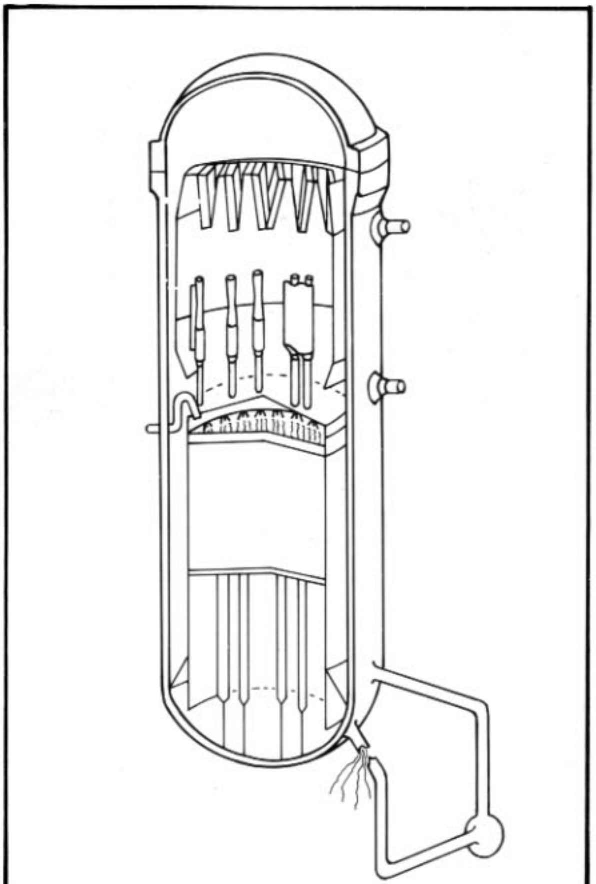
NORD

DECEMBER 1985

MANAGEMENT OF RADIOACTIVE WASTE RESULTING FROM NUCLEAR FUEL DAMAGE



nka

Nordic
liaison committee for
atomic energy

nka

Nordisk
kontaktorgan for
atomenergispørgsmål

Nordiska
kontaktorganet för
atomenergifrågor

Pohjoismainen
atomienergia-
yhdyselin

Nordic
liaison committee for
atomic energy

SKI 85:2

MANAGEMENT OF RADIOACTIVE WASTE RESULTING FROM NUCLEAR FUEL DAMAGE

SUMMARY REPORT ON PROJECT NKA/AVF-1

JAN ELKERT and MATS ULLBERG

ASEA-ATOM

SWEDEN

DECEMBER 1985

STEERING COMMITTEE FOR THE WASTE PROGRAMME
1981-1985 (THE AVF-PROGRAMME)

R Boge	National Institute of Radiation Protection, Sweden (SSI)
M Bonnevie-Svendsen	Institute for Energy Technology, Norway (IFE)
K Brodersen	Risø National Laboratory, Denmark
P Emmersen *	National Agency of Environmental Protection, Denmark
A Larsson	Swedish Nuclear Power Inspectorate (SKI)
B Mandahl	Oskarshamn Nuclear Power Plant, Sweden (OKG)
F Marcus	Nordic Liaison Committee for Atomic Energy (NKA)
H Raumolin	Teollisuuden Voima Oy, Finland (TVO)
M Snellman	Technical Research Centre of Finland (VTT)
S Väisänen	Finnish Centre for Radiation and Nuclear Safety (STUK)

THE PROJECT GROUP FOR THE NKA/AFV-1 PROJECT HAS INCLUDED:

R Boge	SSI
A Eriksson	Swedish State Power Board
A Larsson **	SKI
P Lehtinen	TVO
B Mandahl	OKG
S Norrby	SKI
U Ottersen	IFE
E Peltonen	VTT
S Ruotsalainen	TVO

LIST PARTICIPANTS FROM ASEA-ATOM:

M Berg, B Bhushan, J Elkert ***, L-G Envall, G Granath, B Grapengiesser, J-E Sandström, M Ullberg, K Westerberg, H Wijkström

* Chairman
** Project leader
*** Project responsible within ASEA-ATOM

ABSTRACT

Additional amounts of radioactive waste are generated in nuclear power reactors in case fuel elements are damaged. Two cases of theoretical damage are studied: Pipe rupture of one of the main circulation loops of the reactor, and blockage of a coolant channel of a fuel assembly. The systems installed for handling of radioactive waste at nuclear power plants in the Nordic countries appear to adequately handle such waste. Higher flexibility in waste management may in certain cases be achieved through additional storage capacities and circulation alternatives.

INIS Descriptors

FUEL ELEMENT FAILURE

INTERMEDIATE-LEVEL RADIOACTIVE WASTES

LIQUID WASTES

NUCLEAR POWER PLANTS

RADIOACTIVE WASTE PROCESSING

RUPTURES

FINLAND

SWEDEN

This report is part of the safety programme sponsored by NKA, the Nordic Liaison Committee for Atomic Energy, 1981-85. The project work has been partly financed by the Nordic Council of Ministers.

SUMMARY

The Nordic AVF-1 project deals with the management of radioactive waste originating from nuclear fuel damage. The intention was to get a good understanding of the nature of different events leading to fuel damage with the main interest directed towards the safe handling of the nuclear waste. The study was administered by the Swedish Nuclear Power Inspectorate. It was financed partly by the Inspectorate and partly by the Nordic Council of Ministers. The study was carried out by ASEA-ATOM.

During the operation of a nuclear power reactor small amounts of radioactive material are released into the coolant system. The material is trapped by the cleaning system for the cooling water by means of filtration and ion exchange. The resulting waste is incorporated in cement or bitumen and is then transferred to a final repository for radioactive waste.

Additional radioactive material can be released to the cooling water if the cladding of the fuel rods is damaged. According to Nordic experience the actual number of fuel rod failures has been considerably lower than was anticipated before the plants were taken into operation. This type of failure usually occurs progressively during operation.

Damage of the fuel can also arise suddenly. Two such cases are treated in this project: the rupture of a pipe in one of the four main circulation circuits of the reactor, and blockage of a cooling channel preventing normal cooling of a fuel assembly. Two Swedish nuclear reactor plants, constructed by ASEA-ATOM, were used as reference plants: Oskarshamn 2 and Forsmark 1, respectively.

The main purpose of the investigation is to verify the principles for the mangagement of the radioactive waste, should such incidents or incidents of similar character nevertheless happen.

In the first case some over-heating of the reactor core would occur in spite of the fact that cooling water is fed to the core by the emergency cooling systems. Pessimistically it is assumed that the cladding of 1 % of the fuel rods is damaged, resulting in a release of 10 % of the noble gases, iodine and cesium. The radioactivity released from defective fuel rods in normal operation is small compared to the contribution from these rods. In the longer term the total activity is dominated by cesium isotopes. Twelve weeks after the incident they amount to 370 TBq (10,000 Ci).

The activity released outside the reactor core is mainly found in the emergency cooling systems, but some would also appear in the system for controlled drainage. During the first weeks after the incident the access to areas where these systems are installed will have to be strictly limited. It should be possible to enter other parts of the reactor building, perhaps somewhat limited if airborn radioactivity is present.

During the weeks following the incident the most important task is to keep the reactor cooled by filling the containment with water. The necessary ventilation of the containment may be performed after some weeks with only minor radiation impact on people living in the neighbourhood.

Before the reactor pressure vessel is opened, the water in the pressure vessel and the containment has to be purified in order to prevent the spread of radioactivity to the reactor hall and reduce the dose rates around the reactor pool. This purification would probably only start about 4 weeks after the incident. A period of 3 to 4 months is estimated to be required to reach an acceptable radiation level around the reactor pool. Between 40 000 and 50 000 m^3 of water would have to be treated resulting in a quantity of solidified waste amounting to 500-700 cement blocks or 700-1000 drums with waste solidified in bitumen.

Initially, the specific activity of the water to be treated is quite high. Since the concentration of the activity in filter materials and ion exchange resins is limited, it results in a correspondingly high production of spent resins. Under these circumstances the system for waste conditioning cannot keep up with the water purification system. If the tanks for storing spent resins are too small the capacity of the waste conditioning system would gradually restrict the rate at which the water can be purified. Later on when the activity level of the water reaches more normal levels, it is the capacity of the purification system that becomes the limiting factor.

Even when the water in the containment has been purified a considerable radioactive contamination remains, especially on concrete surfaces and in insulating materials. The clean-up of the containment is a difficult job, which is estimated to go on for some months and to give rise to a large part of the total occupational radiation exposure from the incident.

In the second case (blocked cooling channel) it is assumed that the release of activity takes place rather slowly. Accordingly, no automatic shutdown of the reactor occurs before it is manually stopped after 30 minutes. The fuel cladding is estimated to collapse due to the strong temperature rise. Based on experience from various experiments, it is assumed for this case that the total content of the noble gases and iodine, as well as 60 % of the cesium in the fuel assembly, is relased. In this scenario the cesium isotopes again dominate the activity picture. After 12 weeks they amount to 500 TBq (14,000 Ci).

The two events studied then give about the same amounts of cesium to be taken care of. The radioactivity in the second case is, however, dispersed into more of the systems outside the reactor containment. This results in retricted access to those rooms where the cooling system for the shutdown reactor and the purification system for reactor water are installed. Restrictions will initially also involve spaces around the turbine condenser and the reactor scram system.

The remainder of the material in the damaged fuel assembly may be released in the form of more or less subdivided fuel fragments. The main part of the fragments would probably settle to the bottom of the reactor pressure vessel, but 0.1 promille (ca 20 g) is estimated to be carried away from the pressure vessel and end up in systems outside the containment. This transport is not expected to have any notable influence on the radiation levels.

Actions after the reactor scram include normal operational measures for shutdown with additional measures to limit activity release. Reconnection of the water purification system is a part of the normal procedure.

Two methods have been proposed to take care of the radioactivity in the reactor vessel. In the first the water would be purified in the reactor water purification system and the ion exchange resin subsequently transferred to the waste handling plant. Due to the necessity to limit the specific activity of the resin, about 50 charges of ion exchange resin are needed. Such a large amount of resin would exceed the capacity of the solidification plant and therefore lead to problems of transfer and storage.

Alternatively, the contaminated water in the reactor vessel could be transferred to the waste treatment plant before purification. This gives greater flexibility. The water can be stored in untreated form; it can be purified in several parallel units; it can be diluted before filteration; or evaporation can be used, etc.

The particular ion exchanger in service during the first 30 minutes after the start of the incident will become strongly radioactive. The resin is assumed to remain in place for a considerable time to allow for decay of the radioactivity. After this, treatment by conventional means would be possible.

Probably the most difficult task in this case is to take care of the damaged fuel and fuel fragments. The damaged fuel assembly has to be handled and removed with the aid of special tools, and sludge suction must be applied, especially to the bottom of the reactor pressure vessel. The finely subdivided fuel debris accumulated in the systems outside the containment can be collected in the ion exchangers of the reactor water purification system after washing and rinsing with pure water.

In the two cases studied no weak point was found in the waste treatment plants which would limit the treatment and conditioning of the radioactive waste after the postulated fuel damage. However, when operating personnel at Swedish and Finnish nuclear power plants with boiling water reactors were asked to submit comments on a number of questions regarding the management of radioactive waste after such incidents, their answers indicated that some problems may exist at certain plants. These findings will have to be examined further, however. The results from the questionnaire can be summarized as follows:

1. Capability to take care of strongly active leakage water. The waste treatment plant should have the capacity to receive of the order of 100 m^3/day strongly active leakage water.

2. Capability to store active water in the reactor containment. As a backup to the purification in the waste treatment plant, means should exist for the transfer of water to the containment, which is the best radiation shielded storage space for large volumes of radioactive water.

3. Capability to purify the water stored in the containment by circulation through the waste treatment plant.

4. Possibility to follow the activity build-up of filters and ion exchangers. The dose to the ion exchange resin can, in a short time, reach such levels that its chemical, and possibly its mechanical, properties are affected. Probably 1 MGy (100 Mrad) can be regarded as an approximate threshold value above which difficulties of emptying the ion exchanger can arise. This dose could be achieved about 1 month after the shutdown of the reactor. This time will be shorter if, after the beginning of the incident, the ion exchanger is operated for more than 30 minutes.

5. Capability for temporary storage of filter materials and ion exchange resins. Purification of strongly active water leads to high production of spent resins because of necessary limitations due to the high specific activity. This imposes special demands on the capacity of the waste conditioning plant.

Measures which in the long term would facilitate the management of the radioactive waste from a moderately large fuel damage incident should primarily be directed towards training of the personnel involved. Present emergency planning exercises are mainly concerned with procedures for the protection of the environment. Such exercises are normally terminated when the cooling of the reactor core has been secured.

From the study it can be concluded that the waste treatment plants at Nordic nuclear power plants are constructed, and have a capacity, such that they can treat the radioactive waste resulting from potential incidents which may give rise to moderately large fuel damage. The involvement of the operating personnel in the study has increased their conciousness and knowledge of the special problems related to such incidents, and this could lead to some additional measures being adopted in the future.

SAMMANFATTNING

Det nordiska AVF-1 projektet behandlar omhändertagandet av radioaktivt avfall efter en bränsleskada. Avsikten var att få en god förståelse av karaktären hos de olika händelser som kan leda till en bränsleskada med huvudintresset riktat mot det säkra omhändertagandet av det radioaktiva avfallet. Utredningen har administrerats av den svenska kärnkraftinspektionen. Den har finansierats delvis av inspektionen och delvis av det nordiska ministerrådet. Utredningen har genomförts av ASEA-ATOM.

I en kärnkraftreaktor frigörs små mängder radioaktivt material från bränslepatronerna till kylvattnet under driften. Materialet fångas upp i kylvattnets reningssystem där det omhändertas genom filtrering och jonbyte. Det resulterande avfallet gjuts in i cement eller bitumen och förs sedan till ett slutlager för radioaktivt avfall.

Ytterligare radioaktivt material kan avges till kylvattnet om bränslestavarnas kapsling skadas. Det verkliga antalet skador av detta slag har enligt nordiska erfarenheter varit väsentligt lägre än vad som förutsatts innan anläggningarna togs i drift. Skadorna är av sådan art att de vanligtvis uppstår successivt under driften.

Skador på bränslet kan också uppkomma plötsligt. Två tänkbara fall undersöks i detta projekt: brott på ett rör i en av reaktorns huvudcirkulationskretsar, och blockering av en kylkanal som hindrar den normala kylningen av en bränslepatron. Som referens har tagits två svenska reaktorstationer med kokarreaktorer konstruerade av ASEA-ATOM: Oskarshamn 2 respektive Forsmark 1.

Syftet med undersökningen är närmast att verifiera principerna för omhändertagandet av avfallet ifall händelser av motsvarande karaktär ändå skulle inträffa.

I det första fallet sker en viss upphettning av härden, trots att härden tillförs vatten via nödkylsystemen. Det antas konservativt att kapslingen hos 1 % av stavarna skadas, varvid 10 % av ädelgaser, jod och cesium frigörs. Den radioaktivitet som avges från de redan under normaldriften skadade bränslestavarna är liten i jämförelse med bidraget från dessa stavar. Den totala aktiviteten domineras på lång sikt av cesiumisotoper. Tolv veckor efter haveriet svarar de för 370 TBq (10,000 Ci).

Den frigjorda aktiviteten finns utanför reaktorinneslutningen huvudsakligen i nödkylsystemen men även i systemet för kontrollerat dränage. De utrymmen där dessa system är installerade får starkt begränsad tillträdbarhet under de första veckorna efter haveriet. Övriga delar av reaktorbyggnaden bör i allmänhet vara tillträdbara med reservation för begränsningar på grund av eventuell radioaktivitet i luften.

Den närmaste veckan efter haveriet koncentreras resurserna sannolikt på att säkerställa kylningen av reaktorn genom att fylla upp inneslutningen med vatten. Den vädring av inneslutningen som då måste äga rum kan ske efter några veckor. Stråldosen till de närmast boende har beräknats vara helt obetydlig.

Innan reaktortanken öppnas måste vattnet i reaktortank och inneslutning renas för att minska spridningen av radioaktivitet till reaktorhallen och minska dosraterna kring reaktorbassängen. Sannolikt kommer man att vänta med denna rening till ca 4 veckor efter haveriet. Att nå en acceptabel strålningsnivå kring reaktorbassängen uppskattas kräva en tid av 3 till 4 månader. Då har mellan 40000 och 50000 m^3 vatten behandlats och överförts till fast form, ingjutet i betong eller blandat med bitumen (asfalt). Förvaringen uppskattas kräva mellan 500 och 700 behållare av betong eller mellan 700 och 1000 bitumenfat.

Det behandlade vattnets specifika radioaktivitet är till en början ganska hög och begränsningar beträffande aktivitetens koncentrering i filter-och jonbytarmassor leder till en motsvarande hög produktion av avfallsmassor. Ingjutningssystemet kan då inte hålla jämna steg med vattenreningen. Om tankarna för lagring av de producerade avfallsmassorna är för små kan ingjutningssystemets kapacitet så småningom komma att begränsa den hastighet varmed vattnet kan renas. I ett senare skede blir vattnets aktivitetsnivå mera normal och kapaciteten hos vattenreningssystemet blir då begränsande.

När vattnet i inneslutningen renats kommer en betydande radioaktiv försmutsning att kvarstå, speciellt på betongytor och i isoleringsmaterial. Städningen av inneslutningen är ett besvärligt arbete som uppskattas ta några månader och svara för en betydande del av stråldosen till personalen.

I det andra fallet (blockerad kylkanal) antas att aktivitetsfrigörelsen sker relativt långsamt, så att ingen automatisk avställning av reaktorn äger rum innan den stoppas manuellt efter 30 minuter. Bränslekapslingen beräknas kollapsa på grund av den kraftiga temperaturhöjningen. Baserat på erfarenheter från olika försök antas här att hela innehållet av ädelgaser och jod samt 60 % av cesium i patronen frigörs. I detta scenario dominerar cesiumsisotoperna i ännu högre grad aktivitetsbilden och ger 12 veckor efter haveriet 500 TBq (14 000 Ci).

De båda studerade primärhändelserna ger sålunda ungefär samma mängd cesium att ta om hand. Aktiviteten sprids emellertid i det andra fallet till väsentligt fler system utanför reaktorinneslutningen. Tillträdet måste t ex begränsas till utrymmen där kylsystemet för avställd reaktor och reningssystemet för reaktorvatten är installerade. Tillträdesbegränsningar kommer till att börja med även att gälla kring turbinkondensorn och snabbstoppsystemet.

Resten av materialet i den skadade bränslepatronen kan frigöras i form av mer eller mindre finfördelade bränslerester. Det mesta sedimenterar sannolikt på botten av reaktortanken, men uppskattningsvis 0.1 promille av bränsleresterna (ca 20 g) förs ut från reaktortanken och hamnar i system utanför inneslutningen. Någon inverkan på strålningsnivån erhålles dock ej.

De åtgärder som vidtas närmast efter snabbstoppet är dels ordinarie driftåtgärder för avställning, dels åtgärder för att begränsa aktivitetsutsläpp. Till de ordinarie åtgärderna hör också återinkoppling av reaktorvattenreningen.

För omhändertagandet av radioaktiviteten i reaktortanken har två metoder föreslagits. Den ena innebär att vattnet renas i reaktorvattnets reningssystem och att jonbytarmassan överförs till avfallsanläggningen. På grund av att specifika aktiviteten hos jonbytarmassan måste begränsas krävs bortåt 50 satser jonbytarmassa. En så stor mängd massa kan dels leda till problem vid överföringen, dels problem med att lagra massan då produktionen kommer att överstiga solidifieringskapaciteten.

Om vattnet i reaktortanken i stället överförs orenat till avfallsanläggningen blir flexibiliteten större. Vatten kan då lagras obehandlat, upparbetning kan ske i flera parallella anläggningar, vattnet kan spädas före filtrering, indunstning kan användas som upparbetningsmetod m m.

Den jonbytare som antagits vara inkopplad under de 30 första minuterna av haveriet kommer att bli starkt aktiv. Massan föreslås bli kvar i jonbytaren under en längre avklingningstid, varefter omhändertagande på konventionellt sätt är fullt möjligt.

Omhändertagandet av skadat bränsle och bränslefragment är den kanske svåraste uppgiften i detta fall. Den skadade bränslepatronen måste plockas upp med specialverktyg och slamsugning måste ske framför allt av reaktortankens botten. Det finfördelade bränslet som hamnat i systemen utanför reaktorinneslutningen kan genom renspolning och sköljning tas om hand i jonbytarna i reaktorvattnets reningssystem.

För de båda fallen visar det sig att det inte finns någon speciell svag del av avfallsanläggningarna som blir styrande för omhändertagandet av det radioaktiva avfallet efter de postulerade bränsleskadorna. Driftpersonal vid svenska och finska kärnkraftverk med kokarreaktorer har fått lämna synpunkter på hur det radioaktiva avfallet kan tas om hand genom att besvara ett antal frågor. Det finns indikationer på att vissa detaljer vid enskilda reaktoranläggningar kan behöva ses över.

1. Möjlighet att omhänderta starkt aktivt läckagevatten. Avfallsanläggningen bör ha möjlighet att under en längre tid ta emot förslagsvis 100 m^3/dygn starkt aktivt läckagevatten.

2. Möjlighet att lagra aktivt vatten i reaktorinneslutningen. Som uppbackning till rening i avfallsanläggningen bör vatten kunna överföras till inneslutningen som utgör den bästa strålskärmade förvaringsplatsen för större volymer aktivt vatten.

3. Möjlighet att rena i inneslutningen lagrat vatten genom att låta det cirkulera genom avfallsanläggningen.

4. Möjlighet att övervaka aktivitetsuppbyggnaden i filter och jonbytare. Dosen till jonbytarmassan kan snabbt bli av en sådan storlek att dess kemiska och eventuellt även mekaniska egenskaper påverkas. Troligen kan 1 MGy (100 Mrad) betraktas som ett ungefärligt tröskelvärde över vilket svårigheter med tömningen av jonbytaren kan börja uppstå. Denna dos uppnås ca 1 månad efter reaktorns avstängning. Om jonbytarens drifttid efter haveriet förlängs utöver 30 minuter ökar dosen snabbare efter avstängningen, och tröskelvärdet uppnås tidigare.

5. Möjlighet att lagra filter-och jonbytarmassor. Vid rening av starkt aktivt vatten kan jonbytningsförmågan ej fullt utnyttjas, och därför ökar produktionen av jonbytarmassa. Ingjutningssystemet borde ha kapacitet att ta om hand detta.

Den åtgärd som på sikt mest skulle underlätta omhändertagandet av avfall från här behandlade osannolika händelser är i dag sannolikt insatser av utbildningskaraktär. De haveriövningar som hålls med driftpersonalen är till för att öva omgivningsskyddet. De avslutas därför normalt då härdens kylning är säkrad.

Allmänt kan man säga att undersökningen har visat att avfallsanläggningarna vid nordiska kärnkraftverk har en sådan utformning och kapacitet att de kan ta hand om det radioaktiva avfallet efter hypotetiska olyckshändelser i reaktorn innebärande begränsade bränsleskador. Vidare har en pedagogisk effekt erhållits genom att driftpersonalens medvetenhet och kunskap om de speciella problemen vid sådana händelser har ökat, och detta kan leda till att ytterligare några åtgärder vidtas i framtiden

LIST OF CONTENTS

1. INTRODUCTION

Management of radioactive waste from normal operation of nuclear power plants has been widely discussed. During the normal operation small amounts of radioactive material, predominantly activated corrosion products, are released from the fuel rod surfaces to the cooling water. The material released deposits on out-of-core surfaces or is trapped by the cleaning system for reactor water by means of filtration and ion exchange. The resulting waste is incorporated in cement or bitumen in the waste plant and eventually transferred to a final repository for radioactive waste.

Additional radioactive material - fission products - can be released to the cooling water if the cladding of the fuel rods is damaged. According to Nordic experience the actual number of fuel rod failures has been considerably lower than the number assumed for design purpose. This type of failure usually occurs progressively during operation. Except for the noble gases krypton and xenon, most of the fission product activity released is taken care of by the water cleaning systems and the waste plant in the same way as for the activated corrosion products.

Within NKA, the Nordic Liaison Committee for Atomic Energy, it has been discussed if the waste plants are suitable for handling the larger amounts of radioactive waste originating from a moderately large fuel damage suddenly arisen. It was decided to study the problem as a Nordic project work. The intention was to get a good understanding of the nature of the different events leading to fuel damage with the main interest directed towards the safe handling of the nuclear waste. The study was administered by the Swedish Nuclear Power Inspectorate and carried out by ASEA-ATOM with the active participation of Swedish and Finnish nuclear power utilities.

The project was divided into three parts:

1. Definition of the primary event

2. Analysis of the spreading of activity

3. Handling of the radioactive waste

Part 1 deals with the primary event where radioactivity is released. The description of the development of the event terminates with the determination of the resulting radioactivity release. Two basic cases are discussed: a combination of minor fuel damage with internal pipe rupture, and a case of sudden major fuel damage (blocked cooling channel), with normal spreading of activity in the systems for 30 minutes (Reference 1). The reference reactor in the first case is Oskarshamn 2, and in the second case, Forsmark 1.

The probability of such events is very low. The main purpose of the study is, however, to verify the principles for the management of the radioactive waste, should such incidents or incidents of similar character nevertheless happen.

Part 2 is a direct continuation of part 1 and comprises analysis of the spreading of radioactivity in the plant, calculations of radioactivity and dose rate and an investigation of how the radioactivity can be transferred to the waste plant for treatment (References 2-4).

Part 3 deals with the options available for dealing with the radioactive waste which has arisen. The aim is to shed light on the strategy for the necessary actions and the principles of waste handling. It is not concerned with the detailed design of processes and equipment. A feature of particular interest is to identify any limitations of the various waste plants that could delay the treatment of the released activity or make it more difficult, and to determine the areas in which additional investigations may be necessary.

Due to the higher levels of radioactivity and the large volumes of contaminated water, it will be impossible to quickly transfer the waste from the reactor plant to the waste plant.

It was considered important to have the operating personnel from the power stations participate actively in the study. To facilitate this participation, the project group has compiled a number of questions to which it considered the answers of the operating personnel to be valuable (Reference 5).

These answers were then compiled and comments on them prepared (Reference 6). This compilation is not related to specific plants, and merely reflects the general situation in the various power stations involved. The comments and conclusions of ASEA-ATOM are presented predominantly in Chapter 6 of this report. The utilities were also given an opportunity to submit their views on the information thus compiled.

The authors wish to thank the participating power utilities for their cooperation in answering the questions. Without this assistance, the investigation could not have been carried out in the intended manner. We trust that the project has also been of interest and value to the personnel at the participating plants.

2. MINOR FUEL DAMAGE AND INTERNAL PIPE RUPTURE

2.1 Description of the primary event

The design basis for Swedish and Finnish BWRs is that the fuel leakage corresponds to damage of approximately 1% of the fuel rods. This is known as Design Fuel Leakage (DFL). In addition, recoil leakage is assumed, in order to take into account possible contamination of the core surfaces with fissile material.

However, experience from reactor stations in Scandinavia and also abroad shows that leakage from the fuel is normally modest and, in the vast majority of cases, is far below the DFL. There are several reasons for this, and factors such as the method of operation and operating strategy play an important role. For Nordic conditions, fuel damage leakage amounting to approx. 0.02 x DFL is a realistic normal operating value. In this report, the conservative assumption is made that the fuel damage leakage during normal operation is 0.2 x DFL.

In the case of a reactor with external main recirculation pumps, the occurrence of leakage or rupture in the pressurized primary parts of the reactor containment will cause a loss of reactor coolant (LOCA) into the reactor containment vessel. The discharge of reactor water will raise the temperature and the pressure in the containment vessel. The safety circuits of the plant then initiate reactor scram and isolation of the containment.

If the flow caused by such a rupture is in excess of the capacity of the auxiliary feed water system, the pressure in the reactor vessel will drop. The drop in the pressure of the reactor coolant will cause an increase in the rate of leakage of fission products from fuel that was assumed to be damaged due to the increased pressure differential between the coolant and the fission gas space of the fuel rod. Experience indicates that the transient leakage will be in the form of a short peak, the maximum value of which may be several powers of ten higher than the normal operating leakage. After the transient stage, i.e. after the blow-down stage, the reactor pressure will stabilize and the discharge of fission products will taper off.

After the blow-down stage, the reactor core will be cooled by the emergency core cooling system. The system supplies cooling water to the reactor core through nozzles located above the core. Some increase in the temperature of the reactor core will then take place. As a result, the internal pressure in the fuel rods will increase, primarily in the rods that are not damaged. The increase in the internal pressure, combined with altered strength properties of the cladding material, may conceivably cause leakage to occur in certain fuel rods which were not previously damaged as a result of a partial rupture of the cladding.

Based on the licensing conditions, the proportion of fuel rods that may start to leak as a result of overheating of the core following a pipe rupture of the type considered here ("bottom rupture") can be calculated. In this connection the global power and burn-up distribution must be processed statistically. Leakage as a result of pipe rupture will only occur from fuel rods with high burn-up and high power loading. The number of damaged fuel rods will probably amount to less than 1%. It is assumed here that 1% of the fuel rods will be damaged as a result of pipe rupture.

Licensing is consciously based on conservative assumptions. The conditions will be appreciably more favourable if more realistic assumptions are made concerning systems available, system capacities and parameters used in the calculations.

The release of activity from the fuel in the event of pipe rupture can be divided into the following three contributory elements:

1) Leakage from fuel rods damaged during normal operation

2) Transient leakage from damaged fuel rods and leaching after pipe rupture has occurred

3) Leakage from fuel rods that are damaged as a result of pipe rupture.

During normal operation, the reactor water is subjected to continuous clean-up. The activity concentration in the reactor water is determined by the magnitude of the fuel damage and the capacity of the clean-up system.

The magnitude of the noble gas and iodine transients has been estimated from operating experience. The transient leakage of Cs-137 has been calculated on the basis of the measured relationship between I-131 and Cs-137 under transient conditions.

In addition to the transient release, consideration must also be given to the iodine and cesium leached out of the damaged fuel rods when the reactor is shut down. On the basis of operating experience, the leaching can be assumed to be 50 times higher than the leakage during normal operation, and this leaching can be assumed to last for 48 hours.

When the cladding of a fuel rod, that has not previously been damaged, is ruptured, a large proportion of the gap inventory of noble gases, iodine and cesium will be released. The fuel temperature will not exceed 1204°C. At temperatures below 1200°C, approx. 1% of the inventory of noble gases and less than 0.01 % of the cesium and iodine inventories will be released. However, at temperatures above 1200°C, the release will quickly increase to approx. 10% of the inventory of noble gases, cesium and iodine. It is therefore here conservatively assumed that partial rupture of the cladding of a fuel rod will result in the release of 10% of the inventory of noble gases, iodine and cesium. According to measurements carried out by ASEA-ATOM, this basically corresponds to the gap inventory of noble gases.

The releases assumed for other substances have been based on their volatility. For tellurium, it is also assumed that 10% of the inventory of the damaged fuel rods will be released, while the release of other substances included here has been somewhat conservatively set at 1%.

Due to their low volatility, actinides are expected to be released only in the form of particles that are stripped away when the rods are punctured. Based on puncturing experiments, it is assumed that 0.02% of the fuel in a damaged rod will be released.

A comparison between the three contributory elements in the release of activity reveals that leakage from fuel rods damaged as a result of pipe rupture makes the dominating contribution. Transient leakage and leaching are much more important than the contribution from normal operation damage.

What would happen if pipe rupture occurred in a reactor with internal main recirculation pumps? There is a vital design difference between reactors with internal and external recirculation pumps. As a result of the introduction of internal recirculation pumps, the design "bottom rupture" has been restricted to the leakage area between the recirculation pump shaft and nozzle. Any leakage along this path will be compensated by the auxiliary feed water system.

A "bottom rupture" would open up a flow area of 80 cm^2 and, based on otherwise conservative assumptions, the maximum estimated temperature of the fuel cladding will be less than 800°C, which involves very little risk of fuel cladding damage.

A "bottom rupture" in an internal recirculation pump reactor will therefore release much less activity than a corresponding pipe rupture in an external pump reactor.

If rupture should occur in a pipe which is pressurized by the reactor but is outside the containment, reactor water will flow out into the building in which rupture has occurred. The safety circuits of the plant will then initiate isolation of the rupture point from the reactor and, if the rupture has occurred in the steam pipes or feed water pipes or if the rupture flow is so large that the coolant level in the reactor pressure vessel has dropped to a low value, reactor scram will be initiated. The reactor is thoroughly cooled throughout the sequence of events, and further damage to the reactor fuel is improbable.

In summary, it can be said that, from the waste aspect, the event of greatest interest is an internal pipe rupture in an external pump reactor. Oskarshamn 2 has been selected as the reference station for this case.

The primary event in the case of minor fuel damage and internal pipe rupture is illustrated in Figure 2.1. More details about the interior of the reactor pressure vessel can be found in Figure 3.1.

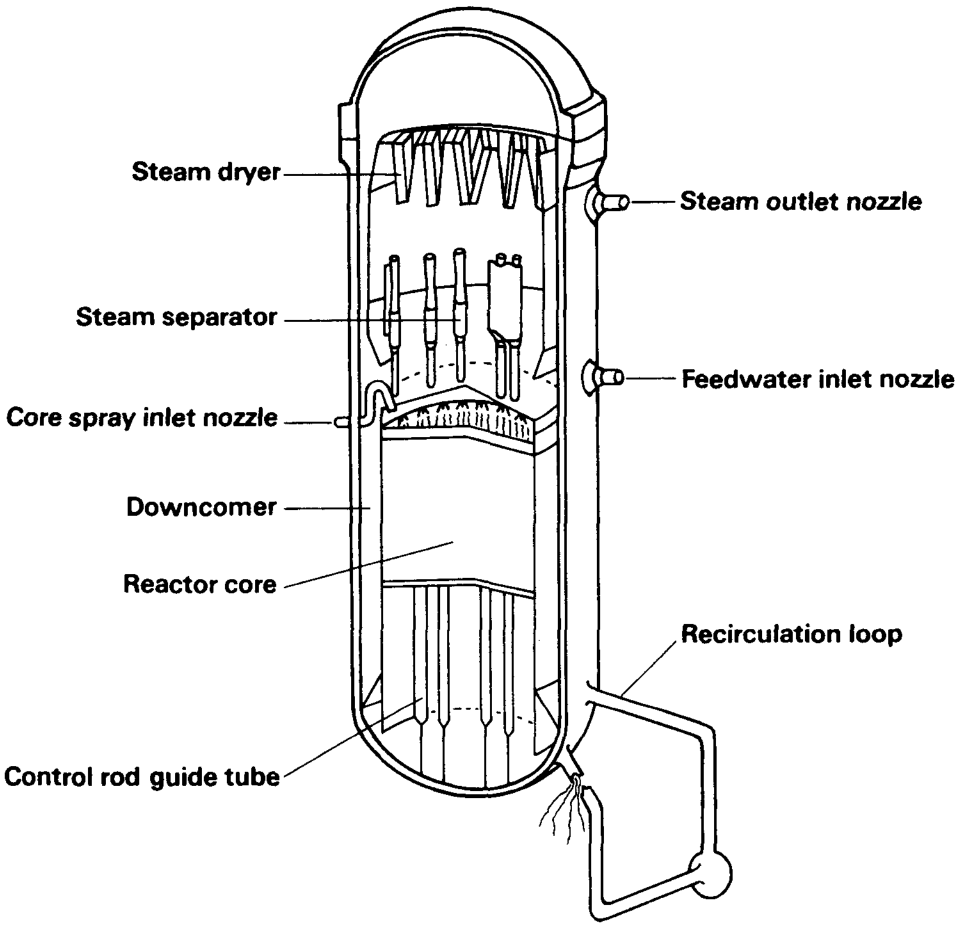


Figure 2.1 Internal Pipe Rupture.

2.2 Spreading of radioactivity in various systems

The activity released is partly gas-borne and partly water-borne. The latter activity is released primarily to the water from the low pressure coolant injection system (system 323) which cools the reactor core. The activity is spread through the rupture point to the condensation pool and the containment vessel spray system (system 322). A large volume of highly contaminated water is thus generated by the transfer to the condensation pool.

The gas-borne radioactivity, primarily consisting of noble gases, is discharged into the reactor containment through the rupture point. Spraying in the containment lowers the iodine activity in the gaseous phase to values significantly below 1% of the total iodine activity in the reactor containment. Iodine has therefore not been considered important in the gaseous phase.

It is assumed that, after a certain period of time, the noble gases will be released to the atmosphere through the filter system comprising particulate filters and iodine filters. The discharge of noble gases to the atmosphere can also be delayed until the direction of the wind is towards the sea. One of the participating power utilities has estimated the doses to the public after venting to be very low (See section 4.1).

System-related activity occurs outside the containment, primarily in the systems which are open to the containment or reactor pressure vessel. But activity can also spread to other systems through leaking isolating valves, system leakage through unexpected leakage points, such as heat exchangers, external leakage, etc. These types of leakage are normally maintained at a low level by design measures, leakage tests, inspection, etc.

Activity flows from expected leakage points may be relatively high, and the parts of the controlled leakage drain system (system 352) which are connected to systems with high activity outside the containment should therefore be considered. The same applies to filter sections in the off-gas system (system 341).

In summary, the systems in Oskarshamn 2 that may contain high activity are as follows:

322, Containment vessel spray system
323, Low pressure coolant injection system
741, Containment gas treatment system

and parts of systems

311, Main steam lines
341, Off-gas system
345, Controlled area floor drain system
352, Controlled leakage drain system
821, Sampling system

2.3 Activity calculations

In the activity calculations, no consideration has been given to activity disappearing from the water in systems 322 and 323 by leakage to system 352 or to system 345.

The quantity of water has been assumed to be 200 tons in the reactor and 1920 tons in the condensation pool. No clean-up of the water is assumed. Due to the magnitude of the pipe rupture, the shut-down cooling system (system 321) and the reactor water clean-up system (system 331) cannot be connected in after reactor scram.

All activity in the gaseous phase is assumed to be in the dry-well of the reactor containment. However, a small proportion may be dissolved in the water, and some may also remain in the reactor pressure vessel.

Part of the iodine and cesium activity may remain on the surfaces in the containment. This may amount to a few per cent of the total released activity.

Table 2.1 shows the total activity that may be discharged to the waste plant after 4, 8 and 12 weeks, respectively. This table also includes the activity of the actinides. The total activity after 12 weeks is 880 TBq (24 kCi) of fission products and 0.09 TBq (2.5 Ci) of alpha activity from actinides.

2.4 Radiation levels and accessibility

The accessibility to the reactor building can be assessed on the basis of Figure 2.2 which shows the radiation level on and around an 8-in. dia. pipe (DN200) belonging to system 322. Due to the high dose rate on and in the vicinity of the unshielded pipe, access to areas containing components belonging to system 322 and 323 is highly restricted during the first weeks following the accident.

To a certain extent, this also applies to areas in which system 352 is installed. Assuming five times lower activity concentrations than in the condensation pool, the surface dose rate 48 hours after the damage has occurred will be 100 mSv/h on the tank and 20 mSv/h on a 7 m long DN100 pipe.

However, by providing a relatively thin (30 cm concrete) radiation shield, the radiation from the system 322 pipe is cut down significantly (Figure 2.2). Other parts of the reactor building should therefore generally be accessible to a normal extent immediately after the accident. One exception is a corridor, to which access is highly restricted by a system 322 pipe.

Due to the 50 cm thick concrete wall, the dose rate outside the tank in the waste plant, to which the water can be transferred from the condensation pool, will be 0.25 mSv/h 48 hours following the accident.

Table 2.1 Minor fuel damage and internal pipe rupture

Total activity that may be discharged to the waste plant after the damage (TBq).

Nuclide	$T_{\frac{1}{2}}$	4 weeks	8 weeks	12 weeks
I-131	8.04 d	150	14	1.2
I-132	2.28 h	6.1		
Cs-134	2.06 y	190	180	180
Cs-136	13.1 d	11	2.6	0.6
Cs-137	30.2 y	200	200	190
Sr-89	50.5 d	120	84	58
Sr-90	28.8 y	14	14	14
Y-90	64.1 h	14	14	14
Y-91	58.5 d	1.1	0.8	0.6
Ru-103	39.4 d	130	78	48
Ru-106	367 d	51	48	46
Rh-103m	56.1 m	120	76	47
Rh-106	29.8 s	51	48	46
Ag-110m	252 d	0.3	0.3	0.3
Sb-124	60.2 d	1.3	0.9	0.7
Sb-125	2.7 y	1.0	1.0	1.0
Te-125m	58 d	0.1	0.1	0.1
Te-127m	109 d	15	12	10
Te-127	9.4 h	15	12	10
Te-129m	33.5 d	53	30	17
Te-129	69 m	34	19	11
Te-132	78 h	5.9		
Ba-137m	2.55 m	180	180	180
Ba-140	12.8 d	66	15	3.2
La-140	40.3 h	76	17	3.7
Np-239	2.35 d	$1.5 \cdot 10^{-2}$		$3 \cdot 10^{-5}$
Pu-238	87.7 y	$6 \cdot 10^{-3}$		$6 \cdot 10^{-3}$
Pu-239	$2.41 \cdot 10^{4}$ y	$1 \cdot 10^{-3}$		$1 \cdot 10^{-3}$
Pu-240	$6.57 \cdot 10^{3}$ y	$1 \cdot 10^{-3}$		$1 \cdot 10^{-3}$
Pu-241	14.4 y	0.45		0.45
Am-241	433 y	$6 \cdot 10^{-4}$		$7 \cdot 10^{-4}$
Am-242m	152 y	$3 \cdot 10^{-5}$		$3 \cdot 10^{-5}$
Am-243	$7.37 \cdot 10^{3}$ y	$3 \cdot 10^{-5}$		$3 \cdot 10^{-5}$
Cm-242	163 d	0.10		$8 \cdot 10^{-2}$
Cm-243	28.5 y	$5 \cdot 10^{-5}$		$5 \cdot 10^{-5}$
Cm-244	18.1 y	$5 \cdot 10^{-3}$		$5 \cdot 10^{-3}$

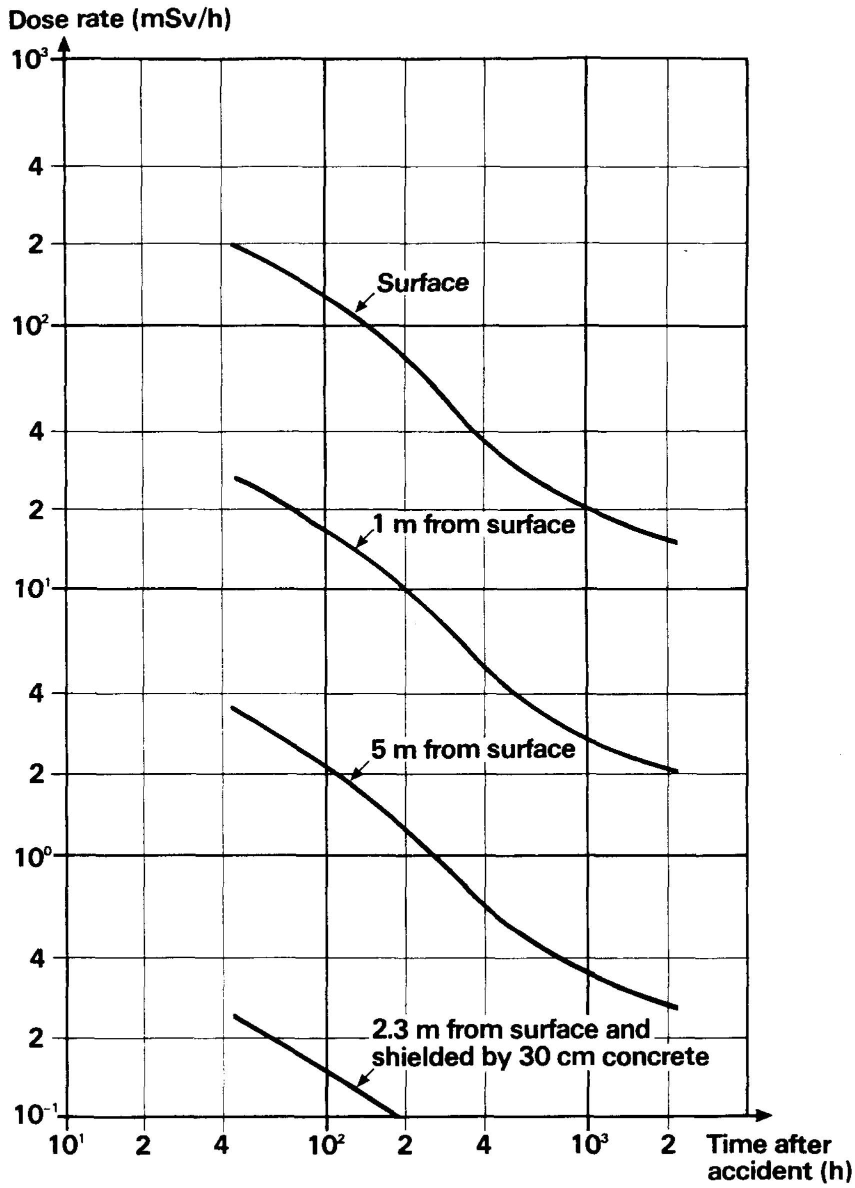


Figure 2.2 Minor Fuel Damage and Internal Pipe Rupture. Radiation Levels On and Around a System 322 Pipe (diam. 8 in., length 7 m).

3. BLOCKED COOLANT CHANNEL (Major fuel damage with normal spreading of activity)

3.1 Description of the primary event

Every fuel assembly in a BWR is provided with an inlet restriction. One of the functions of the restrictions is to reduce the effect of the power of the fuel assembly on the distribution of the coolant flow onto the various assemblies in the core. If the flow area of the inlet restriction is reduced, the coolant flow to the corresponding fuel assembly will be reduced, thus impairing the cooling. However, a reduction in the coolant flow will increase the steam content in the fuel assembly, which will lower the fuel assembly power. Due to this negative feedback, large flow reductions are necessary before dry-out can occur.

Figure 3.1 shows the location of the blockage assumed. Four fuel assemblies - fuel bundle in a fuel box - are placed on top of a control rod guide tube containing the coolant channels.

The magnitude of the flow reduction which is acceptable before dry-out can occur is dependent primarily on the power level in the relevant fuel assembly. Considering a fuel assembly with medium loading in a Nordic BWR, the flow must be reduced to approx. 25% before dry-out occurs, whereas the corresponding flow reduction in a fuel assembly with a high load is approx. 40%. To achieve this, the flow area of the inlet restriction must be reduced by approx. 80% and 60%, respectively.

The experience gained from the reactors which are in operation does not indicate any risk of the coolant flow to the fuel assembly being restricted to such an extent that dry-out would be caused. The experience is based on two cases in which components have worked loose in Nordic BWRs.

If the coolant flow should drop further after dry-out has occurred, the top sections of the fuel assemblies will be cooled predominantly by steam, and the consequent reduction in the heat transfer coefficient will cause an increase in the fuel temperature. However, the increase in the fuel temperature will lower the fissile power of the fuel, due to the negative temperature coefficient of the fuel.

The impaired internal moderation in the bundle also gives rise to a negative reactivity and power feedback. This applies primarily to the central rods in the fuel bundle, whereas the power level of the peripheral rods may still be relatively high, since these are supplied with thermal neutrons from adjacent fuel assemblies and gaps.

If the flow should be blocked entirely, the water in the fuel assembly will boil away within a matter of seconds and the temperature in the fuel will increase. Heat transfer from the fuel rods to the cold box wall will then take place by radiation. Since the radiant heat emitted to the surroundings is proportional to the fourth power of the absolute temperature, a stabilization of temperature will be obtained. No credit is taken for cooling by means of the water flowing into the box from the surrounding boxes due to the relatively high power of the fuel assembly.

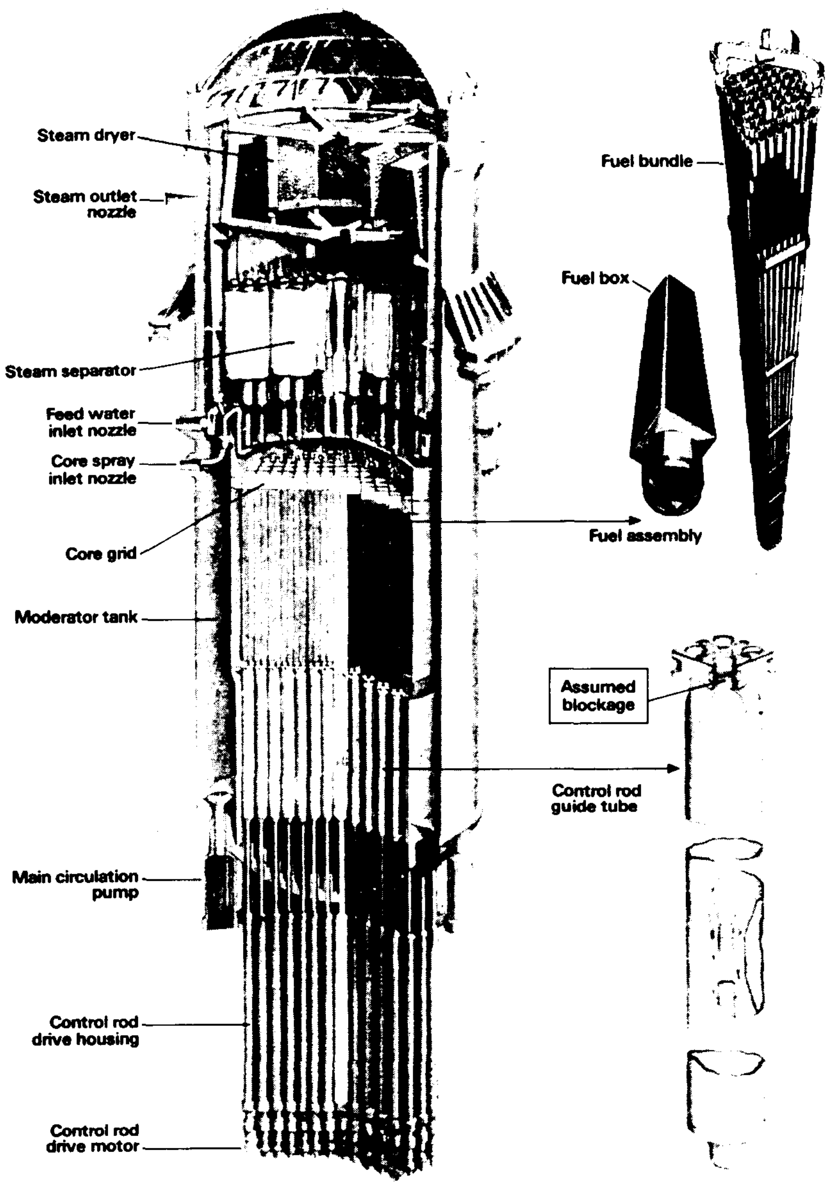


Figure 3.1 Blocked Coolant Channel. Location of the Blockage Assumed.

It cannot be excluded that, even after complete loss of coolant flow, the fuel assembly power will be at a level of several tens of a per cent of full power. At this power, the temperature will increase to a couple of thousand degrees if the heat is only dissipated from the fuel by radiation.

The melting point of the fuel cladding material is approx. 1850°C. However, the cladding reacts very strongly with steam in the temperature range below and at the melting point, and the cladding material will therefore be oxidized to ZrO_2 within a matter of minutes. In addition, the spacer material melts at around 1600°C, and it must therefore be assumed that the fuel bundle in the relevant channel will have collapsed entirely by the time the temperature has reached 2000°C.

If the temperature stabilizes at a level so high that extensive damage should occur instantaneously of the fuel cladding, then the release of activity to the reactor water and steam may initiate automatic scram and isolation of the reactor by the activity monitoring system in the main steam pipes. As a result of the scram the power of the fuel assembly to which the coolant flow is blocked will drop to the decay power level.

If the release of activity should take place more slowly, in case the flow to the fuel assembly is not completely restricted, activity metering in the off-gas system will draw the attention of the operator by initiating an alarm after than 5 - 20 minutes.

To make the subsequent treatment in the project of greater interest, it is assumed that the release of activity takes place relatively slowly and that the reactor is shut down manually after 30 minutes.

Due to the substantial temperature increase described above, it may be assumed that a large quantity of highly volatile fission products will be liberated when the fuel cladding has collapsed. These fission products are the noble gases xenon and krypton, as well as cesium in forms such as Cs_2Te, Cs_2Se, CsI, CsBr and Cs. Iodine and bromine will also be liberated in the form of RbI and RbBr.

At temperatures above 2000°C, additional fission products are liberated, such as barium (in the form of BaO) and molybdenum (in the form of MoO_2). In addition, antimony and silver have relatively high vapour pressures, even at temperatures below 2000°C.

Strontium occurs during operation in oxidized form (SrO). It has a very low vapour pressure at high temperature (20 Pa at approx. 2000°C). The release of strontium should therefore be small.

However, the above qualitative description based on the volatility of the substances does not provide a complete picture, since the release also includes other mechanisms, such as diffusion in UO_2, release in conjunction with the oxidation of UO_2, etc. But even so, the description agrees well with experiments carried out in order to quantify more accurately the release of fission products from fuel at high temperatures.

Based on the results of various tests, it is assumed here that 100% of the inventory of noble gases and iodine and 60% of the cesium inventory in the fuel assembly is released when the fuel decomposes. The release of other materials of interest is 10% or less of the fuel assembly inventory. For strontium, for instance, a value of 1% has been selected.

The release of actinides when the fuel assembly has collapsed is assumed here to be 0.02%, based on the estimate of release in the event of core melt-down. It is probable that the actinides will be released as finely distributed particles when the cladding of the fuel rods is ruptured before the assembly has collapsed. This is a further reason for the assumed proportion of actinides released (See section 2.1).

Since there is serious risk that the collapsed fuel will burn through the box wall, it must also be assumed that the remainder of the material in the damaged assembly can be released in the form of more or less finely distributed fuel. These uranium fragments will spread with the reactor water and will settle out, primarily on the bottom of the reactor pressure vessel.

In time, a certain proportion of the radioactivity in the uranium fragments will be leached out. This activity contribution can be estimated by means of the leaching rates for spent fuel. Most of the actinides have been shown to be released to the reactor water by this leaching. Fission products will also be leached out in time from the uranium fragments. However, this does not affect the total quantity of activity released to such an extent that taking it into account would be justifiable.

3.2 Spreading of radioactivity in various systems

The release of activity is assumed to take place relatively slowly in this scenario, so that the reactor will not be shut down by the activity monitoring system for the steam pipes, and normal operation will continue for 30 minutes. Scram and isolation will then be initiated manually due to high activity in the off-gas system.

In the 30 minutes during which operation continues, activity will spread to the primary systems and to the systems connected to the primary systems, including the turbine plant, condensate clean-up system and off-gas system. The activity will increase in the following systems of Forsmark 1:

321, Shutdown cooling system
331, Reactor water cleanup system
332, Condensate cleanup system
336, System for sampling and analyses
341, Off-gas delay system
345, Controlled area floor drain system
348, Recombiner system
352, Controlled leakage drain system
354, Hydraulic scram system
411, Main steam system
412, Turbine
413, Condenser and vacuum system
414, Condensate system

415, Feedwater system
418, Steam reheat system
419, Steam extraction system
421, Seal and leakage steam system
749, Off-gas filter system

After isolation, steam will be discharged to the condensation pool (approx. 250 tons over a period of 2.8 h), and activity will therefore be transferred to the condensation pool and thus also to system 322. When the operator has made certain that no pipe rupture has occurred inside the reactor containment, system 322 will probably be changed to only cooling the pool, and systems 321 and 331 will be restarted. However, in the subsequent work on the project, it is assumed that the ion exchanger in system 331 is not taken into operation. The heat exchangers are used to provide a certain amount of cooling.

Noble gas activity is transferred to the gaseous phase of the reactor containment (dry-well) via the condensation pool. In the same way as in the first scenario, it is assumed that this activity will be released to atmosphere via the filter system, which comprises particulate filters and iodine filters. In this case, the activity is appreciably lower and, as a result, so are the doses.

The fuel in the blocked fuel assembly is released entirely or partially in finely distributed form. Most of these uranium fragments will settle in the bottom of the reactor pressure vessel.

The amount of fuel residue deposited in the pipes, heat exchangers and ion exchanger of systems 321 and 331 have been estimated by means of flow calculations in the reactor pressure vessel and a realistic particle size distribution, based on experiments and a model for particle deposition in horizontal pipes (Reference 7). In the case of an internal pump reactor, such as Forsmark 1, in which the reactor water for system 321 is taken from the gap between the reactor pressure vessel and the moisture separator shell, only particles smaller than 40 μm are expected to reach system 321. The calculations indicate that only 20 g (equivalent to 0.01 % by weight) of the fuel residue is carried out of the reactor in 30 minutes. Of this amount, 83% is trapped in the ion exchanger in system 331, and the rest is deposited in the heat exchangers and on horizontal pipe surfaces.

3.3 Activity calculations

With the exception of the noble gases, which are entrained with the reactor steam, it is assumed that the released activity will be distributed in the 250 tons of reactor water. The clean-up flow to the ion exchanger of system 331 is 25 kg/s. The concentration of iodine in the reactor steam is assumed to be 0.02 times the reactor water concentration, whereas other fission products are assumed to be entrained with the steam in proportion to the estimated steam moisture content of 0.1%.

Iodine and other fission products are assumed to be entrained by the steam blown down into the condensation pool for 2.8 h, in the same way as the fission products which are entrained with the reactor steam to the turbine.

The noble gas activity liberated when damage has occurred and that caused by decay during the first 30 minutes are assumed to go directly to the delay tank in the off-gas system. This tank is assumed to be isolated after such a time that all noble gas activity and daughter product activity transferred to it will be retained in the delay tank, the total volume of which is 900 m^3 (gaseous volume 350 m^3).

Out of the remaining activity discharged from the reactor pressure vessel with the steam, 90% will be retained in the condensate filters (system 332). The remaining 10% will not reach these filters when the reactor is shut down and will remain in the turbine systems upstream of the filters of system 332.

The noble gas activity (xenon) formed by decay after the first 30 minutes is assumed to remain dissolved in the medium in which decay takes place. This does not apply to the condensation pool, in which the noble gas activity is assumed to rise into the dry-well of the reactor containment.

The activity in the ion exchanger is that which has been accumulated during the first 30 minutes after occurrence of the hypothetic accident. A small and probably negligible proportion of iodine and caesium activity in the condensation pool may go into the gaseous phase of the containment. No spraying takes place in this case.

Table 3.1 shows the total activity that may be discharged to the waste plant after 4, 8 and 12 weeks, respectively. This table also contains the actinide activity. The total activity after 12 weeks is 780 TBq (21 kCi) of fission products and 1.8 TBq (50 Ci) of alpha activity from the actinides. The released activity is of the same magnitude as in the first scenario, although it is spread to many more systems.

3.4 Radiation levels and accessibility

The radiation level on and around an 10-in. dia. (DN250) pipe belonging to system 321 is shown in Figure 3.2.

The dose rates will be roughly as high from a DN100 pipe belonging to system 352. The access to areas with systems containing high activity will therefore be highly restricted during the first weeks following the accident assumed.

The calculated radiation levels from deposited fuel residues are much lower than those obtained from the activity relased to the reactor water (Reference 7). At a distance from the pipes and heat exchangers in system 321 and 331, the dose rate is generally less than 1 mSv/h only a few days after an accident occurs. Accessibility to the installation is therefore not affected by deposited fuel residues.

Table 3.1 Blocked coolant channel

Total activity that may be discharged to the waste plant after the damage (TBq)

Nuclide	$T_{\frac{1}{2}}$	4 weeks	8 weeks	12 weeks
I-131	8.04 d	330	30	2.2
I-132	2.28 h	1.5		
Cs-134	2.06 y	250	250	240
Cs-136	13.1 d	15	3.3	0.8
Cs-137	30.2 y	260	260	260
Sr-89	50.5 d	28	19	13
Sr-90	28.8 y	3.1	3.1	3.1
Y-90	64.1 h	3.1	3.1	3.1
Y-91	58.5 d	0.3	0.2	0.1
Ru-103	39.4 d	0.6	0.4	0.2
Ru-106	367 d	0.2	0.2	0.2
Rh-103m	56.1 m	0.6	0.4	0.2
Rh-106	29.8 s	0.2	0.2	0.2
Ag-110m	252 d	0.8	0.7	0.7
Sb-124	60.2 d	0.9	0.6	0.5
Sb-125	2.7 y	0.7	0.7	0.7
Te-125m	58 d		0.1	0.1
Te-127m	109 d	3.6	3.0	2.5
Te-127	9.4 h	3.6	3.0	2.5
Te-129m	33.5 d	12	7.0	4.0
Te-129	69 m	8.0	4.5	2.5
Te-132	78 h	1.4		
Ba-137m	2.55 m	240	240	240
Ba-140	12.8 d	7.9	1.7	0.4
La-140	40.3 h	9.1	2.0	0.4
Np-239	2.35 d	$9.4\cdot10^{-2}$		$4\cdot10^{-4}$
Pu-238	87.7 y	$3.5\cdot10^{-2}$		$7.0\cdot10^{-2}$
Pu-239	$2.41\cdot10^{4}$ y	$8\cdot10^{-3}$		$1.5\cdot10^{-2}$
Pu-240	$6.57\cdot10^{3}$ y	$9\cdot10^{-3}$		$1.7\cdot10^{-2}$
Pu-241	14.4 y	2.7		5.4
Am-241	433 y	$4\cdot10^{-3}$		$8\cdot10^{-3}$
Am-242m	152 y	$2\cdot10^{-4}$		$4\cdot10^{-4}$
Am-243	$7.37\cdot10^{3}$ y	$2\cdot10^{-4}$		$4\cdot10^{-4}$
Cm-242	163 d	1.2		1.6
Cm-243	28.5 y	$5\cdot10^{-4}$		$8\cdot10^{-4}$
Cm-244	18.1 y	$5.4\cdot10^{-2}$		$8.0\cdot10^{-2}$

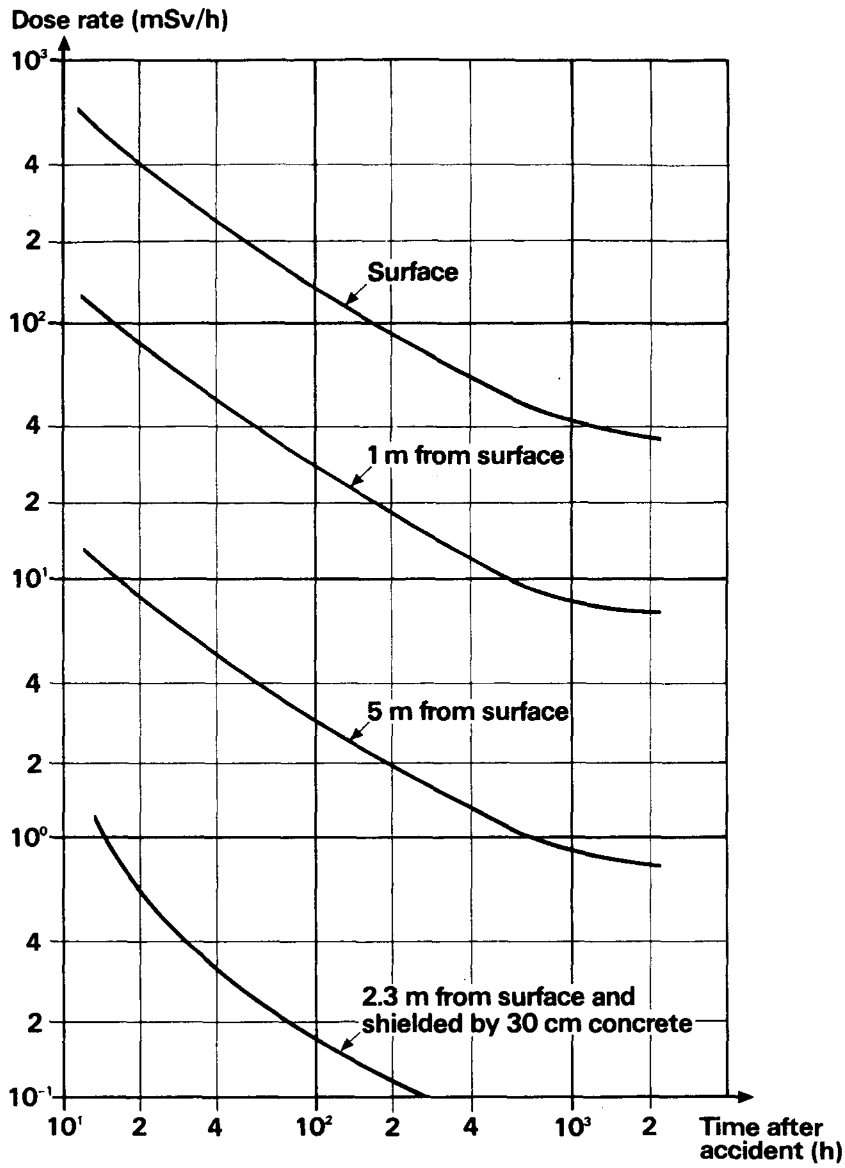


Figure 3.2 Blocked Coolant Channel. Radiation Levels On and Around a System 321 Pipe (diam. 10 in., length 4 m).

The ion exchanger in system 331 will accumulate high activity during the first 30 minutes after occurrence of the damage. However, dose rate calculations show that 100 cm thick concrete provides good shielding no later than 1 h after the accident. The dose rate will then be 0.2 mSv/h. The activity is assumed to be contained in a 1 cm thick radial layer in the outer part of the ion exchanger. Uniformly distributed activity gives approximately 40% lower dose rates.

Dose rate calculations have also been carried out for a DN200 pipe containing condensation pool water and a DN100 pipe belonging to scram system 354 (1% of the activity concentration in the reactor water). Twelve hours after the accident, the surface dose rates will be 2.0 mSv/h and 7.0 mSv/h, respectively. Access to areas containing system 354 will therefore also be restricted during the first week.

The reactor building should otherwise be accessible roughly to the normal extent after the accident.

The dose rate around the condenser in the turbine building has been determined on the assumption that it contains half of the activity in the turbine systems upstream of the filters in system 332 at the instant of scram. Twelve hours after occurrence of the damage, the radiation level will be 30 mSv/h on the surface and will have dropped to 2 mSv/h after one week. At a distance of 5 metres, the corresponding values will be 8 mSv/h and 0.5 mSv/h, respectively. The accessibility in the vicinity of the turbine condenser will thus initially be restricted.

The dose rate calculations for the waste tank mentioned in section 2.4 have been repeated with the reactor water activity concentrations. As compared to the earlier calculations, the radiation levels will generally be higher. A figure of 0.60 mSv/h is obtained, for instance, for 50 cm of concrete and 48 hours after the accident.

3.5 Radiation dose to the ion exchange resin

In addition to the dose rate of the ion exchanger in system 331 being very high, the dose to the ion exchange resin will also quickly be of such magnitude that the chemical and possibly also the mechanical properties of the ion exchange resin will be affected. The magnitude of the dose up to a certain time depends on how the activity is distributed in the resin bed. Collection takes place in a zone between spent and fresh ion exchange resin. The width of the zone depends on the type of ion, flow rate, temperature, etc. and obviously on the properties of the ion exchange material. The calculation of the dose to the ion exchange resin is based on the assumption that the activity is uniformly distributed in a 1 cm thick layer. After 12 weeks, the dose will then be 1.4 MGy (140 Mrad) in this layer. If it is assumed instead that the activity is uniformly distributed throughout the bed, the dose after the same period of time will be 0.15 MGy.

How ion exchange resins are affected by ionizing radiation is an area which has been thoroughly investigated at an early stage. The threshold value for damage is about 0.1 MGy for anionic resin and, depending on the degree of cross-linking, up to ten times higher for cationic resin. Mixed beds release significantly lower quantities of decomposition products than pure anionic and cationic beds. Doses up to 4 MGy do not affect the volume or pressure drop across an anionic bed. However, at doses in excess of 1 MGy, tendencies towards the bed lumping together have been observed. The value of 1 MGy can probably be considered to be an approximate threshold value, above which difficulties may start to occur in discharging the bed ion exchanger belonging to system 331.

4. HANDLING OF THE RADIOACTIVE WASTE AFTER MINOR FUEL DAMAGE AND INTERNAL PIPE RUPTURE

4.1 Measures to be taken in the reactor plant

According to the accident management procedures, the sequence comprising I isolation (i.e. isolation of the containment) and starting of the emergency cooling system and containment spray system leads to the internal emergency force being called and being established within one hour. The high level of activity in the containment will be indicated by the existing equipment. Two of the stations included in the study have radiation protection personnel on each of the shifts, whereas the other two have radiation protection personnel on call, and these can be at site within one hour.

During the days immediately after an accident, attention will mainly be concentrated to the emergency cooling systems. In addition, a close check will be kept on the activity pattern and the emission situation.

Operation of the waste plant may have to be re-scheduled, since it will have to accept drainage water originating from the reactor containment. However, planning of the collection of the released activity will probably not be necessary during the first week following an accident. Priority will be given to filling the containment with water, in order to safeguard the cooling of the reactor.

The means available for sampling the water and gas in the containment following an accident varies on the reactors that were studied. However, it is not considered that the activity level in the condensation pool is of any significance to the operating status of the reactor. On the other hand, the activity of the containment atmosphere is important for the venting of the containment which must be carried out at the time when the containment is filled with water. In view of the need for a planned emission of activity, filling of the containment will be a matter for consultation with the authorities, and it is assumed that good monitoring of the activity in the containment gaseous phase and of the activity emitted to atmosphere will be carried out.

The doses which occur when venting after four weeks have been studied in more detail for one of the reactors. The whole body dose to which the critical group will be subjected by the noble gases will be approx. 10^{-6} mSv, based on the annual mean values. The corresponding collective dose will be 5 mmanSv (Kr-85). Assuming that 3% of the iodine released is in the gaseous phase and that the collecting efficiency of the iodine filters in the ventilation system is 99%, the iodine discharged to atmosphere would be $4.5 \cdot 10^{10}$ Bq, corresponding to a weighted dose equivalent of $4 \cdot 10^{-4}$ mSv. These low values indicate that venting to atmosphere could be carried out appreciably earlier, if it is considered important from the core cooling aspect.

Before the reactor pressure vessel is opened and the fuel in the core is removed, the water in the reactor pressure vessel and reactor containment will probably have to be cleaned, in order to minimize the spreading of radioactivity to the reactor service room and to minimize the dose rates around the reactor pool. Appreciable surface contamination will remain, even after the water in the containment has been cleaned. This contamination is expected to be severe, particularly in insulating material, on concrete surfaces, on horizontal surfaces and in pockets into which radioactive matter has settled. The time necessary for decontamination is expected to be several months.

One power utility points out that, if 1% of the released radioactivity is assumed to be deposited as surface contamination, this will give rise to very high dose rates in the containment and will represent a large contribution to the collective dose for processing the waste. Gradual decontamination and removal of the insulation under a covering of water is mentioned as a method which may possibly be usable.

Before clean-up of the water in the containment has been completed, leakage water with very high activity will be supplied to the drain systems, and this will have to be collected and processed.

The leakage from the containment is estimated to be less than 1 $m^3/24$ hours. Since complete separation of this water from other drainage water is not generally possible, the volume of water which must be processed is estimated to be approx. 25 $m^3/24$ hours. Processing may be carried out by storing the water in the collecting tank to allow the activity to decay, by direct processing through filters and ion exchangers or evaporators, or by returning the water to the containment. The first alternative presupposes that collecting tanks with sufficient capacity and radiation shielding are available, whereas returning the water to the containment presupposes that a suitable return circuit is available or can be established.

4.2 Clean-up of the highly radioactive water in the reactor containment

As mentioned earlier, priority in the postulated situation will be assigned to filling the reactor containment with water, to safeguard the cooling of the reactor core. Only then will it be necessary to collect the radioactivity released. The initial situation will then be as follows: The containment is filled with more than 10 000 m^3 of highly active water, the fuel is still in the reactor pressure vessel and a certain amount of radioactivity is released by the leaching which is still in progress. In view of the risk of radioactivity spreading and the relatively high radiation levels around the reactor pool, the reactor pressure vessel cannot readily be opened for removing the fuel, which would enable the highly radioactive water to be drained from the containment. The dose rates on the service platform and at the reactor pool have been estimated by ASEA-ATOM. On the basis of the conditions applicable to the activity, the dose rate four weeks after the accident will be 10 mSv/h and eight weeks after the accident, 7 mSv/h. The best alternative therefore seems to be to start by cleaning the water in the reactor containment, so that the above radiation levels are preferably reduced to approx. 0.1 mSv/h, which should be acceptable from the radiation protection aspect.

In the reactors studied here, the highly radioactive water in the containment can be cleaned in the corresponding waste plant. The water can be drained from the containment to the waste plant through the containment vessel spray system. In two of the reactors studied, the water can also be returned, after clean-up, through the containment vessel spray system. In the other two reactors, no return line is available from the waste plant to the reactor containment, and the cleaned water must therefore be returned to the reactor through the turbine condenser and feed water system, and will then flow into the containment through the rupture point.

It may be assumed that clean-up of the water in the full reactor containment will be carried out so slowly that it is reasonable to expect complete mixing of the water in the containment. The water volume will therefore have to be cleaned several times, in order to obtain a condition of equilibrium with the leaching in the core and in the radioactivity settled and adsorbed in the containment. Up to 100 000 m^3 of water may therefore have to be processed. This volume is vastly in excess of the existing and conceivable tank storage capacity. Appreciable water quantities must also be processed before the water is clean enough to enable the fuel to be removed from the pressure vessel (see below).

Clean-up of the highly active water in the containment can be carried out by evaporation or by filtration and ion exchange.

Evaporation is considered by all participating power utilities to be a usable method. However, the capacity is low (1 - 5 m^3/h), and filtration and ion exchange must therefore be the principal clean-up method.

One circuit in the waste plant can normally be allocated to filtration and ion exchange of the water from the containment. The nominal capacity is typically 10 kg/s or 36 m^3/h. The usable capacity is lower due to the necessary interruptions for back-washing, precoating and filter service.

This particular water has an appreciably higher specific activity than the water normally treated in a waste plant. It is therefore of interest to know what the limiting factor is for activity levels that can be handled in the waste plant and the highest specific activity levels that can be handled.

Radioactivity in ionic form can always be handled and should not therefore cause any problems. The radiation levels from the tanks and pipe systems that are expected to be used are relatively moderate. However, one power utility specifies 10^8 Bq/kg as a limit value.

Radioactive material in suspension may give rise to problems of deposits and, in the worst case, clogging of the systems. However, the handling of highly active ion exchange resins in the waste plants has provided a background of experience that can be applied to the studied case.

Powdered and bead resins are subject to limitations (long-term stability, risk of swelling, handling prior to terminal storage, etc.), and one power utility now applies an approximate limit value of 20 GBq/kg for the specific activity of the filter materials and ion exchange resins. It is suggested that this value should be adopted for this particular case. Judging from experience, this criterion is also fully acceptable from the service aspect. Service requirements that may be of interest are primarily expected on the filters.

Several factors must be taken into account in the assessment of how quickly after an hypothetic accident the activity in the containment can be collected and processed. As mentioned earlier, most of the attention will be devoted to cooling the reactor immediately after the accident, and priority will probably be given to filling the containment with water. The containment must be vented before it can be filled with water. In view of the activity emission that this would involve, it is probable that venting will be delayed for a couple of weeks. Filling of the reactor containment is expected to be completed about four weeks after the accident. Factors such as the need for planning the handling, drawing up the strategy and possible criteria, as well as the benefits of allowing short-lived activity to decay, also favour the handling not being started until about four weeks after the accident.

With one exception, none of the participants in the study could see any reason for modifying the normal method of operation of the waste plant, to take into account the high activity levels. On the contrary, it was pointed out that any deviation from the normal method of operation increases the risk of disturbances and may give rise to unforeseen complications. The exception is a proposal for employing transportable filters for clean-up of the water from the reactor containment. It is uncertain whether or not this would be beneficial. Although the conditions available for cleaning the water by means of the existing equipment obviously differ somewhat from one station to the next, the difference is not of a decisive nature.

The strategy for handling the activity from the water-filled reactor containment which appears to be best suited is as follows: Water is transferred in batches to the waste plant, through the emptying pipe of the condensation pool. Evaporator equipment and filters and ion exchangers are used for water clean-up. In the latter case, it may be sufficient to use only filters coated with ion exchange resin in powder form.

The operating times of the filters and evaporator should be adjusted so that the concentration of activity obtained will allow for further handling and final disposal of the waste. The specific activity of the water drained from the containment is initially high, and the filter operating time must therefore be restricted to approx. 15 minutes, so that the limit value of 20 GBq/kg mentioned earlier will not be exceeded. In view of this short operating time, a disproportionately large part of the time will be allocated to back-washing and pre-coating of the filter. As a result, the clean-up capacity will initially be only a fraction of the normal filter capacity. Clean-up of the water in the containment will therefore take a long time. In the introductory stages, the average filter capacity is estimated to be less than 10 m^3/h. But as clean-up of the water continues, it will be possible to increase the operating time of the filters, so that the filtration rate will gradually approach the normal filter capacity. This is illustrated in Appendix 1.

Due to the short filter operating times, three-shift operation will be necessary to keep the clean-up process running. The opportunities available for three-shift operation are expected to be relatively good, since other personnel can quickly be trained to operate the waste plant. Increased manpower in the waste plant is also important from the occupational radiation exposure aspect in the situation studied here.

The time necessary for clean-up of the water to reduce the radiation levels around the reactor pool to approx. 0.1 mSv/h, is theoretically estimated to be between 3 and 4 months, depending on factors such as whether the plant is operated on two - shifts or three - shifts and provided that no operational problems arise in the waste plant. To achieve this stage of clean-up, between $4 \cdot 10^4$ and $5 \cdot 10^4$ m^3 of water must be treated. It may thus be assumed that, from the radiation aspect, removal of the fuel could start within 6 months after the accident.

Another consequence of the short filter operating times, which is also illustrated in Appendix 1, is that the solidification system will not be able to keep pace with the rate at which filter materials and ion exchange resins become available for disposal, and their buffer storage will therefore be necessary. In some of the waste plants, limited storage capacity for resins may possibly form a bottle-neck in the handling of the released activity. However, planned additions will improve the situation in this respect.

50 GBq/kg is specified as the maximum activity level for cement solidification of evaporator concentrate. This limit value is based on the maximum permissible surface dose rate on the cement moulds and on the minimum concentrate volume which can practically be handled in the solidification system. Good opportunities are available throughout for storing the evaporator concentrate in tanks while awaiting solidification.

Every power station normally stores relatively large quantities of fresh filter materials and ion exchange resins (powdered and beads). Moreover, the delivery time for these materials is short, i.e. about 6 weeks. Shortage of these materials will therefore not delay the handling of the activity.

Inorganic ion exchangers may possibly also be of interest. Since these are not as sensitive as conventional ion exchange resins to radiation-induced decomposition, the radioactivity could be concentrated even further. The handling would then be facilitated, since no risk would be involved of the solidification capacity being insufficient and since the attainable filter operating times would be longer.

The fast build-up of activity in filters and ion exchangers during the first stage of clean-up of the water from the reactor containment vessel will make it desirable to supervise the process. Permanently mounted equipment for this purpose is not available, although it is considered that supervision would be relatively simple to arrange.

4.3 Handling of filter materials and ion exchange resins

The waste plants at the power stations participating in the study normally carry out terminal treatment of about 10 m^3 of granular ion exchange resin (bead resin) and 50 -100 m^3 of settled ion exchange resin in powder form every year. At two power stations, cement is used as the solidification matrix, and the waste is solidified into concrete moulds with an external edge length of 1200 mm and varying wall thicknesses. At the other two power stations, the waste is dried and is then mixed with bitumen. The mixture is subsequently emptied into sheet steel drums with a volume of 200 litres.

Evaporator concentrate has not been solidified to any significant extent.

Varying values are specified by the different power utilities for the maximum activity level of the processed resins. Between 1 and 20 GBq/kg of settled resin is specified. The different values may partially be due to the activity being measured after different decay times and may also be dependent on the nuclide composition of the activity. Normal activity levels are a factor of 10 lower than the maximum.

During the postulated clean-up of the highly active water from the reactor containment, waste will be produced at a higher rate than that which can be processed in each solidification system (see Appendix 1). The available tank storage capacity for filter materials and ion exchange resins may therefore be a limiting factor for the removal of radioactivity from the reactor containment vessel. It is estimated that about 100 m^3 of filter materials and ion exchange resins will be produced. The storage capacity for filter materials (not only low-level) in the plants studied here varies at present between 30 and 160 m^3 and the capacity for granular resins between 15 and approx. 250 m^3. A certain expansion of the storage capacity is being planned at a couple of the power stations. Only one of the power stations currently stores major quantities of filter materials and ion exchange resins.

At the present time, the normal capacity of the solidification systems on one-shift operation is 1 - 2 waste packages daily. The amount of activity in a package should be limited to approx. 1 TBq in concrete moulds with a wall thickness of 10 cm, and to approx. 0.8 TBq in steel drums containing bitumenized waste. The radioactivity is assumed here to consist of the released cesium activity after a few months, i.e. approx. 50% each of Cs-134 and Cs-137. Concrete moulds with a wall thickness of 25 cm can accommodate four times as much activity without the surface dose rate on the mould exceeding 30 mSv/h, which is the upper limit applicable to these waste moulds. For steel drums containing bitumenized waste, one power utility specifies the permissible surface dose rate as 1 Sv/h, which leads to the above mentioned quantity of activity.

ASEA-ATOM has also estimated the corresponding quantities of activity per waste package for the postulated release of activity. At the activity composition 8 weeks after the accident the activity load is approx. 2 TBq for a concrete mould with a 10 cm thick wall, approx. 9 TBq for a concrete mould with a 25 cm thick wall and approx. 1.5 TBq for the steel drums. In this case, 1 TBq of activity corresponds to approx. 10 m^3 of untreated water from the reactor containment or the activity collected in the filter during about 15 minutes. A significant increase in the solidification capacity should be possible by operation on several shifts and by various provisional measures. However, the matter has not been investigated in detail.

The factors that make it desirable to restrict the specific activity in a waste package are:

- Demands associated with handling and storage

- Occupational radiation exposure during service and repair work

- Risk of long-term dose damage to the waste form

According to the utilities, if only the handling and storage of concrete moulds are taken into account, the above limit values could be raised by a factor of up to 10. However, it should be more realistic to assume a factor of only 2 to 3. The situation is somewhat less clear concerning the other two factors.

On the basis of the above assumptions for the processing of the waste, the number of waste packages may be estimated to be between 500 and 700 thin-wall (10 cm) concrete moulds or 700 to 1000 steel drums.

4.4 Radiation protection aspects

Due to the radioactivity released, the access to certain areas of the reactor building may be expected to be restricted, as a result of the high radiation levels and the airborne radioactivity. It is primarily rooms in which systems 322, 323 and 352 are installed that access will be highly restricted due to high radiation levels during the first weeks following occurrence of the accident assumed.

The elevated radiation levels here and in the waste plant are generally acceptable, provided that the access routines are modified to suit. Whether or not elevated radiation levels are acceptable is entirely dependent on the type of work which is to be carried out, the time necessary, the degree of priority and, above all, the availability of qualified manpower. A couple of power utilities specify 1 mSv/h as an acceptable general level.

The emergency ventilation system will start automatically after the accident. The main functions of the emergency ventilation system are to prevent emission at ground level and to filter the exhaust air from the reactor building. However, since the supply air is shut off and the exhaust air flow is substantially reduced, the airborne activity conditions in the reactor building will deteriorate appreciably, and the access may therefore be restricted.

The management of the radioactivity released will obviously be affected by these factors. Delays will probably result, although not of such magnitude that they will be decisive to the timing. It is considered that this restricted accessibility should be acceptable for a certain period of time. The rounds by the operating personnel and maintenance work are some of the activities that can be restricted to a minimum.

According to two of the power utilities routine work is responsible for most of the collective dose received by the personnel in the waste plant, whereas the other two state that disturbances and repairs are most dose-intensive. The normal annual occupational exposure in a plant is 0.01 manSv. Surface contamination in the plants is normally responsible for an insignificant part of the dose.

In order to reduce the radiation exposure, the power utilities specify conventional measures, such as radiation protection supervision, decontamination, radiation shielding, work planning and documentation and, in one case, also modifications to the system design. The measures are assigned different degrees of priority at the various power stations.

According to a couple of the power utilities, if large quantities of high-level radioactive filter materials and ion exchange resins must be stored for an extended period of time in tanks in the waste plant, substantial additional radiation shields will be necessary. In addition, the access restrictions may give rise to difficulties.

Only one of the power utilities was willing to hazard a guess concerning the occupational exposure for the entire work of disposing of the waste. The answer was approx. 4 manSv, half of which is estimated to stem from the decontaminaton of the containment after the highly radioactive water has been cleaned and disposed of. The radiation exposure in the waste plant is expected to be less than 1 manSv. It should also be pointed out that, if clean-up of the containment water is not pursued to the point assumed in section 4.2 before the fuel is removed, the dose for this work may be significant (assuming no more than 1 mSv/h around the reactor pool, the dose may be of the order of 1 manSv).

Several of the power utilities state that problems may arise with the dose limitations for the personnel, and personnel shortage may therefore arise. The power utilities consider that the handling of the waste cannot be regarded as special work for which the individual dose may exceed the normal limit value of 50 mSv/year. Exceptions may be allowed for acute situations, such as work for safeguarding the cooling of the core.

4.5 Capacity of the waste plant

One of the objectives of this project is to identify possible limitations of the various waste plants which would render the handling of the radioactivity released during the hypothetic accident more difficult or would delay it.

The demands made on the waste plant in the postulated situation are obviously heavier than during normal reactor operation. One of the difficulties is that the specific activity of the water is initially very high. In view of the limitations which must be taken into account in the concentration of activity in filter materials and ion exchange resins, the ordinary water treatment capacity will lead to a correspondingly high production of spent resins. The solidification system is sized for the spent resin occurring when the water activity is normal, and the system will not be able to keep pace with the water clean-up after an accident. The storage tanks available between the water clean-up systems and the solidification system will therefore start to become filled. If the tanks are too small in relation to the quantity of radioactivity released and waste produced, the capacity of the solidification system will eventually limit the rate at which water can be treated.

More normal activity levels of the water processed in the waste plant can be expected at a later stage of the recommissioning work. The solidification system will then have excess capacity in relation to the water clean-up system, and the capacity will be limited by the latter. This situation may be expected when clean-up of the water in the reactor containment has been in progress for some time, and during the subsequent decontamination stage.

In summary, it may be said that the study has not highlighted any special weak link that may restrict the handling of the activity. This may be regarded as a consequence of the loading on the waste systems deviating quantitatively rather than qualitatively from the situation which is normal and for which the waste plant should be optimized.

5. HANDLING OF THE RADIOACTIVE WASTE WHEN A COOLING CHANNEL IS BLOCKED

5.1 Measures to be taken in the reactor plant

An alarm for high activity in the off-gas system will be initiated a few minutes after the release of radioactivity has started. This will lead to the steam pipe and stack activity being checked. If elevated steam pipe activity is established, the measurement room personnel will be contacted for carrying out check measurements. If the steam pipe activity should exceed the action level, the power will be reduced by diminishing the speed of the recirculation pumps. In reactors having only about 30 minutes of dealy time in the off-gas system, an increase in the activity emission from the stack will occur after this period of time. The control room personnel will then presumably initiate scram and isolation (I or A isolation, i.e. isolation of the containment or separation of the reactor from the turbine). In reactors equipped with advanced off-gas systems, the stack activity will be affected to an appreciably smaller extent, and the accident may conceivably lead to a slower reduction in power, without isolation.

The actions taken immediately after scram and shut-down are the ordinary operating measures for cold shut-down and actions intended to restrict the emission of radioactivity, such as isolation of the reactor building and filtration of its exhaust air. The ordinary actions include re-establishment of reactor water clean-up if this has been interrupted by the shut-down. The possible consequences of this action are discussed in Chapter 6 below.

Even if the reactor water clean-up system is in operation only during the first 30 minutes after the accident, the ion exchange resin will be very radioactive. Any problems that this may cause are not very familiar to the operating personnel and the problems are not discussed in the accident management procedures. It is uncertain when during the course of the accident the high activity will be noticed and how it will be noticed. Ion exchange resin with a high activity load in the reactor water clean-up system is normally not a problem. Only one of the power utilities has had reason in conjunction with fuel damage to restrict the operating time of the ion exchangers as a result of a high activity load.

A blocked cooling channel leads to appreciably more extensive spreading of radioactivity in the plant than an internal pipe rupture. The quantities of radioactive leakage water discharged to the waste plant through the drain systems will thus increase. One power utility estimates the leakage to amount to 25 m^3/24 hours of highly radioactive water from the reactor plant and an equal amount of water of lower activity from the turbine plant. Another power utility states that the total leakage from the plant is normally less than 100 m^3/24 hours and points out that the leakage from the reactor containment in accordance with the regulations is restricted to a maximum of 1 kg/s (86 m^3/24 hours). The former figures can probably be regarded as the normally expected values, but this does not exclude the possibility that the leakage may be higher in one of the plants.

The radioactivity spread with the steam contaminates water in the condenser, condensate system, feed water system, etc. The total volume of water in the systems amounts to 300 -500 m^3. The water can be cleaned in the condensate clean-up plant.

The radioactivity collected in the condensate clean-up filters before the plant is shut down is not regarded as a major problem. The surface dose rates on the filters will be lower than those occurring today on the spent resin storage tanks.

The release of radioactivity during the hypothetic accident will restrict the access to certain areas for a period of time. This will make it difficult to handle the activity released. The same comments are generally applicable as those made on the case of minor fuel damage and internal pipe rupture (see section 4.4).

As compared to this case, a blocked cooling channel will result in highly restricted access during the first weeks after the accident to areas containing systems 321, 331 and 352. During the initial period, access will also be restricted to the areas around the turbine condenser and system 354.

Manual actions will be necessary during work on the ion exchangers of system 331, and this work is expected to be dose-intensive. As a result, use of the reactor water clean-up system may initially be restricted.

5.2 Handling of the released activity

Most of the radioactive material, particularly that which is long-lived, will accumulate in the reactor pressure vessel and the ion exchanger of the reactor water clean-up system. The radioactivity which will be spread to the turbine plant and the condensation pool is principally short-lived (noble gases and iodine) and will place minor demands on collection and processing.

Two different methods are suggested for the handling of the ionic and suspended particulate activity in the reactor pressure vessel. One of the methods involves cleaning the water in the ion exchangers of the reactor water clean-up system, and the spent ion exchange resins are then transferred to the waste plant. In the second method, the contaminated water is transferred from the reactor, e.g. through the reactor water clean-up system, to the waste plant for filtration and ion exchange or evaporation.

Certain potential problems appear to be associated with clean-up of the water in the reactor water clean-up system. In the same manner as mentioned above in conjunction with the pipe rupture case, the specific activity of the ion exchange resin must be restricted to a certain maximum value. If the limit value of 20 GBq/kg employed by one power utility is assumed, the operating time of the ion exchanger will initially be only a few minutes and the activity will distribute itself very unevenly in the resin bed. Up to 50 batches of ion exchange resin may be necessary for collecting all of the radioactivity. As a result, the probability cannot be ignored of the pipes becoming clogged during one of the occasions when the resin is transferred to the waste plant. Since the spent resin produced is greatly in excess of the solidification capacity, storage facilities must also be provided for the spent resin. At the present time, this would only be possible at one of the power stations included in the study.

If the water in the reactor pressure vessel is instead transferred to the waste plant before cleaning, better flexibility of processing will be achieved. The water can be stored in untreated condition, processing can take place in different circuits, the water can be diluted before filtration (to achieve a reasonable operating time of the filter), evaporation can be employed as a method of processing, etc.

A combination of the two methods mentioned above can obviously also be used for processing the activity.

The resin in the ion exchanger of the reactor water clean-up system which is assumed to be in operation during the first 30 minutes following the accident will be very highly active. It is suggested that the resin should be left in the ion exchanger for an extended period of time, to enable the activity to decay. The iodine collected and also the noble gas activity produced will then have time to decay. After about 12 weeks of decay time, the activity inventory will be at the same level as that which has previously occurred in conjunction with fuel damage. Handling can then be carried out in a conventional manner.

The number of waste packages is estimated to be about the same as in the case of minor fuel damage and internal pipe rupture.

5.3 Handling of damaged fuel and fuel fragments

The handling of the damaged fuel and fuel fragments is probably the most difficult problem in this case.

The damaged fuel assembly must be removed by means of some special tool. The time necessary for removing the fuel assembly itself is estimated to be about 24 hours, provided that the fuel box is intact. One power utility has estimated that it would take 22 days for the entire work, including preparations, such as removal of the fuel from the core and extraction of the sludge from the fuel assembly support plate.

A large proportion of the damaged fuel will probably end up at the bottom of the reactor pressure vessel in the form of fragments. Some form of remotely controlled special equipment is suggested for removal of the sludge.

A small amount (approx. 20 g) of fuel residues in the form of fine particles below 40 μm in diameter will be transferred to systems 321 and 331. These finely dispersed fuel particles can be collected in the ion exchangers of system 331 by flushing the systems.

The waste occurring, with the exception of the small quantity collected in the ion exchangers, will be collected into storage boxes, which will temporarily be stored in the fuel pools.

Only one of the power utilities has estimated the personnel dose that could be obtained from the handling of the damaged fuel. Cleaning of the systems will give rise to 0.2 -0.3 manSv, whereas the remainder of the work will not contribute any significant dose.

5.4 Capacity of the waste plant

In the case of the blocked cooling channel, the release of radioactivity will be roughly of the same magnitude as that caused by an internal pipe rupture. If the handling of the radioactivity is pursued in the same manner in both cases, the limitations will therefore be similar (see Chapter 4).

However, in the case of the blocked cooling channel, most of the radioactivity will be concentrated to only about 200 m^3 of water. As a result, storage in tanks of non-reprocessed water and evaporation will be realistic alternatives to filtration and ion exchange. The opportunities for handling the radioactivity released in the case of a blocked cooling channel will thus be appreciably better than in the case of pipe rupture, and the study carried out does not indicate any limitations in the capacity of the waste plant, of such a magnitude that it would significantly delay the restarting of the reactor.

6. CONCLUSIONS AND COMMENTS

6.1 General remarks about the study

Certain conditions for the work should be known before any conclusions are drawn from the material presented in chapters 4 and 5.

The accident excercises undertaken by the operations personnel at the reactors are designed to provide training in the protection of the environment, and therefore normally terminate when core cooling has been assured. Re-commissioning work, such as the handling of waste, is not dealt with. The personnel is therefore not very well prepared for answering questions of this nature. Also, the time that the operating personnel has been able to allocate to the study was limited and has not allowed for any investigative activities. Against this background, some of the assessments made must be regarded as doubtful. This naturally applies in particular to power utilities that have modest experience of fuel damage or none at all.

As a consequence of the above, the study carried out has also had an secondary objective, i.e. to improve the knowledge and consciousness of the operations personnel concerning the questions discussed. From this aspect, the study must be regarded as being successful.

One of the questions discussed when selecting the two events as a basis for the study is their credibility. As regards the quantity of contaminated water, the two cases represent conceivable extremes: the water volume in the reactor pressure vessel and the water volume in the entirely filled reactor containment vessel. Furthermore, since the two hypothetic accidents lead to radioactivity release of the same magnitude, the study should provide a good idea of how the volume of contaminated water affects the handling and processing. On the other hand, as regards the quantity of activity released, the study represents basically only one point - fairly probable though it may be - in a broad spectrum. If the quantity of radioactivity released is a vital parameter, caution should therefore be excercised when drawing conclusions.

6.2 Conclusions

As mentioned earlier, one of the objectives in the project was to identify any limitations in individual waste plants that could make it difficult to collect and process the radioactivity released during the accident or that could cause delays. No weak point was found in the waste plants which would limit the treatment and conditioning of the radioactive waste after the postulated fuel damage. However, the answers to the questions given to the operating personnel indicated that some problems may exist at certain plants. This findings will have to be examined further.

The points on which the study indicates certain weaknesses are as follows:

- Opportunities available for handling highly radioactive leakage water.

The waste plant should be capable of receiving, say, 100 m^3 of highly radioactive leakage water per day. It cannot be regarded as verified today that this is possible neither that the requirement is unnecessary.

- Opportunities available for clean-up and storage of radioactive water inside the containment.

The reactor containment vessel contains large quantities of water that may be contaminated in the event of an accident. It would therefore be desirable to provide facilities for clean-up of the water inside the containment by circulation through the waste plant. The containment is also the only well-shielded storage place for large quantities of radioactive water. It therefore ought to be possible to transfer water to the containment, as back-up for clean-up in the waste plant.

- Possibilities to follow the radioactivity build-up in filters and ion exchangers.

Earlier during this study, the assumption has been made that, in the case of a blocked cooling channel, the operating time for the ion exchangers in the reactor water clean-up system will be 30 minutes from the beginning of the accident. During this time, approx. 20% of the activity remaining in the reactor pressure vessel will be collected in the ion exchanger. This will cause the radiation dose to the ion exchange resin to quickly become of such magnitude that the chemical and possibly also mechanical properties of the ion exchange resin will be affected.

The value of 1 MGy (100 Mrad), which will occur roughly one month after the accident, can probably be regarded as an approximate threshold value, above which it may become difficult to discharge the exchanger bed in system 331 (See section 3.5). Less than 6 months after the accident, the dose will have doubled. If the operating time of the ion exchanger is extended beyond the assumed 30 minutes following the accident, the dose will increase more rapidly. If it remains in operation for a whole hour, the value of 1 MGy will be reached within one week.

Judging by the comments of the power utilities on the assumption that the ion exchanger will be isolated after 30 minutes, it would appear that the accident management procedures are such that the personnel will endeavour to keep the ion exchanger in operation. An additional ion exchanger may possibly be taken into operation in order to boost the clean-up of the reactor water during shut-down, which normally gives rise to a chemical transient, implying among other things a large release of radioactive material to the cooling water.

For several reasons, it is considered desirable to allow very highly active ion exchange resin in the reactor water clean-up system to decay in place, before it is transferred to the waste plant. Activity levels that may affect the mechanical properties of the ion exchange resin during this time must then be avoided.

- Opportunities available for providing temporary storage of filter materials and ion exchange resins.

During clean-up of highly active water, the limitations of the specific activity of filter materials and ion exchange resins will lead to large quantities of spent resins being produced.

In the present-day situation, efforts of a training nature at the power stations are the measure that would provide the greatest long-term benefits in the handling and processing of the waste after a reactor accident. A review of accident conditions such as those studied here is a desirable element in the training. Greater consciousness of the complex of problems should contribute to identification and correction of any weaknesses in the systems and equipment. Another important measure is to review the instructions that are relevant in this respect. This is illustrated below by the rules for taking the ion exchangers in system 331 into and out of operation following scram.

The reply from one of the utilities includes a warning of the risk of over-estimating the capacity and under-estimating technology. This is undoubtedly a sensible viewpoint in the preparations for a situation for which the conditions can hardly be defined. The study also indicates that flexibility is an important quality - difficult though it may be to define - of a waste plant and its interaction with the reactor. The existence of many different facilities for transferring water, filter materials and ion exchange resins between different parts of the plant and items of equipment obviously makes it easier to allocate the resources optimally in an extraordinary situation.

6.3 Concluding comments

The study has disclosed a couple of points on which it would be valuable to shed more light. One of these is the opportunities available for employing transportable filters of TMI type for water clean-up, whereas the other is the opportunities available for increasing the amount of activity per waste package. Each of these alternatives has the potential for improving significantly the capacity for handling large quantities of radioactivity.

The capacity of the solidification system can also be increased by the provision of a mobile solidification plant or by arranging a special cover casting station adjacent to the concrete solidification station. These proposals would be worth studying in more detail in the above situations.

As mentioned above, the study covers the entire probable spectrum as regards the volume of containated water. On the other hand, the release of radioactivity is a very uncertain parameter which has been studied in detail at only one point. The capacity of the solidification system is restricted at present by the quantity of activity per waste package. But there are also other capacity-restricting limitations on the levels of activity that can be handled in the various waste plants. These limitations have so far been poorly mapped out and quantified. A couple of the power utilities point out that the problem should be tackled.

According to the principle of the "rule of three" between the collective dose and the quantity of activity handled, the operational experience obtained hitherto should provide a good basis for identifying the operations which could be problematic in an accident situation that gives rise to high activity levels. A study based on the assumption of source strengths and occupation times has already been carried out for the waste plant of Forsmark 3. One of the results of the study was that the servicing of the solidification system is responsible for more than 50% of the occupational radiation exposure.

Corresponding studies should be carried out for the other waste plants which are in operation, using measured source strengths and occupation times based on interviews with the operating personnel. One plant using cement solidification and one plant using bitumen solidification should preferably be studied initially, possibly the Ringhals and Barsebäck plants.

The suggested investigations will hopefully confirm the main conclusion from this study:

> The waste plants at Nordic nuclear power plants 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.

7. REFERENCES

1. J-E. Sandström and G. Granath
   SKI project - Management of radioactive waste resulting from nuclear fuel damage. Part 1. (In Swedish)
   Memorandum TQB 83-428

2. G. Granath
   SKI project - Management of radioactive waste resulting from nuclear fuel damage. Part 2.
   The form of the radioactivity. (In Swedish)
   Memorandum KPB 82-97

3. J. Elkert and L-G Envall
   SKI project - Management of radioactive waste resulting from nuclear fuel damage. Part 2.
   Activity and dose rate calculations. (In Swedish)
   Memorandum KPC 82-85

4. M. Ullberg
   SKI project - Management of radioactive waste resulting from nuclear fuel damage. Part 2.
   Transport of the radioactivity to the waste plant. (In Swedish)
   Memorandum KVC 82-290

5. M. Ullberg and J. Elkert
   SKI project AVF-1 - Management of radioactive waste resulting from nuclear fuel damage. Part 3.
   Questionnaire. (In Swedish)
   Memorandum KVC 83-59

6. J. Elkert and M. Ullberg
   SKI project AVF-1 - Management of radioactive waste resulting from nuclear fuel damage. Part 3.
   Summary of the replies from the power utilities, with comments. (In Swedish)
   REPORT KPC 84-22

7. H. Wijkström and J. Elkert
   SKI project AVF-1 - Management of radioactive waste resulting from nuclear fuel damage. Complement to part 3.
   Handling of fuel residues in basic case 2. Distribution outside the reactor pressure vessel and associated radiation levels. (In Swedish)
   REPORT KPA 84-382

These reports can be obtained from

ASEA-ATOM AB
Box 53
S-721 04 Västerås
Sweden

APPENDIX 1

CAPACITY OF THE WASTE PLANT AS A FUNCTION OF THE SPECIFIC ACTIVITY

An attempt is made in this appendix to illustrate graphically how the capacity of a waste plant is affected by increasing specific activity of the water and ion exchange resins. The graphs correspond to the waste plant in Oskarshamn 1 and 2. Similar graphs should be applicable to other waste plants.

Figure A1 shows the capacity of a 10 m^2 disc filter as a function of the specific activity of the water. In Oskarshamn 1 and 2, this type of filter can be connected in for recirculation clean-up of the water in the reactor containment. The conditions for the indicated relationship is that the specific activity of the filter material must be restricted to 1 TBq/50 dm^3 and that 1 h is necessary for pre-coating the filter. The assumed limitation to the specific activity of the filter material is due to the fact that 50 dm^3 is the minimum filter material volume that can be metered into one package by means of the existing equipment. The time necessary for pre-coating a filter is normally shorter than 1 h. However, at a high frequency of back-washing, the average pre-coating time should increase, since time must be allowed for settlement in the back-washing tank, etc.

Figure A2 shows how spent precoat material from a filter is produced at an increasing rate on increasing specific activity of the water. The conditions are the same as for Figure A1.

The same figure also shows the activity processing capacity of the filter. If the specific activity of the water is high, the filter capacity is restricted by the maximum number of pre-coating operations that may be done on the filter per 24 hours.

Figure A3 shows the volumetric capacity of the solidification system as a function of the specific activity of the filter material. The conditions are that two moulds are produced per day, and the limit is set at a maximum of 1 TBq/mould.

Figure A4 shows the activity capacity of the solidification system as a function of the specific activity of the filter material. The conditions are the same as for Figure A3.

APPENDIX 1

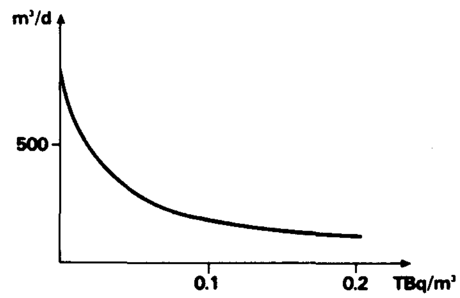


Figure A1

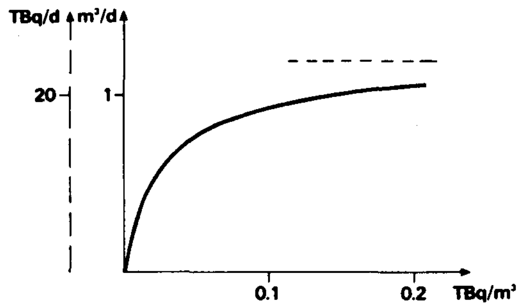


Figure A2

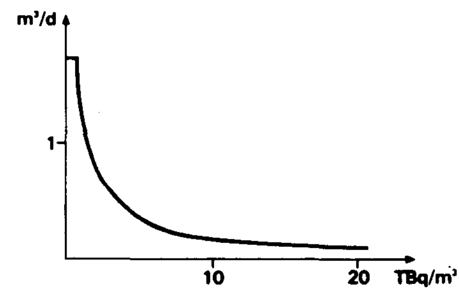


Figure A3

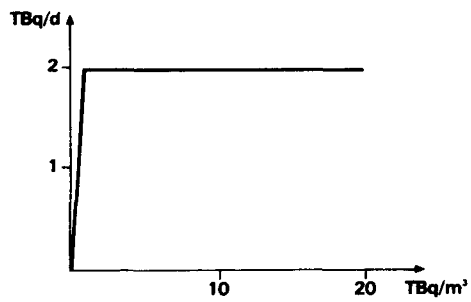


Figure A4

gotab Stockholm 1986