup of the water in the reactor containment (In Swedish).  
Report RVB 88-296, June 1989.
- TR7. Pettersson C.-B.  
SKI-project - Handling, transport and storage of waste moulds after an assumed, larger core damage in a BWR power plant (In Swedish).  
Report RM 88-1130, Rev 1, January 1989.
- TR8. Torstenfelt B.  
SKI - Production of additional waste related to an assumed, larger core damage ( $AB_m$  or  $TB_m$  event) in a BWR power plant (In Swedish).  
Report RVC 88-99, Rev 1, February 1990.
- TR9. Torstenfelt B.  
Waste form properties and the final disposal.  
Report RVC 88-110, November 1988.

**APPENDIX 1****LIST OF SYSTEMS MENTIONED IN THIS REPORT**Primary process systems

- 311 Steam lines
- 312 Feedwater system
- 313 Recirculation system
- 314 Relief system
- 316 Condensation system

Secondary process systems

- 321 Shutdown cooling system
- 322 Containment vessel spray system
- 323 Core spray system (Low pressure coolant injection system)
- 324 Pool water cooling and cleanup system
- 327 Auxiliary feedwater system

Primary system water cleanup systems

- 331 Reactor water cleanup system
- 332 Condensate cleanup system (with precoat filters)
- 337 Accident sampling system

Waste systems

- 341 Offgas system
- 342 Liquid waste system
- 343 Solid waste system
- 344 Cleaning (decontamination) system
- 345 Controlled area floor drain system

Other reactor process systems

- 351 Boron system
- 352 Controlled leakage drain system

Water treatment and distribution systems

- 731 Raw water treatment system
- 732 Water demineralization system
- 733 Demineralized water distribution system

Air treatment systems

- 741 Containment vessel gas treatment system
- 742 Ventilation system for restricted areas

Miscellaneous systems

- 821 Sampling system
- 861 Fire water system

Some of the process systems and water cleanup systems are seen on the main flow sheet, Figure A2.1.

**APPENDIX 2****EXISTING CLEANUP SYSTEMS**

This appendix gives a brief description of the cleanup systems of interest - including the liquid system in the waste plant - designed for the conditions during normal reactor operation.

**System 331, the reactor water cleanup system**

The reactor water is purified in deep bed ion exchangers with radial flow (RWCU in Figure A2.1). The normal cleanup flow with one of the two ion exchangers in operation corresponds to 2 % of the feedwater flow at full reactor power and is accomplished by one of the pumps in System 321. The impurities and radioactive products are retained using bead resins. The spent resins are transported hydraulically to the waste plant for solidification in concrete moulds.

The high temperature mechanical filters (HTF) located prior to the ion exchangers were originally intended for cleaning the water from suspended impurities. As mechanical filtering was found to be unnecessary, these filters were taken out of service after one year of operation.

**System 332, the condensate cleanup system**

The condensate is purified in a full flow precoat filter system of ABB Atom design (CCU in Figure A2.1). The system consists of six filter units in parallel. Each unit has filter cartridges coated with powdered ion exchange resin. If the conductivity in the water downstream from a filter is greater than 20  $\mu\text{S}/\text{m}$  the filter must be recoated with fresh resin.

**System 324, the pool water cooling and cleanup system**

The system includes a precoat filter of the same type as used in System 332.

**System 342, the liquid waste system**

The system receives and treats drainage water, flushing and wash water, and spent resins from the reactor units Oskarshamn 1 and 2. The system consists of seven lines, corresponding to the different waste categories and having a varying radioactivity except for Line 6.

Line 1 - Pure drainage water having a low conductivity. Figure A2.2.

Drainage water from System 352 is collected in a receiving tank. The water is cleaned in a disc filter coated with powdered ion exchange resin and a filtering agent. A final cleaning is carried out in a deep bed ion exchanger filled with mixed bead resin. The spent resins and filtering agents are transferred to Line 4. Most of the cleaned water is returned to the process systems.

- Line 2 - Dirty water from floor drainage gutters, showers and the laundry, having a high conductivity. Figure A2.3.
- Water from System 344 and 345 is collected in a receiving tank and then pumped to Line 3.
- Line 3 - Collection of water from Lines 2 and 4. Figure A2.3.
- Generally, water is pumped further to Line 6. If the concentration of radionuclides is too high to permit release to the recipient, the water is purified in a disc filter. An option is to transfer the water to Line 5.
- Line 4 - Reception and collection of spent powdered and bead resins and filter agents.
- Depending on radioactivity concentration the waste is either solidified with cement into concrete moulds (System 343) or de-watered and stored in concrete tanks.
- Line 5 - Water intended for evaporation and decontamination liquids.
- The line receives from Line 3 water that is not possible to filtrate and from System 344 some of the decontamination liquids. The evaporator bottoms are transferred to System 343.
- Line 6 - Water intended for release to the recipient
- The water has such a quality and insignificant radioactivity that it can be released. During the pumping out a proportional sample is taken and stored for later analyses.
- Line 7 - Water from the reactor pools (handling of reactor internals) and the condensation pool.
- During reactor operation water from the condensation pool is purified in a disc filter. During reactor outages for refuelling, water in the above mentioned pools are stored and purified for later reuse. One example is draining of the pool above the reactor in preparation for removal of the reactor vessel head.

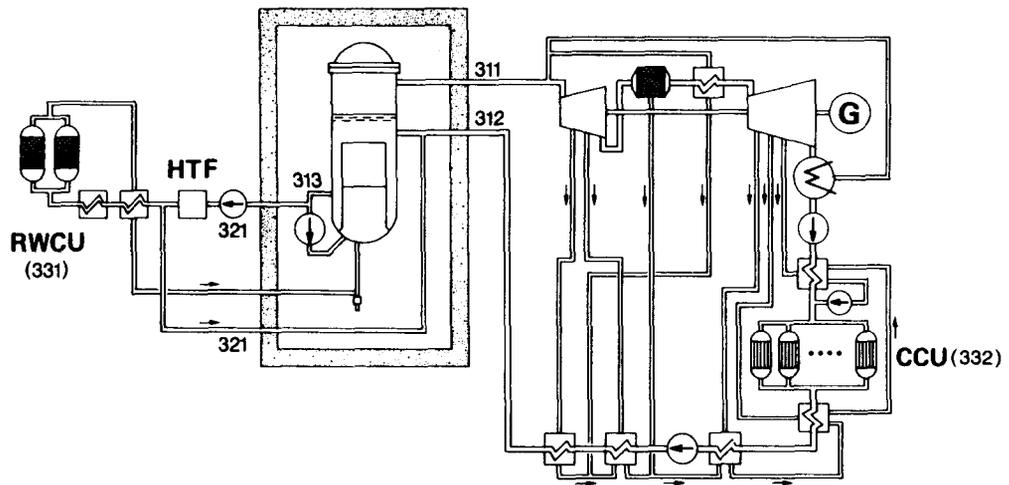


Figure A2.1. Main flow sheet of Oskarshamn 2.

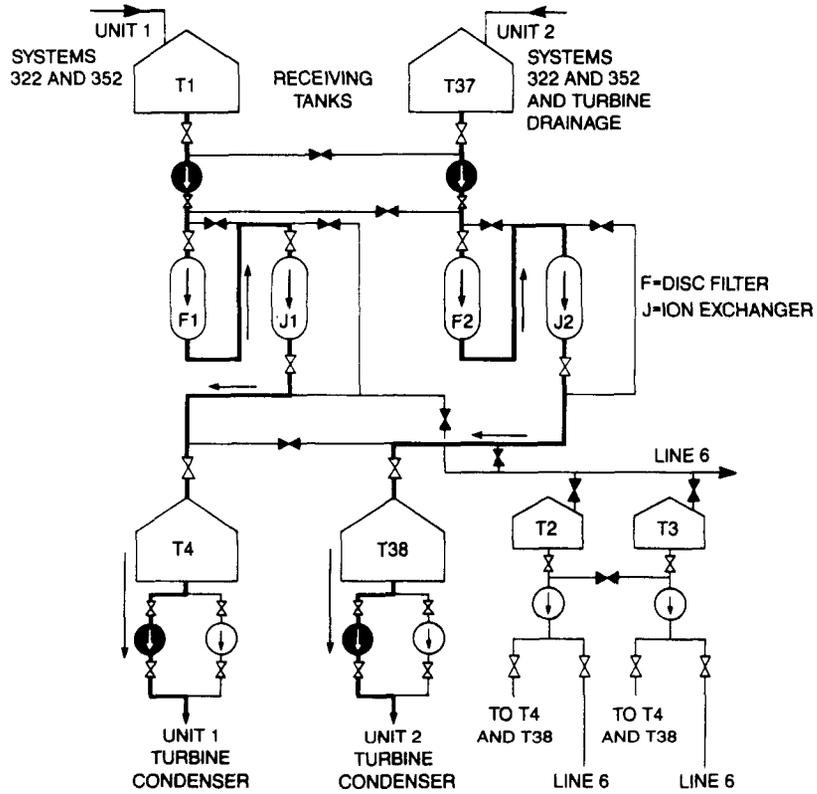


Figure A2.2. Simplified flow sheet of Line 1 in the waste plant of Oskarshamn 1 and 2.

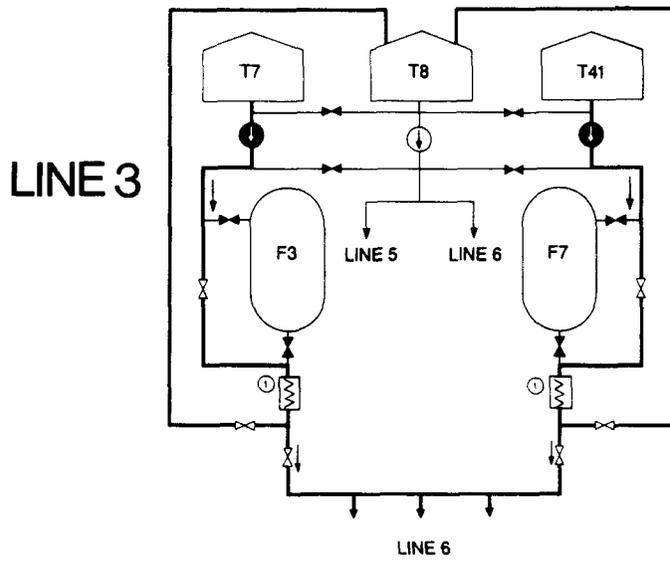
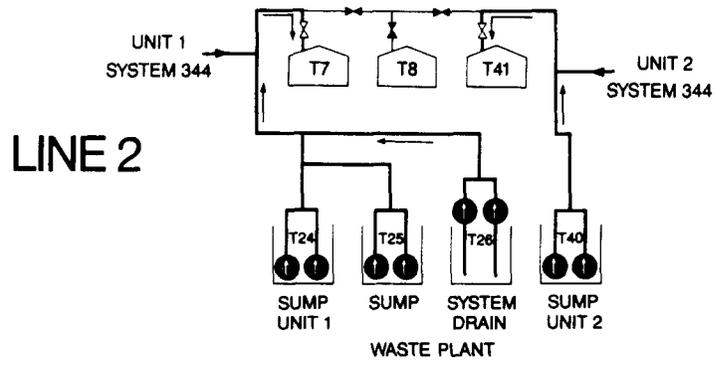


Figure A2.3. Simplified flow sheet of Lines 2 and 3 in the waste plant of Oskarshamn 1 and 2.

**APPENDIX 3****SOME BUILDING LAYOUTS**

This appendix contains some selected building layouts to visualize parts of the reactor building and most of the waste building of the reference reactor Oskarshamn 2.

Contents:

**Reactor building**

- Figure A3.1, level +87.3 Bottom level including Systems 322, 323, 345 and 352.
- Figure A3.2, level +111.0. Level including Systems 321, 331 and 337
- Figure A3.3, vertical section Showing the reactor containment and the reactor and fuel storage pools.

**Waste building**

- Figure A3.4, level +102.5 Level including the solidification cell and pump rooms.
- Figure A3.5, level +106.0 Level including main parts of System 342 and the mould handling area.
- Figure A3.6, level +109.0 Level including the main control room in addition to above mentioned for level +106.0.
- Figure A3.7, level +112.0 Level including a second control room and service areas.

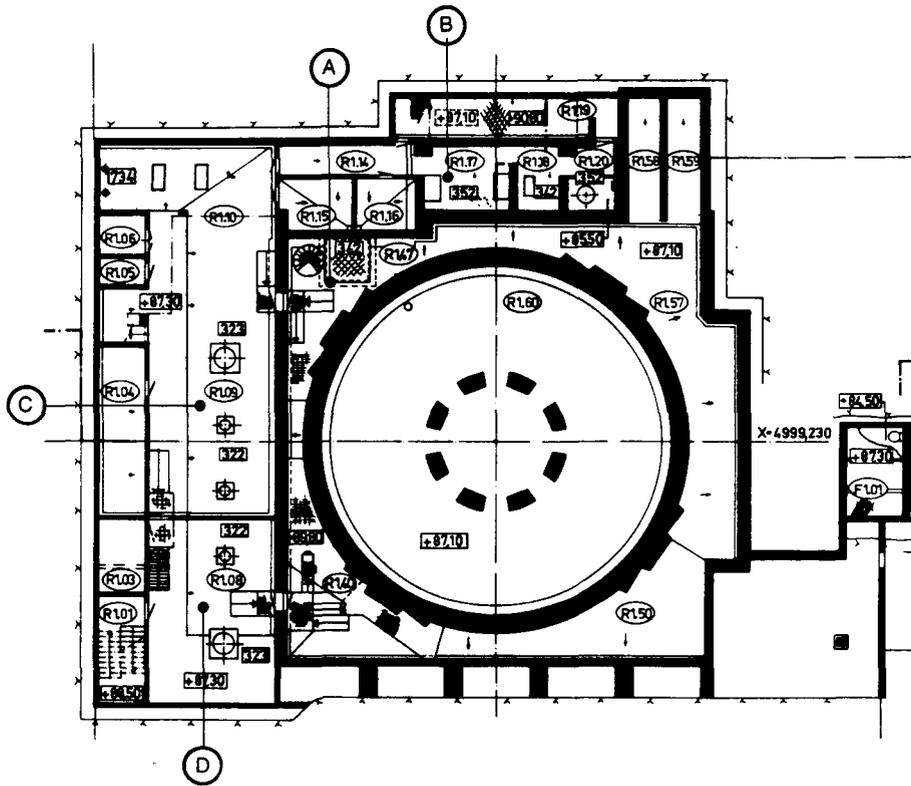


Figure A3.1. Oskarshamn 2 - Reactor building, level +87.3.

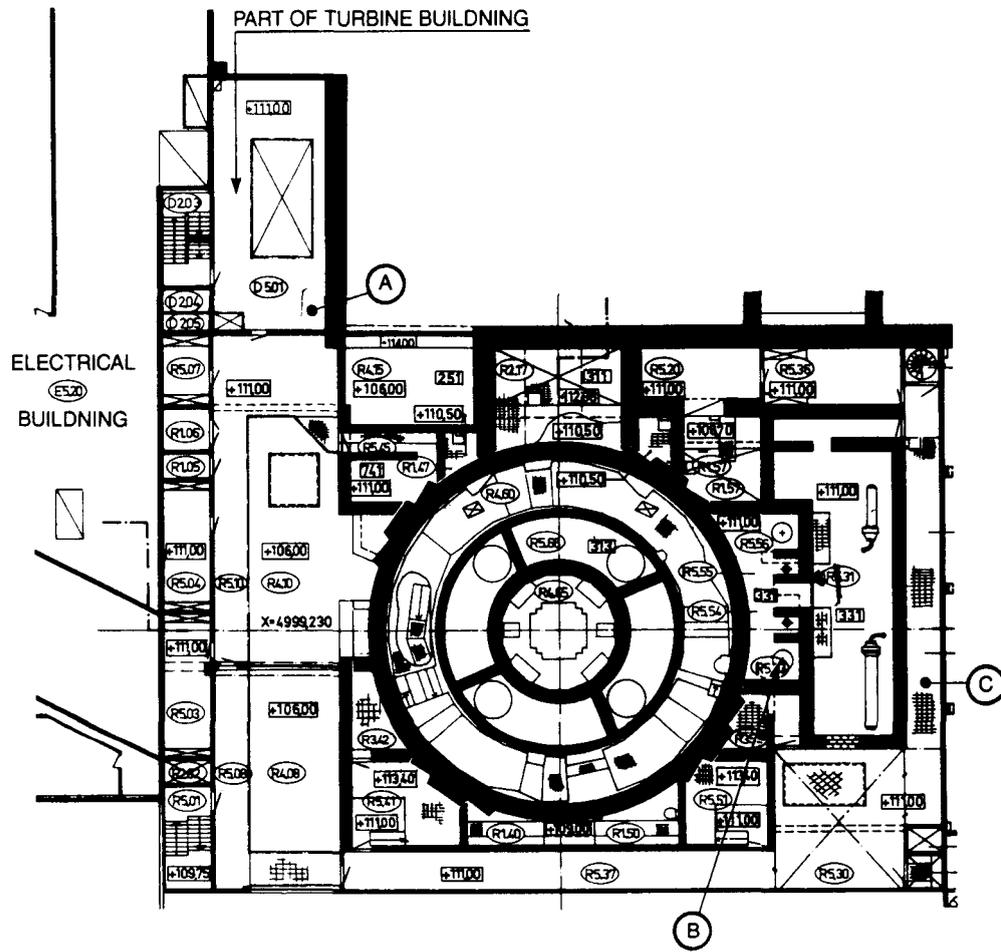


Figure A3.2. Oskarshamn 2 - Reactor building, level +111.0.

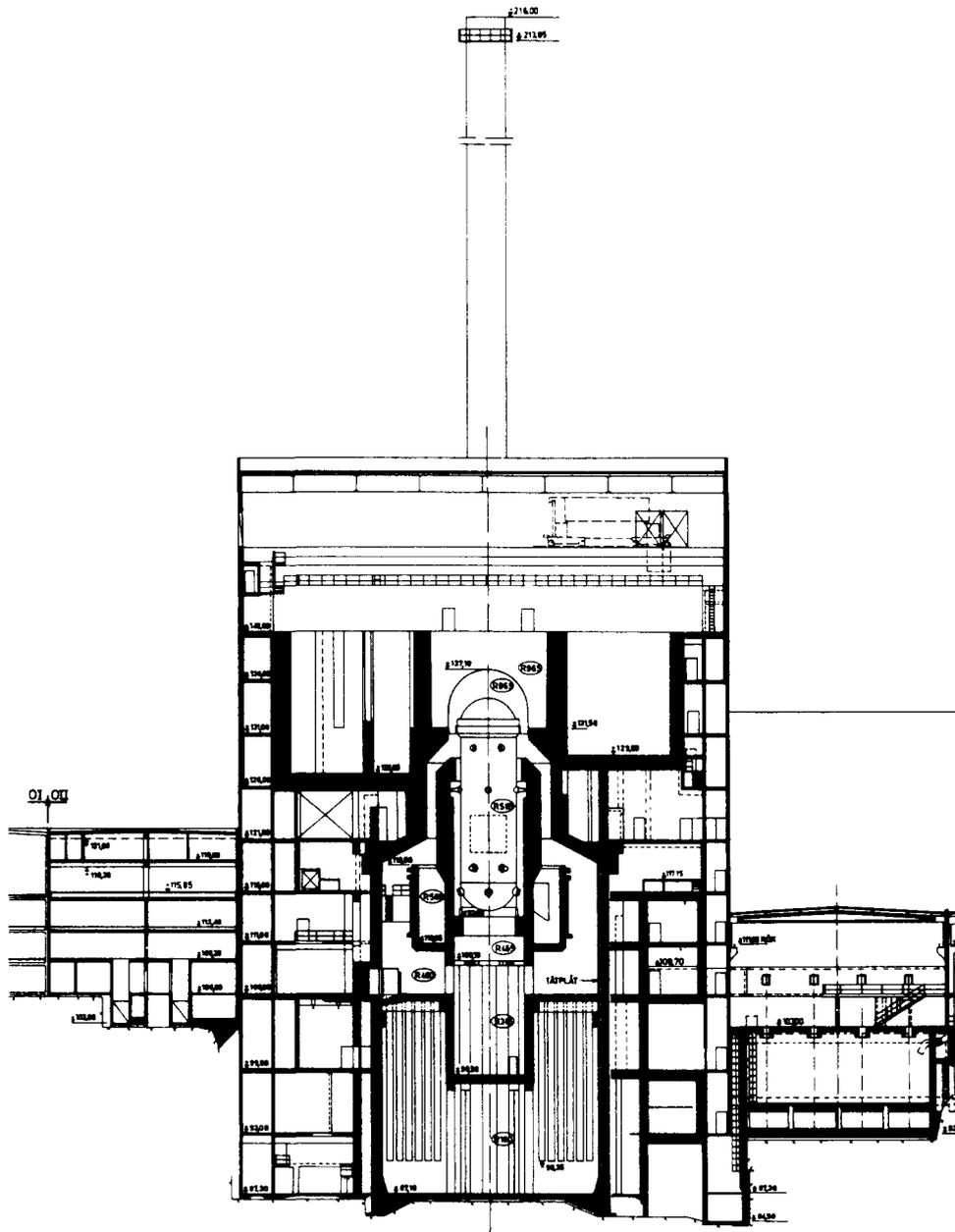


Figure A3.3. Oskarshamn 2 - Reactor building, vertical section.

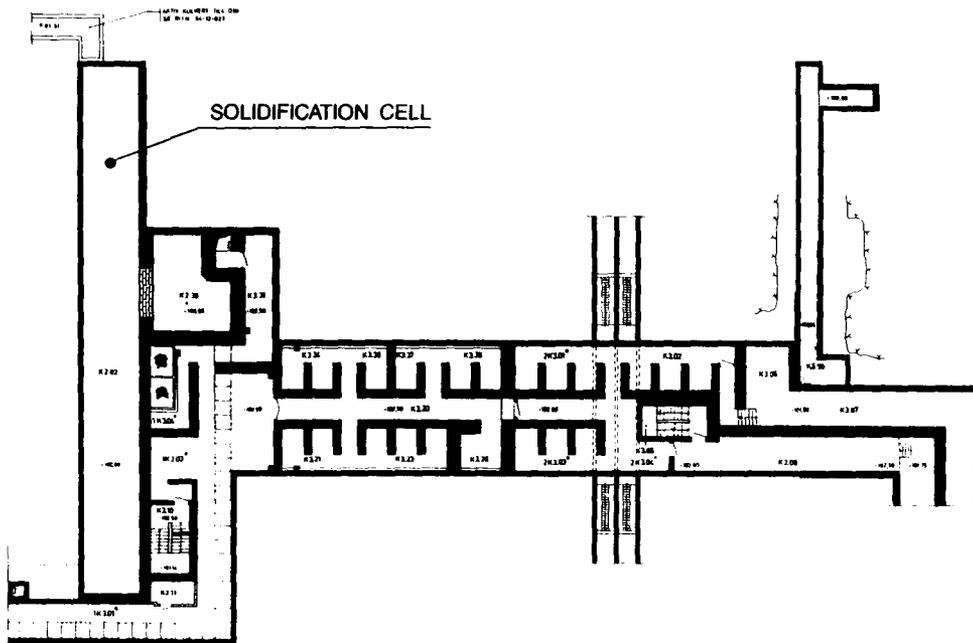


Figure A3.4. Oskarshamn 1 and 2 - Waste building, level +102.5.

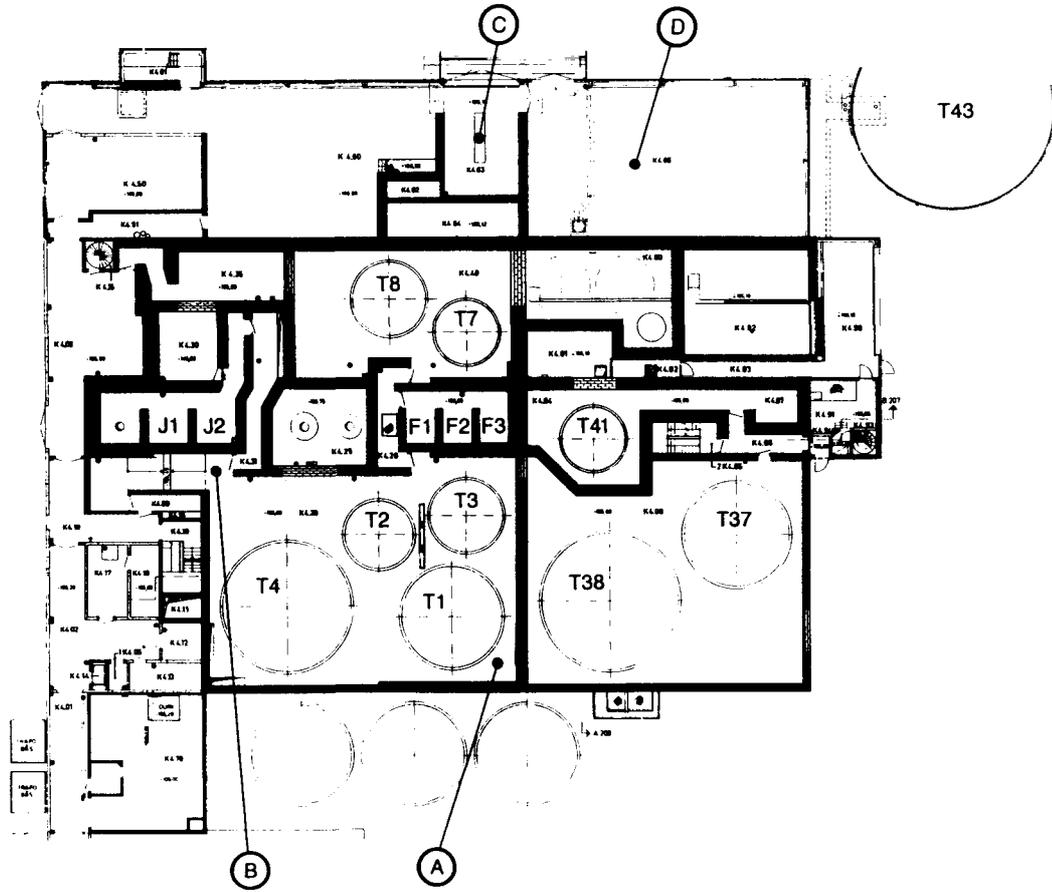


Figure A3.5. Oskarshamn 1 and 2 - Waste building, level +106.0.

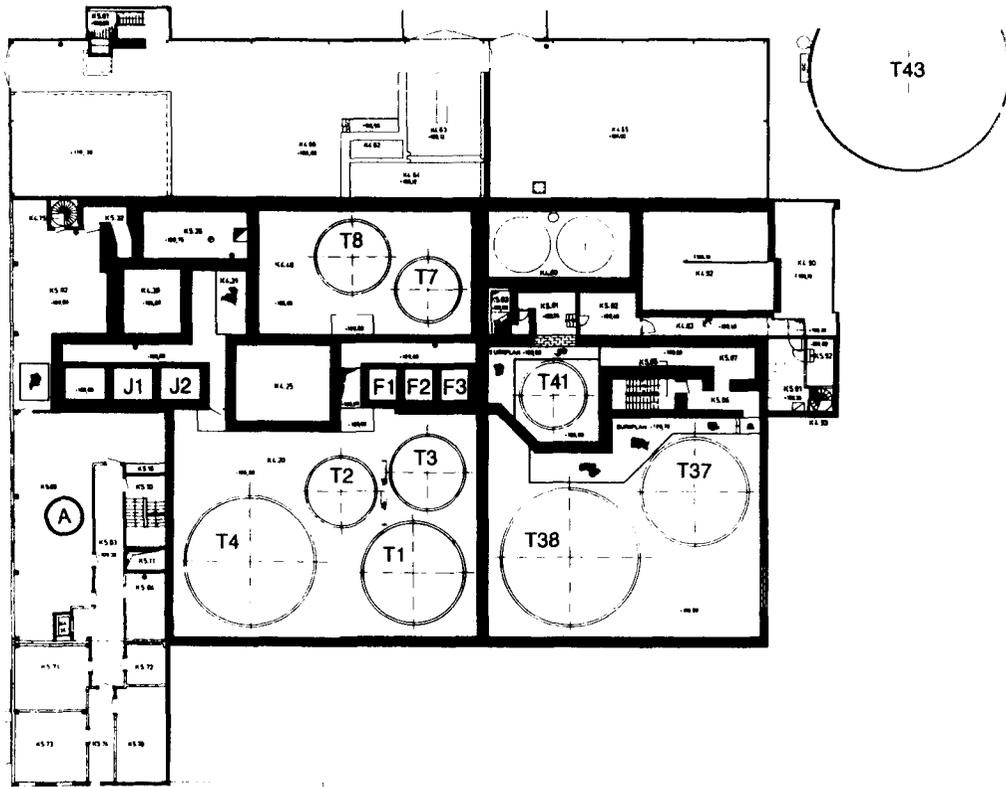


Figure A3.6. Oskarshamn 1 and 2 - Waste building, level +109.0.

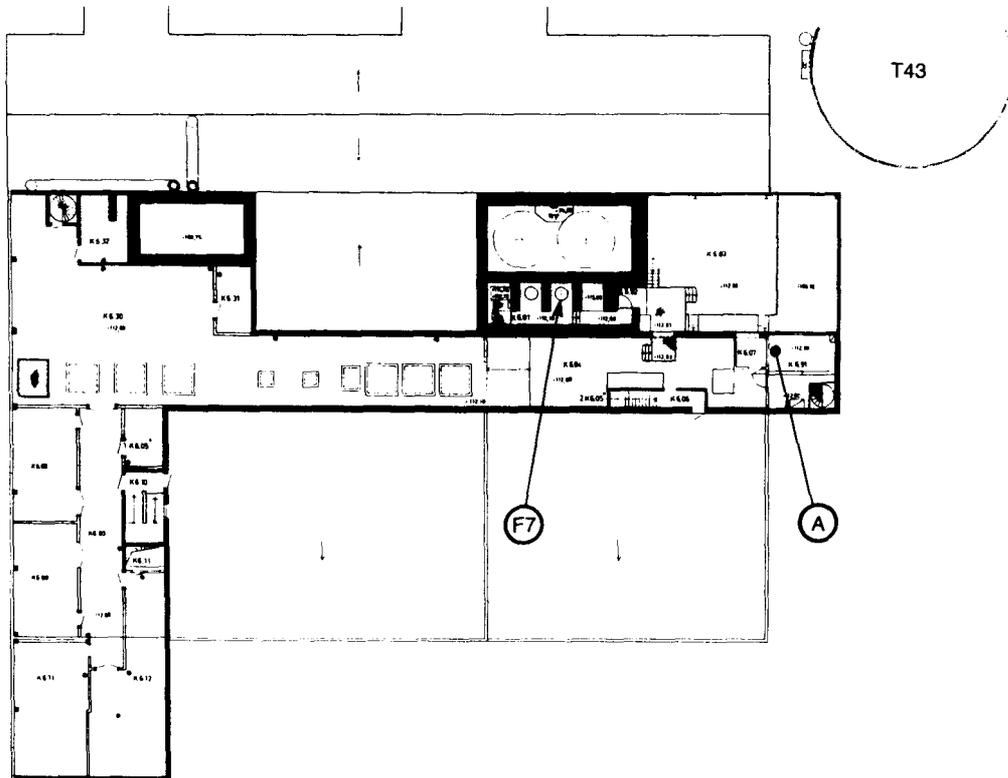


Figure A3.7. Oskarshamn 1 and 2 - Waste building, level +112.0.

## APPENDIX 4

## DOSES TO THE ENVIRONMENT AFTER RELEASES FROM THE REACTOR CONTAINMENT

It is assumed that the reactor containment atmosphere is pressure relieved by means of the Multi Venturi Scrubber System (MVSS), which retains most of the radioactive substances except for noble gases and organic iodine (methyl iodide). According to Chapter 3 maximum 0.1 % of the iodine released exists in form of organic iodine. The release from the containment of remaining fission products in the atmosphere (in aerosol form) is small and more than 99 % is retained in MVSS. Consequently, it can be expected that only noble gases and organic iodine will contribute to the environmental dose.

The effective dose equivalent to the most exposed individual members of the public has been calculated using dose conversion factors (mSv per Bq emitted) for accidents from the Final Safety Analysis Report for Oskarshamn 3. The estimated dose for two release moments of interest -one day and one week - is based on the following assumptions:

- release during 24 hours
- stack height 20 m
- distance to most exposed individuals 2 km
- deposition velocity for organic iodine  $5 \cdot 10^{-4}$  m/s
- inhalation rate  $2.5 \cdot 10^{-4}$  m<sup>3</sup>/s.

As regards weather conditions, the relative concentration used corresponds to a cumulative frequency of 5 %, i.e. the dose is as calculated or less for 95 % of the cases. This gives a relative concentration of  $2.5 \cdot 10^{-5}$  s/m<sup>3</sup> based on weather statistics.

The doses obtained are given below. The noble gas dose is entirely from the cloud. The iodine dose is obtained predominantly from the ground deposit, while the cloud contribution is small.

Effective dose equivalent (mSv)

	<u>One day</u>	<u>One week</u>
Noble gases	10	1.8
Organic iodine	0.3	0.1
Total	----- 10	----- 2

The dose to the most exposed individuals will be much smaller if average weather conditions are considered, i.e. the cumulative frequency is 50 % instead of 5 %. The combination of a much lower relative concentration and a doubled shielding factor results in doses that are a factor 300 lower than those presented above for the extreme weather.

#### **THE KAV STEERING COMMITTEE**

Knud Brodersen	Risø, Danmark
Gordon Christensen	IFE, Norway
Franz Marcus	NKA
Sören Norrby	SKI, Sweden
Veijo Ryhänen	TVO, Finland
Pekka Silvennoinen	VTT, Finland

#### **THE KAV PROGRAMME COORDINATOR**

Leif Moberg	SSI, Sweden
-------------	-------------

#### **THE PROJECT GROUP FOR KAV 390 HAS INCLUDED**

Jan-Erik Andersson*	SKI, Sweden
Ragnar Boge	SSI, Sweden
Ingmar Eriksson	SKI, Sweden
Bengt Löwendahl	OKG, Sweden
Bertil Mandahl	OKG, Sweden
Sören Norrby	SKI, Sweden
Ulf Ottersen	IFE, Norway
Lauri Pöllänen	TVO, Finland
Jörgen Ransmark	Sydskraft, Sweden
Björn Thorlaksen	Inspectorate of Nuclear Installations, Denmark

\* Project Leader