FILTRA

FILTERED ATMOSPHERIC VENTING OF LWR CONTAINMENTS

Progress report. March 1981

The FILTRA project is a joint undertaking between the Swedish Nuclear Power Inspectorate and Oskarshamnsverkets Kraftgrupp AB, Sydkraft AB and the Swedish State Power Board with Studsvik Energiteknik AB and Asea-Atom AB as principal contractors.

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The judgements, conclusions and recommendations expressed in the report are those of the steering group. They are not binding on the organisations cooperating in the project.

ABSTRACT

The FILTRA project is a cooperative Swedish programme which started in February 1980. It is aimed at investigating the possibility of reducing the risk for a large release of radioactivity, assuming a severe reactor accident. The project has been focussed on filtered venting of the reactor containment. The first stage of the project has dealt with two types of severe accident sequences, namely core meltdown as a result of the complete loss of water supplies to the reactor pressure vessel and insufficient cooling of the reactor containment. Some important conclusions are the following. The applicability of computer models used to describe various phenomena in the accident sequence must be scrutinized. The details of the design of the containment are important and must be taken into consideration in a more accurate manner than in previous analyses. A pressure relief area of less than 1 m^2 appears to be adequate. The following principles should guide the technical design of filtered venting systems, namely reduction of the risk for the release of those radioactive substances which could cause long term land contamination, provision for a passive function of the vent filter system during the first 24 hours and achievement of filtering capabilities which make leakages in severe accidents comparable to the leakages of radioactive substances in less severe accidents, which do not necessarily actuate the pressure relief system. Nothing indicates that a system for filtered venting of a BWR containment would have a significant negative effect on the safety within the framework of the design basis. Efforts should be directed towards designing a filtered venting system for a BWR such as Barsebaeck.

DESCRIPTORS

SWEDEN, REACTOR SAFETY, CONTAINMENT, SAFETY ENGINEERING, PRESSURE RELEASE, MELTDOWN, LOSS OF COOLANT, BWR TYPE REACTORS, FILTERS, VENTS

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SUMMARY

General background

Discussions concerning reactor safety have recently been focussed, in particular at the political level, to issues such as

- how large is the risk for a reactor accident which will result in considerable releases of radioactivity, and how are these risks assessed?
- what possibilities are there to reduce further the risks for considerable releases of radioactivity in the event of a reactor accident?
- what emergency planning for evacuation etc, is prudent, considering the risks for reactor accidents of varying severity?

Studies of reactor safety, such as the Rasmussen report (WASH-1400), showed that severe damage could occur _if_ a large part of the radioactive contents of a large nuclear power reactor core were released to the environment. For such large releases to take place it is necessary not only that serious core damage has occurred, i.e. that the accident prevention systems of the reactor have failed, but also that the safety systems for mitigating such releases have been damaged. Amongst other things the reactor containment, a thick barrier made from steel and concrete must be penetrated.

The Swedish Government Committee on Reactor Safety stressed in their recommendations for measures to improve reactor safety that there was a strong case for continued forceful efforts on measures to prevent and control accidents. In

addition, however, much more attention should be paid to measures to mitigate the consequences of an accident.

According to the Committee on Reactor Safety, efforts should in the first place be directed towards reducing the risk for releases resulting in extensive long term ground contamination. They recommended therefore an intensive R&D programme aimed at enabling a decision to be made within about two years (before the end of 1981) concerning the possibility of installing further measures to limit releases of radioactivity.

The initial plans for such a R&D programme were drawn up in the autumn of 1979. In February 1980 the Swedish Nuclear Power Inspectorate started the FILTRA project (a Swedish abbreviation of Filtered Pressure Relief) as a co-operative programme between the Oskarshamnsverkets Kraftgrupp AB, the Sydkraft AB and the Swedish State Power Board and with Asea-Atom and Studsvik Energiteknik as principal contractors for different parts of the project. The project is divided into three stages, the first of which is reported here. It consists of a general assessment of the problems involved. More comprehensive studies of the principal problems will be continued in the second stage. Finally the results of the entire project will be evaluated and presented in stage 3. The project is expected to take somewhat more than 18 months to complete.

General goals. Limitations

The FILTRA project is aimed at clarifying the possibility of considerably reducing the risk for large releases of radioactivity, assuming a severe reactor accident beyond the limits of the present design basis accidents. The project includes tentative designs of technically possible solutions as a basis for, inter alia, cost estimates.

For reasons discussed in the main body of the report, the project has been focussed on filtered venting of the reactor containment. Other consequence-mitigating measures have so far not been studied within the FILTRA project. FILTRA is a R&D project aimed principally at developing the technical and scientific background within the relevant problems areas. It is however not withing the scope of the project to recommend that a system for filtered venting should be installed or not on the basis of some overall evaluation of risks, benefits and costs. Such considerations must be made in another context. It can be noted that the Swedish Government, through the Minister of Energy, made the following statement in their bill 1980/81:90:

> "Despite the fact that existing reactor installations present an extremely small risk for uncontrolled releases of large amounts of radioactive material, which would result in radioactive ground contamination, I am of the opinion that all possibilities should be exploited to further reduce the risks for such releases. This is particularly the case for plants, such as Barsebäck, which are situated near densely populated regions. Such measures should be taken even if they involve a not insignificant cost for the owners, as seen in relation to the reduction of the release risk. Filtered venting of the reactor containments in Barsebäck should be ready to be taken into operation in 1985 at the latest, or at the next following revision. It rests with the Swedish Government to issue further directives in this matter. Filtered

> venting of the reactor containments in Ringhals, Oskarshamn and Forsmark may also come under consideration. It is however important that the experiences from Barsebäck, and the technical developments, which are under way within this field, are taken into account. Directives considered in other countries using nuclear power, such as USA, concerning measures to reduce the risks for radioactive ground contamination, should be taken into account, if possible, when specifying the requirements for the later reactors. If continuing research shows that other methods than filtered venting of the reactor containments give comparable reductions in the risk for large releases of radioactive material, or if the present risk assessment for accidents which result in large releases of radioactive material is considerably changed the safety requirements for the nuclear power plants in Ringhals, Oskarshamn and Forsmark should be adjusted accordingly. The necessary decisions in this matter should be taken at such a time that the resulting measures are completed by 1989.

The statement cited above demands concentrated efforts under heavy time pressure, within the FILTRA project.

Technical background to the FILTRA project

WASH-1400 primarily specified two sequences of events which were assumed might result in very large release of radioactivity in connection with severe reactor accidents. One type of sequence involved steam explosions in connection with a core meltdown. It was assumed that such explosions under some circumstances might become powerful enough to cause serious damage to the reactor pressure vessel and containment. The other type of sequence involved the complete loss of all cooling systems (heat sinks) within the containment, resulting in overpressure failure.

A survey study made by SANDIA Co, USA, pointed out the possibility of greatly reducing the risks for large releases, caused by overpressure failure of the containment, by complementing current containment designs with venting devices either through a filter to the atmosphere or to a large additional containment volume.

The SANDIA study was based entirely on the models for accident and release sequences described in WASH-1400. The Swedish Government Committee on Reactor Safety recommendations concerning more detailed studies of measures to mitigate releases refers to this American study, as well as complementary Swedish studies commissioned by the committee.

Against this background the FILTRA project was focussed on primarily the following problem areas;

- Studies of temperature and pressure changes in the containment for sequences of events not covered by current design basis accidents. The studies have primarily concerned sequences putting maximum strain on the containment - not necessarily the most probable sequences. (Principal contractor: Studsvik Energiteknik.)
- Studies of the transport and removal of radioactive substances by means of various filter designs. (Principal contractor: Studsvik Energiteknik.)
- Studies of various designs for filtered venting, including condenser and filter functions. (Principal contractors: Asea-Atom.)
- Analyses of how filtered venting systems would affect other safety functions. (Principal contractor: Asea-Atom.)

Work within the FILTRA project has been directed mainly towards accident sequences and venting systems for Swedish boiling water reactors (BWR). The work has been carried out in close co-operation with American and German consultants and research workers.

Similar studies in the USA have primarily concerned pressurized water reactors. The american work in this field is being followed closely.

In this report the work status after the first stage of the FILTRA project is summarized for the various problem areas listed above.

Pressure and temperature changes in the containment

Steam explosions: The stresses on the reactor pressure vessel and containment as a result of steam explosions associated with a core meltdown were studied during the autumn of 1980 by the Swedish Government Committee on Steam Explosions. The technical background material for the committee was compiled in close co-operation with the FILTRA project. Considering the available scientific evidence the committee concluded that they had not found any descriptions of sequences of events, based on more thorough technical and physical analyses, according to which steam explosions in connection with reactor accidents, could reach such a strength that they would result in failure of the pressure vessel and the containment. Therefore, according to the committee, steam explosions and associated release sequences need not be considered specifically when designing safety systems and in emergency planning. Studies of less powerful steam explosions which can occur as a result of a core meltdown should however be continued.

The FILTRA project concurs with the conclusions and recommendations of the Committee on Steam Explosions and has based the continued project work on them.

Other events which result in pressure increases:

In phase 1 of the FILTRA project two types of severe accident sequences have been studied which may under certain conditions result in overpressure failure of containments of the type used in Swedish BWRs.

These two sequences are:

- Core meltdown as a result of the complete loss of water supplies to the reactor pressure vessel.
- Insufficient cooling of the reactor containment.

Some important conclusions from the first phase are:

- The computer models used to describe various parts of the accident sequence must be analysed critically in order to interprete the results correctly and be able to judge if, for example, certain rapid increases in pressure have a realistic physical basis.
- The design details of the containment, such as the positions of doors and other penetrations, drainage of the intermediate floor, etc are very important and must be taken into consideration in a different manner than in previous analyses.
- A pressure relief are of 1 m^2 appears to be sufficient to control the pressure increases studied up to now, and which without being relieved would result in failure of the containment.

It has also been noted in the FILTRA project that the pressure relief systems which have been studied can improve the safety margin for some other pressure increase sequences which do not necessarily lead to core degradation. These sequences, which include incomplete pressure suppression function, are currently being studied in another context.

Studies of pressure increase sequences other than steam explosions in PWRs have not progressed so far in phase 1 as for BWRs. In this area, the results of the American studies are awaited with interest.

Transport and removal of radioactive substances

The same models and programs were used as in WASH-1400 and in the German reactor safety study to study transport and removal of radioactive substances in the containment and in different filter and condenser designs. These models only take into account some simple removal processes, such as the comparatively slow binding of molecular iodine in a water solution, and the gravitational sedimentation of metallic aerosols. The models have been adapted for Swedish reactor containments and the tentative condenser and filter designs. The results of the initial studies in phase 1 show that, with the tentative filter designs, it should be possible to achieve an appreciable removal efficiency for those radioactive substances which cause long term ground contamination: primarily iodine and cesium as well as other metallic aerosols, merely by taking simple physical and chemical separation processes of the sort mentioned above into account.

The models used for calculation in WASH-1400 reflect the state of knowledge in 1975. The models were in the first instance aimed at trying to estimate maximum releases for various accident sequences making relatively pessimistic assumptions.

In the Autumn of 1980 several critical reviews were published of the models which had been used in WASH-1400 and other studies to determine the release and transport of radioactive material during an accident involving severe core damage. These reviews are based on both experimental and theoretical evidence, partly from the TMI-2 accident, and partly from a number of intentional and unintentional core degradation events in smaller research reactors. The results of such reviews have to date indicated that in accidents under wet conditions (in the presence of water or steam) primarily the amounts of iodine, released to the containment, and possibly to the atmosphere, are much lower than assumed in WASH-1400. The following additional removal processes are pointed out:

- Iodine forms iodides, which are effectively bound in a water solution.
- Cesium and other metals can also form water-soluble salts, for example cesium iodide which is bound very effectively in water.
- The condensation of steam on the aerosol particles and other processes can under some conditions result in a more rapid and more effective removal than has been assumed previously.

In the USA the NRC has announced that these questions will be evaluated further in the near future.

That evaluation can be expected to be of great importance in the preparation of possible new American regulations concerning measures to mitigate releases of radioactivity and in relation to emergency planning.

Possible designs for filtered venting systems

The work during phase 1 has shown that the following principles should guide the technical design of filtered venting systems:

- The goal is a considerable reduction of the risk for the release of radioactive substances which can cause long term ground contamination assuming that a severe reactor accident has occurred.
- There should be a minimum of interaction with existing safety systems and their functions during accident sequences covered by current design basis accidents.
- The system should function passively during the first 24 hours or more of the accident sequence - then more credit can successively be taken for preplanned active measures.
- The required removal efficiency should not be higher than corresponding to the risk for leakage of radioactive substances in less severe accident sequences which do not necessarily trigger venting of the containment.
- Diversified solutions to the condenser and filtration functions should be studied, partly in order to be able to take advantage of different types of removal processes.

The BWR design studies during phase 1 have shown that a suitable solution would be to connect the systems for filtered venting of the service dome on the BWR containment. This would constitute a suitably positioned and sufficiently large pressure relief opening. Via a rupture disc and an isolation valve (normally in the open position) the venting tube would pass into the condensation and filter chamber, which can be built underground.

The rupture disc would be designed to open at a pressure which is higher than the design pressure for the containment (about 0.5 MPa) but below the pressure at which, according to the design calculations, the integrity of the containment would be markedly affected (about 0.65 MPa). This design has minimal effects on existing safety systems and their function within the limits of current design basis accidents. Further, it should be possible to carry out the necessary construction and installation work without closing down the plant for longer periods than the usual annual revision periods.

For PWR, preliminary studies have shown that there are relatively good possibilities to connect venting systems to containments such as those in Ringhals. One alternative is to make the connection via the lock for handling bulky components. The venting processes appear to be considerably more complicated for PWRs, partly because of the interaction with other safety systems.

Different technical solutions have been studied in phase 1 for the condensation and filter functions. For a BWR plant such as Barsebäck a crushed rock condenser with a volume of 10 000 m^3, followed by a scrubber pool of 100 - 200 m^3 at present appears to provide a technically well balanced solution (the crushed rock condenser would thus in this case have a comparable volume to that of the containment itself). Larger condenser volumes have also been studied, but these have been considered to be uninteresting in view of current knowledge.

The influence on other safety systems

The technical solution outlined for filtered venting will have little influence on the functions of other safety systems, at any rate for Swedish BWRs. Preliminary analyses have been made with regard to the consequences of a malfunctioning of the venting system, for example unwarranted triggering.

The analyses performed to date for a BWR show that the introduction of a filtered venting system would have very little effect on the safety for such events which are included in the safety reports on which the operating licenses for the existing plants are based.

Similar studies for a PWR indicate that the possible system interactions are more complicated for such reactors, and it would thus be premature to make any conclusions.

Summarizing conclusions of phase 1

The most important conclusions which can be drawn from phase 1, with regard to continued work, can be summarized as follows:

- On the basis of the results to date it should be possible to find technical solutions which will greatly reduce the risk for reactor containment failure caused by overpressure in connection with some types of severe accident sequences outside the boundaries of current design basis accidents.

- It should be possible to equip the venting system with a filter to contain most of the radioactive substances wich can give rise to ground contamination (in the first instance iodine and cesium) even if most of the core contents of such elements were released to the containment atmosphere in the event of a severe accident.

- Nothing indicates that a system for filtered venting of a BWR containment would have a significant negative effect on the safety within the framework of the design basis accidents currently defined. In the case of PWR the results of more detailed studies must be awaited.

- More recent research results indicate that the risk for considerable releases of radioactive substances in connection with certain accident sequences can have been overestimated in previous safety studies. There are however still large areas in which there is a serious lack of knowledge concerning the physical and chemical processes which control the release, transport and retention of radioactive substances in reactor systems, containments, and any additional filter systems in the event of a severe accident including core degradation.

The direction of continued work

With regard to the direction of the continued FILTRA project the following conclusions can be drawn from phase 1:

- Efforts should primarily be aimed at describing a technically well balanced design for a filtered venting system for a BWR such as Barsebäck. More important factors which can influence the corresponding design for the containments in BWR plants such as Forsmark and for PWRs should be identified and analysed.

- Continued and more detailed studies of the different accident sequences are of great importance to obtain better knowledge of the factors which should determine the dimensioning of a filtered venting system. This applies both to heat transfer and flow rates, and such technical factors as how much of the various radioactive substances will be released to the containment and filter in the event of an accident involving severe core damage.

- More detailed studies of the retention processes in different sorts of filter designs are of great importance for enabling the choice of a technically well balanced and functionally well verified filter design.

- Semi-scale experiments should be carried out to verify, in the first instance, the heat transfer and flow rates in various designs.

Looking at the present situation, it is particularly important to get more detailed and realistic physical and chemical descriptions and analyses of severe accident sequences and the related processes for the release and transport of radioactive substances. The studies included in the FILTRA project must to a great extent be based upon the more extensive studies underway in other countries, especially in the USA. The results of such studies are expected to be of great importance in the following respects:

- The design of a technically well balanced system for filtered venting.

- The assessment of alternative to a filtered venting system when considering further strengthening of the barriers against the release of radioactivity in the event of accident involving severe core damage.

- An overall assessment of the value with regard to risk reduction of adding filtered venting systems or other release-mitigating systems to existing reactor plants.

1. GENERAL BACKGROUND

Various risk assessment studies (1, 2, 3) have shown that if large amounts of the radioactive fission products in the core of a large power producing light water reactor should become dispersed to the surroundings in an accident they can, particularly under unfavorable weather conditions, have a very damaging effect. Partly there is the risk that a large number of people will be subjected to either acute radiation injury or delayed radiation effects (cancer) and partly there is the risk that large areas of ground will need to be evacuated for a long period because of contamination from long lived radioactive isotopes.

Against this background recent discussions concerning questions of reactor safety, in particular at the political level, have been concentrated on questions such as

- how large is the risk for an accident to a reactor which will result in appreciable releases of radioactivity, and how are these risks assessed?
- what possibilities are there to reduce further the risks for appreciable releases of radioactivity in the event of a reactor accident?
- what emergency plans are reasonable for evacuation, and suchlike, considering the risks for different reactor accidents of varying severity?

The first requirement for there to be an extensive release of radioactivity to the surroundings is that the reactor core is seriously damaged. A light water reactor intended for the commercial production of electricity is equipped with a series of safety systems to prevent the core from being damaged during various sorts of operational disturbances and incidents. If all of the preventive safety systems were to fail and the core were to be damaged the reactor is in addition equipped with a number of systems to control accidents and to limit the releases of radioactivity to the surroundings.

The strong reactor containment, made of iron and concrete, which encases the reactor is there primarily to limit the release of radioactive substances.

The total protection against damage to the surroundings is thus composed of the combined effectiveness of the preventive, controlling and release limiting measures.

The Swedish Government Commission on Reactor Safety considered in its report (5) the balance between safety measures for preventing and controlling accidents, limiting releases, and mitigating consequences of accidents the Commission concluded e.g. the following:

> "There are still good reasons for forceful efforts on preventive and controlling measures. Amongst other things there should be a systematic application of statistical fault analysis, and operational experience should be gradually followed up to provide a basis for further safety improvements of that sort. The committee considers however that the measures to mitigate the consequences have been paid much too little attention. Preliminary American analyses using a statistical method show that there are considerable reductions in the risks for serious consequences to the surroundings which could be achieved by such methods. In

> addition to this systems to limit consequences can give appreciable protection to the surroundings from the effects of accidents, which have not been predicted in their entirety in the statistical systematic fault analysis."

The Swedish Government Commission on Reactor Safety recommended forceful efforts on systems for mitigating the consequences aimed at improving such systems in Swedish nuclear power plants. Thus the Commission, against the background of the TMI accident, emphasized that it must be assumed in safety work that a serious accident to the core can occur, with the resultant release of considerable amounts of radioactive substances from the core and subsequent overloading of the release limiting systems. The Swedish government and parliament have in principle supported the conclusions of the Commission on Reactor Safety (Government bill 1979/80:170).

In order to study in more detail the possibility of, by modification or additions to the existing reactor containment, further reducing the risk for extensive releases of radioactive material in the event of a serious reactor accident, the Swedish Nuclear Power Inspectorate initiated the project known as FILTRA in February 1980. It is being carried out in co-operation with the Oskarshamnsverkets Kraftgrupp AB, the Sydkraft AB, the Swedish State Power Board: Asea-Atom and Studsvik Energiteknik are leading different parts of the project.

The project is divided into three phases. Phase 1 is the overall study of the problem. More detailed studies of important problems will be performed in phase 2. Finally the results of the entire project will be assessed and evaluated in phase 3.

In this first progress report within the FILTRA project, the results from the first phase are described in the form of an overall inventory of the problems and draft proposals for constructions to relieve overpressure in the reactor containment, and cleansing of the gases in a filter chamber.

The results presented in this report are, in accordance with the goals of the project, primarily applicable for the conditions in Swedish BWR nuclear power stations.

Whilst waiting, amongst other things, for the results from current American studies a more general description is given for Swedish PWR nuclear power plants.

A plan for the continuation of the work is given at the end of the report. Phases 2 and 3 are expected to take about one year each. The work in phase 3 can to some extent be carried out simultaneously with the completion of phase 2. The total amount of time which will be needed to complete this research project is judged therefore to be somewhat more than 1½ years.

2. THE NUCLEAR POWER PLANTS CONSIDERED

The discussions in this report, of the various accident sequences and limiting measures against releases, assume some knowledge of the design of the various nuclear power plants of interest. In Appendix 1 of the fuller Swedish report the most important technical details of the Swedish nuclear plants are therefore summarised.

There are two sorts of light water reactor power plants in Sweden, BWR, Vendor Asea-Atom; and PWR, Vendor Westinghouse.

Some of the most important data for the Swedish plants are given in Table 2.1.

Plant	Taken into operation	Type	Thermal power, gross MW	Electrical power, net MW	No of fuel assemblies in the core	Weight of UO_2 fuel (ton UO_2)	Weignt of core (ton)
Oskarshamn 1	1972	BWR	1365	460	448	90	133
Oskarshamn 2	1974	BWR	1700	580	444	89	132
Ringhals 1	1976	BWR	2270	750	648	130	192
Ringhals 2	1975	PWR	2440	800	157	81	104
Barsebäck 1	1975	BWR	1700	580	444	89	132
Ringhals 3	*	PWR	2783	915	157	81	104
Barsebäck 2	1977	BWR	1700	580	444	89	132
Forsmark 1	1980	BWR	2700	900	676	135	200
Ringhals 4		PWR	2783	915	157	81	104
Forsmark 2	*	BWR	2700	900	676	136	201
Oskarshamn 3		BWR	3000	1050	700	141	208
Forsmark 3		BWR	3000	1050	700	141	208

* Being commissioned.

Table 2.1

Swedish nuclear power plants.

3. THE BACKGROUND FOR FILTRA IN ALREADY PUBLISHED RISK ANALYSES

3.1 A survey of the accident sequences and safety systems

3.1.1 Barriers against the release of radioactive material

In order for large amounts of radioactive material to be released from the reactor core and dispersed into the surroundings a number of different barriers must be breached.

The first barrier is the uranium dioxide fuel pellet itself. Uranium dioxide is a ceramic material which has the property of being able to bind a large amount of the radioactive fission products. It also has a very high melting point, approximately 2 850°C.

The second barrier consists of the completely closed zirconium alloy tubes in which the uranium dioxide pellets are enclosed. The fission products which escape from the fuel pellets during normal conditions are mostly gaseous and remain in the space between the pellets and the cladding.

The third barrier is the reactor pressure vessel and its piping system. These systems have no contact with the outside under normal conditions other than via carefully controlled auxilliary systems and filters. In the event of an accident automatic systems should ensure that the radioactive substances are completely isolated from the surroundings.

The fourth barrier is the reactor containment which encases the reactor pressure vessel and some of its systems for the circulation of cooling water and steam. In the event of an accident it is assumed that some radioactive substances will be released from the pressure vessel and the associated pipe work which is situated whithin the containment, for example because safety valves are opened or because the accident is caused by a pipe break. The reactor containment is built to withstand large stresses in the event of an accident, without radioactive substances leaking out to the surroundings.

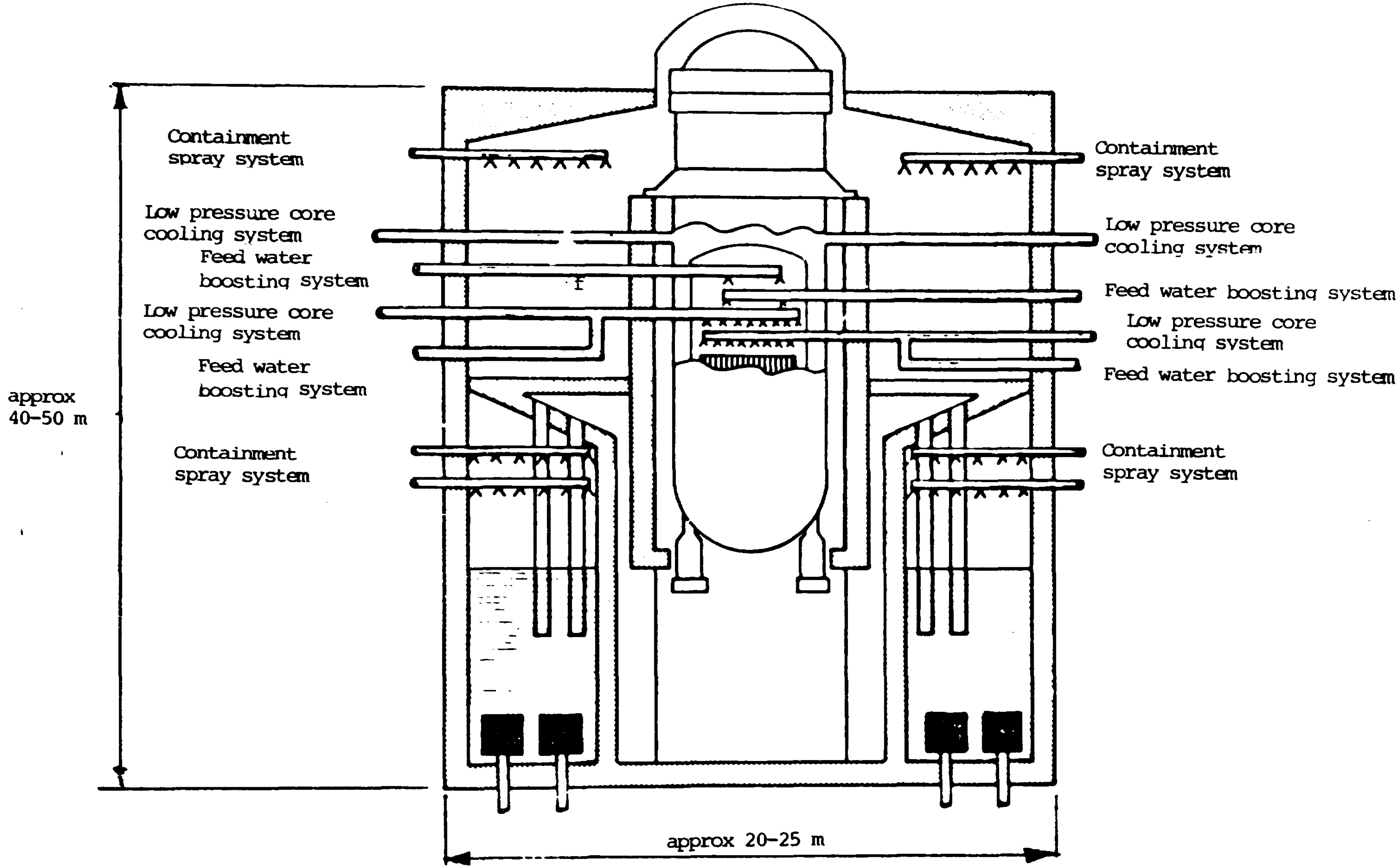
Accidents in nuclear power plants should be prevented by extensive safety systems which are ultimatly intended to prevent all the barriers mentioned above from being breached. The safety systems are usually divided into measures for accident prevention, accident controlling and mitigating the consequences of an accident.

An example in the first category is the reactor tripping system which automatically shuts down the reactor if the pressure, water level or the power diverge from their normal values. The emergency cooling system delivers water to the reactor to prevent the core from becoming too hot if the ordinary cooling water supply fails or if it is lost due to a break in one of pipes connected to the pressure vessel, for example.

The reactor containment and its systems are part of those measures which are aimed at mitigating the consequences of an accident, such as preventing large amounts of radioactive material from being released to the surroundings in the event of a serious accident to the core. The cooling system in the containment is designed to remove the decay power from the core (see below) and to prevent the pressure within the containment from becoming too high. The spray system in the containment helps to wash air borne radioactivity in the atmosphere of the containment down into the water in its bottom. The most important safety systems for cooling the core and containment of a BWR and PWR are shown in Figures 3.1 and 3.2 respecitively.

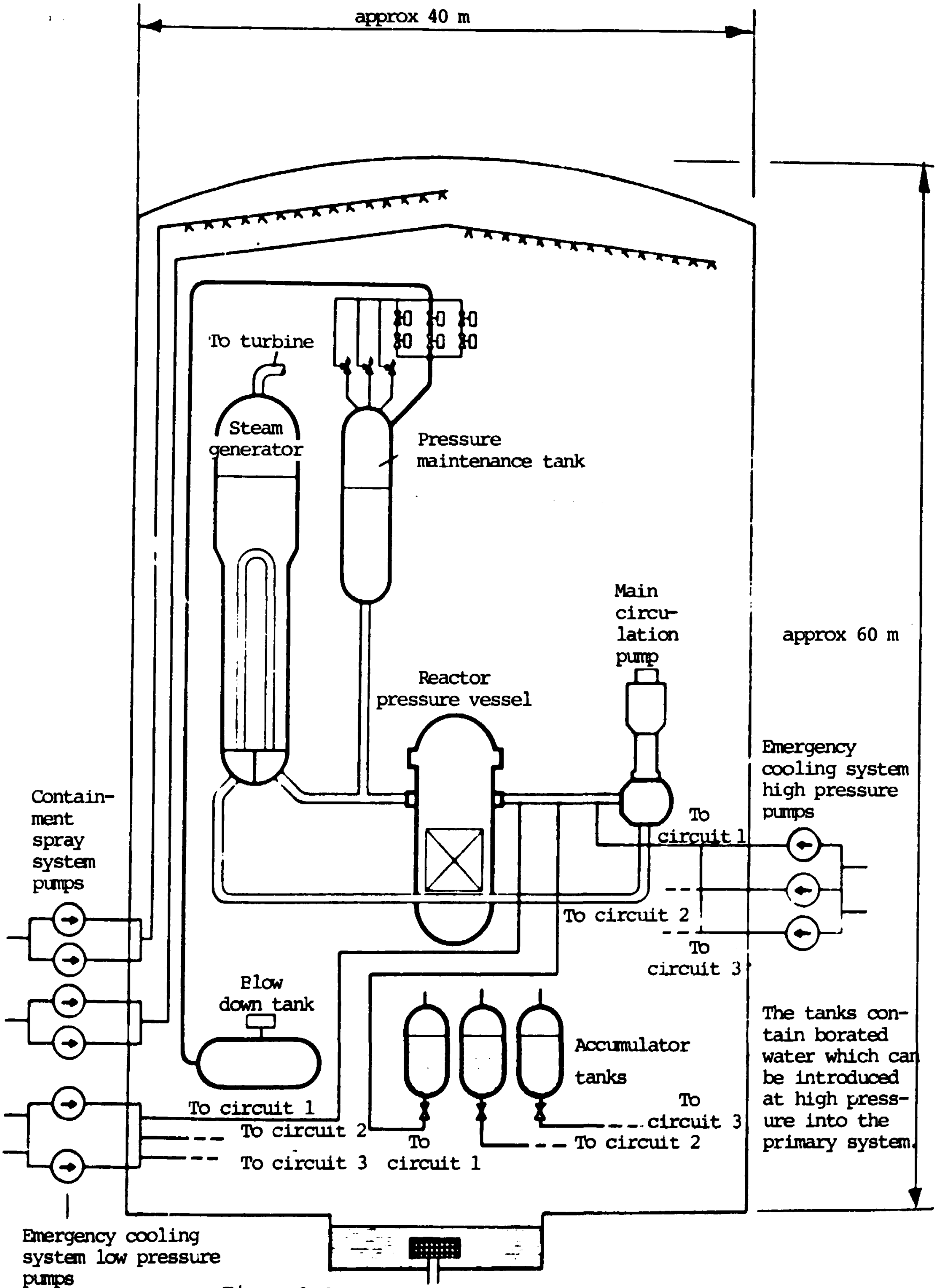
The different safety systems consist of separate independent subsystems which are designed such that the safety function can be fulfilled without all the various subsystems working. Wherever possible the subsystems are placed in separate rooms so that if one subsystem fails, because of for example a fire or flooding in one room, the failure cannot spread to the other subsystems. The aim is also that the same function can be fulfilled by different safety systems based on different technical principles. For axeample a reactor can be shut down either by introducing control rods into the core, with both an electrically and a pneumatically operated actuator, or by a system which sprays a boron solution into the reactor water.

In order for an accident with serious damage to the core to occur it is thus necessary for several preventive safety systems to fail simultaneously, and in some cases they must be out of action for a considerable length of time. In order for a serious release of radioactive material to the surrounding to occur, given core damage, the release limiting safety systems must also be out of action. The probability of such a combination of events to occur is studied in different types of risk analyses.



<u>Figur 3.1</u>

More important safety systems of a modern BWR.



Figur 3.2

More important safety systems of a PWR.

3.1.2 Accidents which are physically possible and which can lead to large releases of radioactivity

The most serious accident sequence which can occur in a light water reactor implies the loss of the reactor core cooling water. If the reactor coolant is lost the energy production, from the chain reaction in the core, dies out even if it has not previously been terminated by the control rods or boron solution. The decay of the fission products continues to produce so much heat, decay power, see Table 3.1, that on the loss of coolant the fuel becomes overheated, and eventually melts.

If the fuel cladding reaches a temperature of 800 - 900°, as opposed to the normal temperature of about 350°C, the cans can begin to crack. In the first instance gaseous fission products, such as the noble gases and iodine, leak out through the cracks into the reactor pressure vessel and the primary circuit.

At even higher temperatures, above 1 200 - 1 400°C, the zirconium cladding is completely destroyed, and some portion of the core can collapse and eventuelly melt. If steam flows through the core, which is probable, the process is speeded up at these high temperatures by the zirconium combining with the oxygen in the steam in an exothermic reaction which produces large quantities of hydrogen. If the core melts, all the gaseous fission products in the fuel pellets are released. It is also expected that other fission products, such as cesium, will be released from the melt which will attain a temperature of 2 200°C or more.

Table 3.1

The decay power after full burn up of a 2 700 MW_{th} PWR core, i.e. comparable in size to the cores in Ringhals 2 - 4*.

Time after reactor trip	Decay power MW
1 sec	163
4 sec	145
10 sec	128
40 sec	103
100 sec	87
400 sec	67
1000 sec	54.6
1 hr	37.3
2 hr	30.3
5 hr	23.8
10 hr	19.9
20 hr	16.5
50 hr	11.5
100 hr = 4.17 days	8.90
200 hr = 8.3 days	6.57
500 hr = 20.8 days	4.31
1000 hr = 1.39 month	3.02
2000 hr = 2.78 months	2.03
5000 hr = 6.9 months	1.02
8760 hr = 1 year	0.609

Source: Kemeny Commission: Technical Staff Analysis Report on Alternative Event sequences.

* Ringhals 2 (~2500 MW)
Ringhals 3 and 4 (2783 MW)

A core meltdown can under certain conditions result in such large increases in pressure t rough the formation of steam or gases within the containment, that the latter can be damaged and the radioactive contents of the core can, to a greater or lesser extent, be released to the surroundings.

3.1.3 General aspects of safety analyses

The probability of accidents of varying severity occurring is being studied in different types of safety analyses. The more comprehensive safety analyses in general consists of studies of the following parts:

1. Analysis of different sequences of events (and their probabilities) which lead to serious core damage, usually due to loss of coolant to the core.
2. Analysis of the events which lead to containment failure, or to other leakage from the containment, and estimates of how much of the radioactivity released during a serious core accident would reach the surroundings through such a leakage.
3. Analysis of the detrimental effects to the population and environment as the result of the release of different amounts of radioactive material.

<u>The first part</u> can be tackled using mostly established methods of reliability theory (event and fault tree analysis). The probabilities estimated for different sequences of events are associated with certain inherent uncertainties. These can however be expected to decrease as, on the basis of actual operational experience, the statistical material (for the critical fault trees) is improved as to the frequency of the malfunctioning of the different components and systems in the plants.

The second part is mostly being studied theoretically by analysing the various accident sequences. These studies have been complemented by laboratory and model experiments on damage from various critical events. Experience from full-scale events including serious core damage exists essentially only from the TMI-2 accident.

Against this background realistic estimations of the probability for the occurrance of specific accidents of varying severity, for example with respect to the magnitude and extent of the releases, will be very uncertain, mainly because it is generally very difficult to express the probability that different systems in a reactor plant (for example for a limited period of time) are out of action, completely or partially, under the severe conditions of a reactor accident. This difficulty stems from the fact that the systems are assumed to experience conditions in excess of the conditions and limits for which they have been designed. Further the physical and chemical processes which determine the release of radioactive material during accident conditions are still not fully understood. The studies of accidents have therefore up to the present time often been directed towards trying to determine the upper limits for the radioactive releases from different types of accidents based upon certain limiting physical assumptions.

The third part, the calculation of the consequences, can, if it is assumed that the amounts and the details concerning the release are well known, be studied using relatively well established methods and models for the dispersion of radioactive material in the atmosphere, and for the relationship between the radiation doses and detrimental effects on the health of the population. Such methods and models are based on both theoretical and empirical studies, for example how the probability of different weather conditions can affect the extent of damage. An estimate of the risk for different degrees of damage can thus be carried out reasonably reliably provided the release conditions are well defined - which because of the uncertainties in the analysis of part 2 - is seldom the case, other than for estimation of the upper limits.

The FILTRA project is concerned primarily with part 2, that is to say the study of accident sequences in those admittedly rare cases in which the preventive systems have failed to such an extent that a serious core accident has occurred. Such failure can be said to be the starting point for the FILTRA project. The probability of such a failure occurring (part 1) is therefore not discussed in the FILTRA project. Nor is a detailed study of the consequences included (part 3). Results from these parts have been used within the FILTRA project when assessing different technical solutions for their ability to reduce the release of the radioactive materials which are expected to result in the most damaging effects.

3.2 Risk analyses carried out to date

In this section some of the main points are summarized in the risk analyses and studies of accident sequences which have been carried out to date. A somewhat more complete resumé is to be found in Appendix 2 of the Swedish report.

3.2.1 WASH-1400 (1)

The first comprehensive and systematic study of possible accident sequences was the American Reactor Safety Study, WASH-1400, which was published in 1975. The study considers a Westinghouse PWR and a General Electric BWR. The PWR plant studied is similar to the corresponding Swedish plants. The BWR plant exhibits relatively large differences in comparison on with the corresponding Swedish plants, amongst other things in respect to the main circulation circuits and the containment design.

A large number of sequences of events which can lead to core accidents and the subsequent radioacitve release are studied in WASH-1400. From the point of view of risk assessment, it was found that the resultant release could be described by a number of categories of varying severity.

3.2.2 The German reactor safety study (2)

In the Autumn of 1979 the first stage of the comprehensive German reactor safety study concerning the KWU PWR was reported. The same methods were used as were used in WASH-1400, but the German reactor design differs from the Westinghouse design in a number of ways. The differences in the containment design are of particular interest to the FILTRA study (Appendix 2 of the Swedish report).

Basically the German study came to the same conclusion about the types of release categories as did WASH-1400, even if the accident sequences differed in a number of ways because of the design differences between the reactors studied.

3.2.3 The study of accident consequences of the Swedish Institute of Radiation Protection

In its report, of December 1979, "More effective emergency preparedness" (3) the Swedish Institute of Radiation Protection (SSI) quoted calculation results of the consequences of reactor accidents with releases of varying severity for Swedish reactor sites.

The SSI report is based to a large extent upon the accident sequences and release categories described in WASH-1400.

When considering the extent of damage the following three types of relases can be distinguished, according to the SSI report:

A. The release of most of the contents of the core both with respect to the inert gases (100 %) and iodine (50 - 100 %) as well as cesium and other metals (approx 50 %). The accident sequences, which according to WASH-1400 were e.g. assumed to result in such releases were caused by very severe steam explosions, or were to be associated with failure of the containment because of excess pressure, or other damage which gives rise to direct and large paths of leakage from the containment to the surroundings. The most serious release situations are assumed in WASH-1400 to arise from a core meltdown associated with steam explosions in the reactor pressure vessel of such force that parts of the pressure vessel, followed by large parts of the reactor core, are expelled through the roof of the containment. The theoretical models for such severe steam explosions, on which the calculations in WASH-1400 were based, are no longer considered to have a physical basis, and therefore such explosive sequences should no longer be taken into account when designing safety systems and planning emergency measures (4).

B. Release of most of the core contents of inert gases (50 - 100 %) and small amounts of iodine and cesium (0.1 - 0.5 %). The accident sequences, which would result in this type of release were, according to WASH-1400, a core meltdown or serious core damage (50 - 100 % of the cladding damaged) combined with damage to the containment, resulting from for example excess pressure, which causes small or indirect paths of leakage between the containment atmosphere and the surroundings.

C. Release of about 10 % of the core contents of inert gases. The accident sequences which result in such a release were in WASH-1400 assumed to consist of core damage of varying severity (both moderate and more serious damage to the cladding) in combination with a small leak through faulty valves, gaskets, or similar, either from the containment, or from systems outside containment which must be kept functioning during and after an accident. The accident in TMI-2 (see below) is an example.

In Table 3.2 typical consequences are summarized (but not for extremely unfavourable weather conditions) according to SSI for the three release types and the reactor site at Barsebäck. The consequences do not vary much between the different Swedish sites; this is particularly so if the case involving a steam explosion is excluded.

In the table it can be seen that there is little damage outside the plant as long as the release of considerable amounts of other radioactive substances than the noble gases can be avoided.

SSIs calculations indicate that for release category C individual doses of a few tens of rad can at the most be incurred by people in the immediate vicinity, a few kilometers from the plant. (In TMI the maximum individual dose outside the plant was estimated to have been 0.1 rad.) A dose of a few tens of rad means that the probability of an exposed individual dying of cancer is increased by a few tenths of a procent above the average, which in Sweden is about 20 %. Spontaneous abortions can also occur in some cases. It should be noted that relatively few people live within five kilometers of a Swedish reactor site.

3.3 Studies of release processes

Comprehensive theoretical and experimental studies are being undertaken in different countries aimed at obtaining better knowledge about which releases can be expected in the event of different accident sequences.

Table 3.2

Types of release and the corresponding damage for the Barsebäck site.

Type	Release core inventory, %	Damages in typical cases (not extreme conditions)
A	Inert gases 100 %	• A few hundred km^2 must be evacuated immediatedly and for several years.
	Iodine 50 - 100 %	• Tens of acute deaths.
	Cesium 50 %	• A few hundred or thousand cases of acute sickness.
		• A few hundred abortions.
		• A few tens of thousands of extra cancers.
B	Inert gases 50 %	• A few km^2 in the vicinity need to be evacuated within a few days and for a few years. Milk control.
	Iodine 0.1 %	• No acute sickness - a few abortions in the vicinity.
	Cesium 0.5 %	• A few extra cancers.
C	Inert gases 10 %	• No acute sickness or abortions.
		• A few extra cancers.

As previously mentioned, such studies have already led to the result that the most severe release events in WASH-1400, and in the German reactor safety study (2), resulting from a steam explosion are now considered to have no realistic physical foundation.

For other accident sequences only relatively simple separation processes for the removal of iodine, cesium an other metallic aerosols from the containment atmosphere were taken into account in WASH-1400 and the German study.

These separation processes were considered to be independent of the physical and chemical conditions in the reactor pressure vessel and containment. The following separation process were considered:

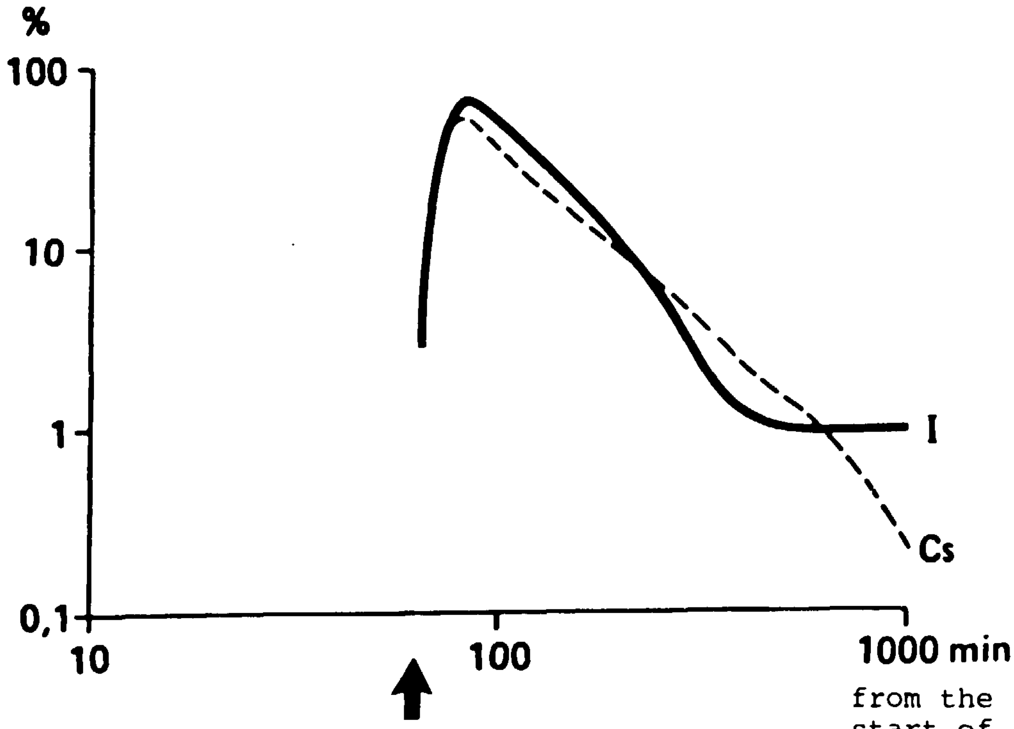
- An equilibrium of molecular iodine in steam and in solution in water (without additions): a separation of 1:100 can be achieved through diffusion and condensation on damp surfaces in the containment.
- Aerosols of cesium and other metals are deposited by gravitation, with a certain amount of agglomeration into larger particles as a contributory factor; the particles can also act as nuclei for the condensation of water droplets.

Figure 3.3 shows typical separation curves for iodine and cesium in a large PWR containment with the above assumptions.

In the Autumn of 1980 several critical surveys of the source-terms for the accidental radioactive releases were published in the USA. The reviewers were critical of the source-terms in WASH-1400 and also of those used as a basis for the NRC guide lines on emergency planning against accidents at nuclear power plants. The scrutinies, which were performed partly by the research institute of the American Electrical Power Industry, EPRI, and partly by research workers at several American National Laboratories, are based upon a perusal and summary of the theoretical and experimental information available to date. The experimental results include measurements of the releases from the TMI-2 accident, and from a number of other intentional and unintentional core accidents in smaller research reactors. A common feature of these core accidents is that they occurred under wet conditions, that is to say that water or steam was present, which is to be expected for accidents to light water reactors. This experimental experience indicates for accidents under wet conditions, that the amount of radioactive substances, iodine primarily, which will be released to the containment atmosphere, and possibly also to the surroundings, is considerably less than was assumed in WASH-1400, or shown in Figure 3.3. The following separation processes are indicated:

- The iodine forms iodides which are bound very effectively in solution in water (compare the TMI-2 accident: 50 % of the iodine in the core was released, two tenths of a million escaped from the containment.)
- Cesium and other metals can form water soluble salts, such as cesium iodide, which are bound very effectively in water.
- The condensation of steam on the aerosol particles and other processes can under certain conditions result in a more effective separation than previously assumed.

Proportion of core content
in the containment atmosphere



from the start of the incident

Source: Deutsche Risikostudie Kernkraftwerke

Core melts

Figure 3.3

Concentration of molecular iodine and cesium aerosol in a large PWR containment assuming simple separation processes.

The discussion of these questions indicates that there are considerable gaps in the knowledge of the physical and chemical conditions prevalent in the reactor system and its containment during different accident sequences. There are also gaps in the knowledge of under which conditions and in which cases one can benefit from the various separation processes for iodine and metal-aerosols, which can be considerably more effective (by a factor of 10 or more) than has been assumed hitherto.

3.4 Previous studies of modified containments

In Sweden, as in several other countries, the discussion of a further strengthening of the containment's function to limit the release of radioactivity was first raised in con·ection with studies of the underground siting of nuclear power plants (7, 8). In this connection Swedish research workers pointed out that it would be appropriate to ventilate the underground reactor cavity of the nuclear power plant through a long tunnel filled with rocks and gravel (39). A sufficiently long tunnel could function as a passive safety system, which could more or less eliminate the risk for an air borne release, even in the event of very serious accidents. The basic principle is that the amounts of steam and gas which can be produced during a serious accident are limited, thus if the volume of the tunnel is sufficiently large, the gases released during an accident will displace the air in the tunnel, and the steam will condense. Subsequently there will only be slow diffusion to the surroundings of such substances as the inert gases, that cannot be filtered by a crushed rock or gravel bed. If the tunnel is large the diffusion will be so slow that a large part of the radioactivity of the noble gases will have decayed before reaching the atmosphere.

Since the analyses in WASH-1400 indicated that a failure of the containment, as a result of excess pressure, made a relatively large contribution to the risk of large releases of radioactivity, the NRC commissioned SANDIA Corporation, in 1977, (50) to carry out a preliminary analysis of the possibility of reducing the risk for a large release, in the event of a reactor accident, by modifying the reactor containment. The designs considered ranged in principle from simple modifications of existing containments to underground siting. The study was concentrated to PWR plants, and BWR plants were considered more summarily. The results of the study indicated that systems for Filtered Atmospheric Venting or Compartment Venting were the most cost effective solutions: they would give the greatest reduction in risk in relation to their cost. The cost effectiveness analysis was based on the risks quoted in WASH-1400, and as a measure of the risk the expectation values for early and late radiation injuries were used, integrated over different release patterns with their associated probabilities.

In Sweden the Commission on Reactor Safety sponsored further studies of modifications to the reactor containments aimed at investigating the possibility of reducing the risk for large releases of radioactivity in connection with accidents. These studies, which were the starting point for the FILTRA project, had obtained considerable information from the previously mentioned SANDIA study and are reported fully in the expert appendices to the main report of the Commission on Reactor Safety (5).

None of the containments of existing large commercial light water reactors have as yet been equipped with the means for filtered pressure relief to the atmosphere or to separate pressure relief chambers. Some Canadian heavy water reactors, CANDU, are however equipped with the possibility of blowing off the gases and steam, which can be produced during an accident, to an extra containment which is common for several reactors on a site. In Germany discussions are under way concerning the reduction of the risk of failure due to excess pressure by introducing an additional spray in the space between the steel dome and the concrete shell in the German reactor containments. The spray system would be used to cool the outside of the steel dome and thus reduce the pressure in the containment in those accident situations in which all the heat sinks are lost.

Based mainly on experience from the accident in TMI, the NRC has declared that it intends to reconsider which requirements should be imposed upon the various safety systems regarding their capability to limit the damage from a very serious accident: accidents which are more severe than those which have up to now been defined as design basis accidents (6).

This review process, including public hearings with various experts and interested parties, is expected to take a couple of years. Comprehensive preparations, and theoretical and experimental investigations, have already been started by the NRC as well as by the utilities and vendors.

3.5 Summary: Background for the FILTRA project in already published risk analyses

Based upon accident studies and experience obtained to date the following conclusions are of interest for studies of filtered pressure relief:

- If the reactor containment remains intact it can prevent large releases of radioactive material to the surroundings even in the event of an accident, which is more serious than the so called design basis accidents. Because of the abnormal stresses to various systems during a serious accident (e.g. the auxiliary cooling systems outside the containment) it may be difficult to avoid completely some leakage of the radioactive noble gases to the surroundings. The maximum damage outside the plant from a release of noble gases can be expected to remain small both to the individual and the community for Swedish reactor sites.

- Failure of the containment creating large uncontrolled paths between the containment and the surroundings greatly incrase the risk for serious releases of radioactivity. Steam explosions are, according to more recent studies, unlikely to be so violent that they will cause major damage to the containment. In the case of certain severe accidents, which are beyond the bounds of the current design basis accidents, the containment can however be damaged as a result of excessive pressure. The location of such damage to the containment and the resulting leak paths can be difficult to predict.

- It is particularly desirable to prevent failures of the containment, from occurring before or early in a core meltdown, since the various separation processes which act within the containment will then have too short a time for acting effectively. Failure of the containment also increases the risk for damage to the systems designed to control and mitigate the consequences of an accident.

- There are still considerable gaps in the knowledge of those processes which during an accident sequence can affect the release and transport of a number of radioactive substances from the core to the surroundings.

Experience gained from accident studies and from actual accidents such as TMI-2 also enable various starting points to be defined for a more detailed study of some of the systems for filtered pressure relief:

- A primary aim should be to achieve considerable reduction in the risk for the release of radioactive substances which can cause long term ground contamination as a result of a severe reactor accident.

- The system for filtered pressure relief should have a minimal influence on existing safety systems and their functions during current design basis accidents. The pressure relief system should not be triggered by other events than those in which the design pressure for the present containments is exceeded.

- The system should be passive during the first day/days of an accident sequence - pre-arranged active measures can then be taken into account successively. The possibility of establishing stable conditions in the long term with minimal risk for leakage of radioacitve substances outside the plant should be discussed.

- The requirements on filtering capabilities should not be set higher than that which corresponds to the risk for the release in the event of less severe accidents which do not (necessarily) trigger the pressure relieving of the containment.

- Diversified solutions to the condensation and filtration functions should be studied, e.g. in order to be able to take advantage of different types of separation processes.

Furthermore the work within the FILTRA project has in the first instance been directed towards the study of accident sequences and measures for pressure relief in Swedish BWRs. One reason, amongst others, is that the Swedish Government Committee on Reactor Safety stated that studies should initially be concentrated to plants sited near large centres of population, for Sweden this is first and foremost Barsebäck. For similar reasons corresponding studies in the USA have been concentrated to PWR plants similar to those in Ringhals. The American efforts in this field are being followed with great interest, partly through the American consultants engaged by the FILTRA project, and partly through a special co-operative agreement for the exchange of information between this project and the NRC, and partly through the Swedish State Power Board membership in the Westinghouse Owners Group.

Thus the section in the progress report concerning PWRs is based on investigations within the Swedish State Power Board section KSP, which are based upon studies by American and German consultants.

The work within the FILTRA project has been divided into two main parts with Studsvik Energiteknik AB and Asea-Atom AB as leaders for the two parts:

Studsvik Energiteknik:

- Studies of temperature and pressure transients in the containment for sequences of events not currently encompassed by the design basis accidents. In the first instance those sequences have been studied which are limiting from the point of view of exceeding the ultimate strength of the materials.
- Studies of the transportation and separation of radioactive substance through various designs of filters.

Asea-Atom:

- Studies of various design solutions for filtered pressure relief, with respect to the functions of a condenser and a filter.
- Analysis of how such a filtered pressure relief system would affect other safety functions.

4. CALCULATED RISE IN PRESSURE IN THE REACTOR CONTAINMENT DURING TWO CORE MELT ACCIDENT SEQUENCES IN A BWR

4.1 Introduction

One of the starting points for work in the project FILTRA is to study accident sequences which are rare but in which the safety functions for accident prevention fail to such an extent that the reactor core and containment are damaged.

The containment would be damaged in some of these cases as a result of excessively high pressure.

According to previous studies, major damage could also be considered possible as a result of steam explosions and complete penetration of the concrete bottom slab of the containment (1). Steam explosions are thought, according to more recently published assessments (4), not to be able to cause major damage to the containment. The penetration process is unknown and will be studied further in the next phase of the FILTRA project.

Two basic types of accident sequences have been studied, which under certain conditions lead to excessively high pressure in the containment, according to the calculations reported below. The pressure will be, according to the calculations, so high that the sequence of events could lead, without pressure relief, to failure of the reactor containment. The calculations also show that the blow-off of gas and steam through an opening with an effective area of 1 m^2 will limit the rise in pressure to within permissable values. Failure of the containment as a result of an overpressure can thus be avoided.

The calculations are valid for conditions in Barsebäck, but can be considered representative for other BWRs as well.

The two basic accident sequences are:

(i) core meltdown as a result of an inadequate supply of water to the reactor vessel;

(ii) inadequate cooling of the reactor containment.

It has also been noted that the pressure relief system studied can increase the safety margin during other incidents involving pressure increases which do not necessarily lead to a core melt accident. These events - which include the incomplete steam condensation during a LOCA (faulure of the pressure suppresion function) - are currently being studied in another context.

4.2 Inadequate water supply to the reactor pressure vessel

The sequence of events for category (i) occurs if the supply of water to the reactor pressure vessel becomes inadequate as a result of several simultaneous malfunctions during operation, for example through the loss of electric supplies to the pumps, which supply water to the reactor. The water inside the pressure vessel boils off, the core is left dry and is eventually melted by the decay heat. Similar core meltdown sequences can result from a pipe break and the loss of cooling to the core.

During such a heating and melting sequence the zirconium in the core, e.g. the fuel cladding, is oxidized exothermically and hydrogen is formed. Oxidation of all the zirconium in the core, under an extreme accident sequence, would result in a volume of hydrogen, which under normal conditions of pressure and temperature, corresponds to approximately twice the volume of the containment.

In the pressure vessel the core continues to melt, and the melt falls towards the bottom of the vessel whilst the remaining water boils away.

During the melting sequence radioactive substances are released from the core, for example inert gases and iodine, and metals, such as cesium, with high vapour pressures.

In addition a cloud of gas borne particles are dissipated from the melt. It consists mainly of non-radioactive construction materials, and to a lesser extent it contains some radioactive material.

The melt heats up and melts through the bottom of the reactor pressure vessel and falls onto the concrete floor beneath. Once on the floor the melt heats up and melts the concrete, which then releases its water of crystallization. The steam bubbles through the melt and oxidizes the remaining zirconium, and possibly other metals such as chromium and iron as well, whilst heat and more hydrogen are being produced.

The steam and hydrogen transport a large part of the remaining radioactive substances of the melt into the atmosphere of the containment. A cloud of non-radiactive particles is released from the melt. The particles consist of the oxides of concrete and construction materials such as iron oxide, silica, and other materials. Most of the radioactive substances condense on the non-radioactive particles in the cooler atmosphere.

The total quantitity of gas borne particles formed during the meltdown and the attack on the concrete is estimated to be nearly two tons. Of this the radioactive material is estimated to be a few tens of kg (see Section 6).

The reactor containment is filled with nitrogen. The hydrogen therefore cannot ignite but it is very hot when it is formed. The hydrogen and steam from the melt could in some cases (when the containment spray is assumed to be inoperative) heat the atmosphere in the containment to a temperature of nearly 300°C. The partial pressure of the non-condensable gases in the containment, mostly nitrogen and hydrogen, can at this stage exceed the design pressure of 0.5 MPa for the containment.

The extent of the formation of hydrogen depends upon the availablility of steam during the core meltdown in the reactor pressure vessel, and upon the later attack by the melt on the concrete. This will be studied in more detail in the next stage of the project.

From the concrete floor underneath the reactor pressure vessel the melt runs through various openings, amongst others the drainage pipes down into the condensation pool.

When the melt runs down into the condensation pool it is split into larger or smaller fragments, which are very hot and encased in a thin steam layer. The transfer of heat to the water is determined by heat radiation, convection and conduction through the steam film, and its total heat transfer area. In the calculations reported below the fragments are considered to be spheres with a diameter of 50 mm.

The steam formed on the surface of the fragments condenses in the surrounding cooler water in the pool. A net production of steam above the surface of the pool will occur only when the water temperature approaches its boiling point. The amount of steam which is formed depends upon the amount of heat in the melt, how much water this heat is dissipated through, and how much the water is heated up.

The condensation pool contains 2 000 tons of water. The heat in the melt is sufficient to raise its temperature by about 25°C, for example from 50°C to 75°C. Thus there will not be a net production of steam if the heat content of the melt is distributed throughout the condensation pool.

Calculations for events during pressure relief have been performed assuming that a rupture disc at 0.65 MPa opens to a pressure relief channel with an effective cross-sectional area of 1 m^2.

The results show that the pressure within the reactor containment, for those cases of the accident type (i) considered to date, can be limited by pressure relief to permissable values.

The calculations have been carried out using the computer program MARCH, which is described in more detail together with the results in (17). The models used in the computer program should be reviewed critically in order that the results can be interpreted correctly, and their physical reality assessed. Sensitivity studies should be carried out to estimate the effect on the results of essential assumptions and uncertainties in the models. These analyses will be performed in the next phase of the work.

The final situation in accidents of type (i) will be that most of the molten core runs down to the bottom of the condensation pool. Small portions of the molten core may remain in the pressure vessel or on the concrete floor below.

The questions as to whether the pile of fragments of various sizes of the molten core on the bottom of the condensation pool can be cooled or not has not yet been studied. It is possible that sufficient water and steam will flow through the pile to keep the fragments cooled. It is possible that the fragments will not be kept cool enough and will be sintered and melted. This melt, which is under the water in the condensation pool, would again attack the bottom slab and possibly melt through it. With the present state of knowledge we cannot state that the bottom of the condensation pool would be penetrated, nor can we state that it would not. These questions will be studied in more detail during the next phase of the work.

4.3 Inadequate cooling of the reactor containment

The sequence of events in category (ii) leads to situations where heat produced in the core is transferred to the condensation pool, and that the cooling of the latter is assumed to be inadequate in comparison to the heat produced. This results in the heat being accumulated in the pool water, its temperature rises and consequently the partial pressure of steam in the reactor containment rises. For a water temperature in the pool of 150°C the steam pressure is 0.5 MPa, and the total pressure within the containment is 0.65 MPa. At that pressure it is assumed that a rupture disc with an effective area of 1 m^2 opens, and the steam and gases are released so that atmospheric pressure (0.1 MPa) is regained within a few minutes. By that time about 8.8 tons of nitrogen and about 220 tons of steam have in some cases been released from the containment.

The rupture disc opens about 24 h after the beginning of operational disurbances which result in the loss of cooling for the pool, in a situation where the decay heat from the shut-down reactor is transferred to the condensation pool.

The pool is heated more quickly in a situation of operational disturbance in which it is assumed that complete shut-down of the reactor has not been successful. The heat production in the core cannot then be reduced to the level of the decay power but continues at a higher level. The rupture disc would then open about 1 h after the beginning of the operational disturbance.

On the sudden opening of the rupture disc the pressure drops rapidly and some of the water evaporates from the pool. As a result cavitation occurs in the emergency cooling system pumps, which feed water from the pool to the pressure vessel. The water supply to the pressure vessel may then be inadequate, and may result in core meltdown, see the events of (i). The same phenomenon can of course occur in the event of an uncontrolled failure of the containment resulting from excess pressure.

Such a development can be prevented by measures aimed at ensuring that water is supplied to the reactor pressure vessel in other ways, for example by the auxiliary feed water systems, which supply water from a store outside the reactor containment.

4.4 Other sequences of pressure increase

At the current stage of the work it is an open question whether certain types of sequence involving potential pressure increase, than those named above and should be considered when selecting the relief area and other items of the design of a filtered pressure relief system.

Such sequences include events in which the pressure in the reactor containment rises to an excessive level, because all the steam which escapes through a postulated pipe break is not condensed in the condensation pool. This situation could arise if, in the event of a pipe break, a leak occurred through the dividing wall between drywell and wetwell. Some of the steam could then leak through the dividing wall, without passing through the blowdown pipes into the condensation pool. It would then not be condensed but would contribute to the rising pressure in the containment, which could exceed the design pressure.

If the leak in the dividing wall is small, most of the steam escaping from a large pipe break, will condense in the pool, and the increase in pressure in the containment will be correspondingly moderate. If the leak is larger the pressure rise will be larger.

The pressure in the containment can be limited by blowing off the excess gases and uncondensed steam through a pressure relief opening, which is normally closed by a rupture disc. Calculations show that pressure relief through a rupture disc, which opens at 0.65 MPa and which has an effective area of 1 m^2, can limit the rise in pressure to within permissible levels, for a large pipe break in combination with a leakage area of about 1 m^2 through the dividing wall.

A large pipe break in this context is a guillotine break in one of the primary recirculation pipes. Such a break on one of the largest pipes connected to the reactor pressure vessel would result in the escape of 16 000 kg/sec of a water/steam mixture through the two ends together.

5. DESIGN OF A PRESSURE RELIEF CHANNEL AND FILTRATION PLANT

5.1 The design principles of the FILTRA plant

The main components of the FILTRA system are shown in Figure 5.1.

The FILTRA plant consists of a pressure relief channel which connects the reactor containment vessel to a filtration plant. The latter comprises a condenser for condensing the steam and a filter for the filtration and removal of aerosols and iodine from the gases relieved from the containment. The condenser and filter can be designed as a single unit or separately. The gases are lead to a chimney after filtration.

Each of the reactor units on a site are connected to the common filtration plant via a rupture disc and an isolation valve. During normal operations the reactor containment is closed and isolated from the filtration plant by the rupture disc. The isolation valve is kept in the open position.

In the event of an accident occurring such that the pressure in the reactor containment becomes too high, the rupture disc will open and the pressure will be relieved via the pressure relief channel to the condenser in which the steam is condensed. The condensate collects at the bottom of the condenser.

The remaining uncondensable gases then pass to the filter where any remaining aerosols and iodine are filtered and removed.

Between the condenser and the filter a reducing valve is located in order to even out the flow to the filter, and prevent it from being overloaded by the initial large flow of gas from the containment.

Contamination, by the steam and gas, of the other reactor units connected to the filtration plant is prevented by the rupture disc in those units. It is also possible to shut the isolation valves which are connected in series to the rupture discs.

The isolation valve is also used some time after the accident, when the rupture disc has opened to close the reactor containment again.

There is also a valve on the outlet side of the filter to completely isolate the plant from the surroundings for a period following an accident. The valves can be manoeuvred by remote control.

5.2 Design of the pressure relief channel

The necessary capacity of the pressure reducing pipe work is determined by the need for limiting the pressure in the reactor containment to permissible values in the event of an accident which could lead to an unacceptably high pressure within the containment.

The highest permissible pressure within the containment under such conditions varies from plant to plant.

In order to determine the maximum permissible pressure for Barsebäck the original design calculations are being re-examined, as well as carrying out some complementary calculations. The re-examination has included the reactor containment vessel, and other pressure boundary parts of the containment such as penetrations, locks, domes etc.

The containment was originally dimensioned for a design pressure of 0.5 MPa. Calculations show that it will withstand an internal pressure of 0.65 MPa in an essentially elastic manner, without takinȝ plastic deformation into account. This implies that the containment will remain virtually leak-light. This is in agreement with the original design premises and has been confirmed by control calculations both now and earlier.

The penetrations, locks, domes, etc in the reactor containment will also withstand a pressure of at least 0.65 MPa.

When the core melts large quantities of hot hydrogen are produced as the fuel cladding reacts with the surrounding water and steam. The gases flow out from the reactor pressure vessel and cause peaks in the temperature of the atmosphere in the containment. The strength of the concrete walls of the containment is not affected by the peaks in temperature because of their high thermal inertia, but som of the thinner steel penetrations may need internal thermal insulation to avoid such high temperatures that their strength is seriously affected. Continued studies will show if any further insulation is needed.

Various accident sequences which involve large increases in pressure within the reactor containment have been studied in order to identify those sequences which determine the size and capacity of the pressure relief channel. The calculations are described in more detail in Chapter 4.

A pressure relief are of about 1 m^2 seems to be able to cope with all the accident sequences, which have been studied, without exposing the containment to unpermissible loads. This area has been used therefore as a basis for the technical lay-out of the pressure relief system.

The pressure relief channel will be connected to the upper drywell of the reactor containment. The service dome, which is used for lifting heavy components in and out of the containment, would be a suitable point for the connection. The interference with the present containment structure would thus be kept to a minimum.

The service dome would be fitted with an extended casing which has an outlet for the pressure relief pipe. This would then be drawn down through the so-called square shaft to the bottom of the reactor building, and would then continue underground to the filtration plant.

In Barsebäck, which is situated on relatively loose soil the pipe will be laid in a concrete culvert which extends all the way to the filtration plant. The distance between the reactor unit and the filter is relatively short: 200 - 300 m.

In Ringhals the pipe will be connected to a rock tunnel which then conveys the steam to the filtration plant. The distance to the filter is different for the various reactor units. The connection between the pipe and the tunnel is located so deep that the necessary cover by good quality rock is obtained. The rock tunnel could, if necessary, be made impermeable to ground water leaking in. The other reactor sites have not been considered at this stage of the project.

Inside the reactor building some complementary radiation shields will be needed for the pressure relief pipe so that admittance to the power plant following an accident with pressure relief of the containment will not be unnecessarily difficult because of radiation hazards from the pipes.

Outside the reactor building the earth and rock will provide the radiation shielding which is required.

The rupture disc in the pressure relief pipe is placed as near to the reactor containment as is practical. The rupture disc is tight and can be tested at regular intervals using the isolation valves which are positioned downstream in the pipes. A rupture disc with the necessary dimensions and properties is available on the market.

The rupture disc has been chosen to open at 0.65 MPa, a pressure at which the leak-tightness of the containment is still virtually unaffected.

Alternatively two pressure relief systems could be connected to the service dome, via pipe systems with separate rupture discs and isolation valves. One of the systems would be dimensioned for accidents which require a large capacity for pressure relief, but relatively low demands on filtration (pressure relief prior to core damage). The other system would be dimensioned for accidents with limited demands on the capacity for pressure relief but high demands on filtration (pressure relief after a core accident). Such solutions are technically more complicated, which can mean that it is more complicated to verify their proper function.

5.4 Design of the filtration plant

5.4.1 General

The filtration plant consists of a condenser and a filter.

The filtration plant shall

- condense the steam which flows out of the containment during the initial stage of pressure relief, and the period thereafter;
- separate aerosols and elemental iodine from the mixture of gases and steam;
- delay the inert gases and the organically bound iodine.

Gravel, sand and water are suitable materials for this purpose.

5.4.2 Alternatives for a condenser

The first alternative consists of a crushed rock condenser with a sump for the condensate to collect. Secondly, certain studies have been carried out concerning the possibility of using a water condenser.

The crushed rock condenser is an entirely passive design without the need for an external cooling system, etc. The volumes and cross-sectional areas are chosen such that the heat capacity and heat losses to the surrounding earth, constitute a sufficiently large heat sink to condense all the steam from the initial stage of pressure relief and for a period thereafter.

In the first alternative for Barsebäck a crushed rock condenser with a volume of about 10 000 m^3 has been studied. As a comparison it can be noted that the volume of the containment of Barsebäck 1 is also about 10 000 m^3.

Even larger condenser volumes have also been considered, about 100 000 m^3, but were deemed not to be of interest for the present. A crushed rock condenser, of about 10 000 m^3, followed by a filter consisting of a water pool, of 100 - 200 m^3, see Figure 5.2 would appear to give a technically suitable and balanced solution.

A crushed rock bed, 10 000 m^3 in volume, is large enough to condense all the steam removed from the containment during pressure relief through the rupture disc, and, together with the condensation pool in the containment, to absorb all the steam produced by the decay power for about 24 hours.

As an alternative to the crushed rock condenser the possibility of using a water condenser has also been studied.

The water condenser can also be designed such that external cooling is unnecessary. It must in that case be relatively large, depending upon how long it must function completely self-sufficiently.

Since water is relatively easily cooled, compared with crushed rock, it is natural to place cooling circuits into a water condenser to take advantage of active cooling, which can continue the cooling process for up to a few day after the initial event. The volume can be reduced accordingly. The cooling system must be independent of the auxilliary cooling and electric power systems of the nuclear power plant since the cause of the accident could be the loss of these systems.

The water condenser could be designed as a condensation pool with a horizontal blow-down opening which gives a robust construction. The cooling loop is located inside the cylindrical pool. It is protected from the violent movements of the water, which can occur during pressure relief, by a cylindrical concrete sleeve.

The total volume depends upon how quickly after the accident the external cooling of the pool can be taken into account. The cooling capacity and the water volume can for example be chosen such that no external cooling is necessary during the first 24 hours after the accident. After that the external cooling can be achieved using a temporary system through previously prepared connections. The total water volume would in that case be about 4 000 m^3.

The water condenser is considered to be a less attractive solution than the crushed rock condenser. One of the main reasons is the difficulty of solving the design problems caused by the pressure increase resulting from the possible hydrogen deflagration in the condenser. The pressure peaks would be high, 3 - 4 MPa, in the gas filled space above the water, and they would be of extensive duration because of the large dimensions. It would thus be difficult to design a concrete construction to withstand such pressures. Siting underground, in rock, would be a possible solution to cope with the high pressures, but in that case a rock filled tunnel would be a more natural alternative.

5.4.3 Design of a crushed rock condenser

The crushed rock condenser has been designed in the form of a tunnel, for plants sited on rock such as Ringhals, see Figure 5.3, or a concrete cylinder, for moraine sites such as Barsebäck, see Figure 5.2.

In the case of the tunnel both horizontal and vertical systems have been studied, the latter has to date appeared to be more expensive without decisive advantages being demonstrated.

The steam and gases enter in the upper compartments of the tunnel and then flow down through the gravel bed. The steam condenses, the gas is delayed and filtered and then exits to the filter from the bottom of the crushed rock bed.

The lay-out of the system can be in the form of a row of parallel tunnels which are connected in series. The rock volume is thus utilized better since the steam and gases pass through as long a distance as possible between the inlet and outlet. The total volume of the condenser can be increased or decreased and adjusted to different sizes of reactors by choosing different numbers and lengths of parallel tunnels.

Any hydrogen explosions in the spaces above the crushed rock beds will be absorbed by the surrounding rock.

A concrete cylinder constitutes a suitable form for the crushed rock condenser for moraine, the sites in Barsebäck. The crushed rock fills the entire space up to the top slab which is made of concrete cast directly onto the bed. In this manner empty spaces are avoided in which hydrogen deflagration could occur.

The pressure relief channel is connected to a number of holes in the upper part of the concrete cylinder. These distribute the steam and gases to the gravel bed through which they then flow. The gases collect at the bottom and leave the condenser through a centrally positioned pipe.

The system can consist of one large or several smaller concrete cylinders, which can be connected in parallel or in series. It is also possible to introduce vertical labyrinth walls in the concrete cylinders and thus extend the length of the filter.

In all the versions of the crushed rock condenser the condensate collects at the bottom. For a condenser with a volume of 10 000 m^3 there is about 500 m^3 condensate. The lower part of the condenser is so designed that the passage of steam and gases to its outlet is not hindered by the condensate. The condensate collects in a sump or pump well designed to enable the highly radioactive liquid to be pumped out.

The condenser is positioned underground so that an even and natural drainage will occur from the containment to the condenser.

This means that the condenser will normally be situated below the ground water level. Any ground water which leaks into the pressure relief channel and the condenser will collect in the condenser sump, and will be removed by drainage pumps so that the condenser is kept free of water under normal conditions. Alternatively, in the case of a rock tunnel, the ground water level could be lowered locally by excavating small drainage tunnels and pumping equipment underneath and around the filtration plant.

In order to limit the pumping capacity which would be required, it could be necessary to make the surrounding rock, or parts thereof, less permeable.

Condensers in the form of concrete cylinders would not usually be subjected to much leakage from ground water, since the concrete walls can be made sufficiently impermiable to withstand the ground water pressure, and the pressure relief pipe is made of steel and laid in a concrete culvert.

The hydrogen formed during a core meltdown is transported through the pressure relief channel to the filtration plant where it is enriched and mixed with air and it may become inflammable.

Spontaneous ignition is then possible by for example the electric discharge of statically charged particles, such as water droplets, grains of sand, etc, or due to a local temperature increase due to the presence of radioactive isotopes trapped in the filter. When, how and where burning could occur in the plant is difficult to predict. The hydrogen can burn slowly, deflagration, which results in moderate pressure increases, or rapidly, explosion, with short, large pressure loads the magnitude of which depends upon the composition of the gaseous mixture, the strength of the ignition process, etc.

If the pressure relief channel and the filtration plant are filled with a non-inflammable gas such as nitrogen, expelling all the oxygen, hydrogen burning can be avoided completely. It may however be difficult to prevent air from entering the filtration plant through the chimney during certain periods after an accident. The containment buildings of the PWR units in Ringhals are in fact normally filled with air which would be removed to the filtration plant together with the steam and hydrogen after an accident.

The filtration plant has been designed therefore so that hydrogen deflagration may occur anywhere in the plant without jeopardizing its integrity or proper functions.

There are several methods available to ensure that hydrogen explosions do not occur and result in high pressure (3 - 4 MPa) of short duration in free volumes. They are all based upon the early, controlled ignition of the gaseous mixture, before the hydrogen concentration has become so high that an explosion could occur. This can be achieved using a spark plug, a heated filament, a catalytic recombiner, etc. An extensive programme for testing such equipment is currently under way in the USA. The systems all have a common feature however: that they are all dependent on active components such as power supplies, and control equipment, which means that the desired passive character of the filtration plant is partially lost.

A simpler and more reliable solution to hydrogen burning in crushed rock condenser is instead to design them so that all free volumes are avoided by entirely filling the condenser with crushed rock. Tests concerning the ignition of air-hydrogen mixtures carried out by the Research Institute of the Swedish National Defense have shown that the rate of burning in gravel beds, with the stone sizes of interest, can be limited considerably, as compared to burning in free volumes of gas. Thus the resultant pressures are so low that they do not constitute a design problem.

5.4.4 Filter

Even if the crushed rock condenser, according to the completed and planned investigations and calculations, appears to have good filtering properties, it is tentatively assumed that a separate filter should be installed after the condenser.

The filter unit may consist of a sand or gravel filter or of a small water pool.

The preliminary first choice is a small water pool. It can be used for two purposes.

Firstly, by a suitable choice of water chemistry, in the first hand elemental iodine, but also to a certain extent other elements such as cesium and strontium which pass through the crushed rock condenser, can be retained in the water permanently bonded. Secondly a cooling system can be connected to the water pool, so that the necessary temperature adjustment to retain the water chemistry conditions, or to prevent the escape of steam to the atmosphere even after the heat capacity of the condenser has been fully exploited.

Sand has well documented filtering properties. Complementary tests have been carried out for iodine retention, and further tests are planned.

The sand filter is designed in such a way that the sand will not become damp and clogged by precipitation the steam flowing through. Traditionally this is achieved by keeping the temperature of the sand bed at abut 100°C using electrical heating. Preliminary results indicate however that such equipment can be avoided by self drainage of sand with larger grains.

The filtering properties of gravel are less wellknown and therefore studies are planned, and in part already completed. Even if the retention properties of the gravel are not taken into account it could constitute a suitable buffer as a drying medium between the condenser and the sand filter. Alternatively self draining gravel could be the main filter.

5.4.5 Design of the filter

The volume of the sand/gravel filter will be small in comparison with the crushed rock condenser. As a basis to selected the dimensions of the filter the residual energy which is absorbed in the form of fission products is an important quantity. As the initial assumption that 90 % of the fission products in question (excluding the inert gases) are retained in the condenser and the remaining 10 % in the filter, the latter will need to have a volume of the order of 1 000 m^3.

The sand filter can in some alternatives be intergrated with the condenser, but it would probably be better to build the filter separately, see Figure 5.4.

The filter would be placed above the condenser so that the condensate can drain into the condenser sump.

Traditionally the sand filter consists of horizontal layers of stone, gravel and sand between which the grain size varies by about a factor of two. The direction of the flow is from the bottom to the top and the grain size decreases in the same direction. The final layer in the filter is normally coarser to prevent the fluidisation of the finer sand. The stone, gravel and sand layers are dimensioned taking into account strength, cleansing capability, particle retenion capacity and usable life. In the ideal case the coarsest layer will remove most of the larger particles, etc. The filter rests on, for example, a bed of hollow concrete blocks which act as a distributer for the inflowing gas, and the entire filter is built into a concrete container.

The flow rate of the gas is low, a few cm/s, and the pressure drop can be of the order of 100 mm water gauge.

The volume of the water filter is also small compared with the crushed rock or water condenser, about 100 - 200 m^3.

The water filter can at its simplest be a small water pool, to which the gas/steam mixture from the condenser is led through a pipe to a suitable depth under the water surface where the steam condenses as long as the temperature in the pool is below 100°C. The remaining gas is washed as it bubbles through the pool. At the entrance to the pipe there is a so-called silent boiler to ensure calm flow and condensation conditions in the pool and pipes from which the steam and gas flow.

Apart from this it may be possible to lead the gas/steam mixture directly into the gas volume above the surface in the water pool. The gas/steam mixture can, if necessary, be washed using a spray connected to the pool cooling system.

Simple dosing equipment is connected to the pool to control the water chemistry. During normal conditions a neutral water chemistry is maintained. Following an accident the necessary chemicals can be added to the pool. The control and adjustment of the water chemistry over long periods can be carried out independently of the other auxilliary systems of the reactor plant.

The problem of a hydrogen explosion is expected to be easier to solve in a small water pool than in a large condensation pool.

5.4.6 Thermal calculations

The conditions of heat transfer and fluid flow in the entire system, from the reactor containment via the rupture disc and pressure relief channel to the various parts of the crushed rock condenser, have been calculated for a number of different cases in which steam and gases flow from the containment to the condenser during the core accident. The flow rates exhibit wide variations as a function of time.

5.4.6.1 Pressure relief of the containment

As an example of the results of calculations concerning the pressure relief of the containment after a core accident, the pressure and temperature conditions are illustrated for a core accident caused by the loss of electrical power supplies to the nuclear power plant, see Chapter 4. The crushed rock condenser, volume 10 000 m^3, connected to the reactor containment by a pressure relief channel and followed by a reducing valve with cross-sectional area 0.1 m^2.

The results show that the pressure in the containment is reduced quickly with the proposed rupture disc area. After 20 - 30 seconds, the pressure in the containment is reduced considerably with a bursting pressure for the disc of 0.65 MPa.

The pressure drop across the gravel bed varies with its volume and length. The smaller the volume and the longer it is, the greater the pressure drop will be. In the case studied the pressure drop before the gravel bed will be at the most 0.3 - 0.4 MPa for a 40 m long bed.

5.4.6.2 Underpressure in the reactor containment

When the rupture disc opens most of the gas is blown out from the containment to the filtration plant. If one assumes an extended loss of cooling to the containment, the molten core will continue to produce more steam by boiling the water in the condensation pool, and eventually all the original gas will have been displaced by steam. After a sufficiently long time, about 29 hours for a condenser volume of 10 000 m^3, the temperature throughout the condenser will have reached 100°C, at which point all the gas in the condensor will be displaced by steam at 100°C.

If the reactor containment cooling system were to start up again at this stage, an underpressure would be established in the containment as the steam would condense. The steam would be sucked back from the crushed rock condensor, and be replaced by air from outside, which enters via the chimney and he vacuum breaker in the filter pool. The flow would not cease until all the steam in the containment and filtration plant had condensed and been replaced by gas.

The reactor containment in Barsebäck can withstand an underpressure of about 0.05 MPa.

Calculations performed to data indicate that the containment will not be subjected to underpressures of this magnitude.

The containment can under some conditions become filled by a mixture of air and the hydrogen remaining in the filtration plant. If this mixture were to ignite such a high pressure could be created in the containment that it could be damaged. The likelihood of such an event happening is however considered to be very small, since the upper part of the containment, where the potential sources of ignition such as electric motors, valves, etc, are situated, is continuously sprinkled with water from the containment cooling system.

The air/hydrogen mixture can be extracted via the filtration plant by introducing nitrogen into the containment.

5.4.7 The influence on other safety systems

The benefits from a filtered pressure relief system depend upon its capability to lessen the consequences of a core meltdown.

Another question in this context is the reverse- whether a filtered pressure relief system can be detrimental to safety in any respect. This question applies to safety during a core melt accident as well as to the accidents for which the nuclear power plant and its normal safety systems are presently designed. The important question whether the normal safety systems could be affected negatively by the pressure relief and filtratration systems has been analysed preliminarily. In this analysis the filtered pressure relief systems have been assumed to be designed as proposed in this report.

For any negative effect to occur at all the rupture disc must open inadvertantly in the event of a design basis accident, despite the assumption that the reactor safety systems are performing as designed, thus keeping the pressure in the containment well below the 0.65 MPa needed to open the rupture disc.

The possible negative effects which have been studied in connection with the accidents described above are:

- Increased radioactive release to the surroundings.

  The releases will be very small. When the rupture disc opens the radioactive gases which are present in the containment will flow into the pressure relief channel. Leakage from the containment occurs to the filtration plant, and thus the increased release of radioactive gases will be limited to those escaping through the chimney. Since the filter is so large it will take the inert gases a long time to reach the chimney, and they will probably begin to be sucked back into the containment before reaching the chimney.

- Contamination of the pressure relief channel and increased radioactivity in the reactor building.

  During most of the first day after the accident the pressure relief channel will contribute to increased radiation levels but after that it will become successively less serious. The increased radiation levels could require extra radiation shielding. An alternative to this is to evacuate the radioactive contents of the pressure relief channel into the filtration plant.

- Increased pressure loads on the reactor containment and its internal components.

  During a sudden pressure reduction in the containment, due to pressure relief, a large pressure difference across the intermediate floor, which separates the upper and lower compartments of the containment, will occur. This pressure difference is due to the initial compression of gas in the lower compartment which is then relieved through vacuum breakers in the intermediate floor to the upper compartment and further out through the opened rupture disc. Analysis has shown that the design limits of the floor will not be exceeded during this sequence of events.

- Impermissible underpressures in the containment, which affect the emergency cooling pumps, could arise if an operator were to realise that the rupture disc had inadvertantly opened after an accident, and he therefore closed the isolation valve, in series with the rupture disc, at the most unsuitable point in time: when most of the gas in the upper part of the containment had blown out into the pressure relief channel. Analysis shows that the resulting underpressure will not be below the design values for the containment, nor will the emergency cooling pumps function be reduced significantly.

- Increased risk for a hydrogen fire in the reactor containment as a result of air being sucked into the containment from the filtration plant.

  The gas which is eventuelly sucked back into the containment from the filtration plant after the rupture disc has opened, will consist mainly of the nitrogen which was originally in the containment, but it cannot be assumed that it will not contain any air, which is normally found in the pressure relief channel and the condenser.

  Analysis shows that the increase in the oxygen concentration in the containment is moderate, and that the probability of a hydrogen explosion occurring is not affected significantly. The probability for ignition to occur at all is, as previously mentioned in Paragraph 5.4.6, very low since the containment is continually sprayed with water from the containment cooling system.

In conclusion, the analyses which have been performed show that the introduction of a filtered pressure relief system, has very little influence on the normal safety functions performed by the safety systems within the frame of design bases accidents considered to date, i.e. those accidents described in current safety reports for nuclear power plants.

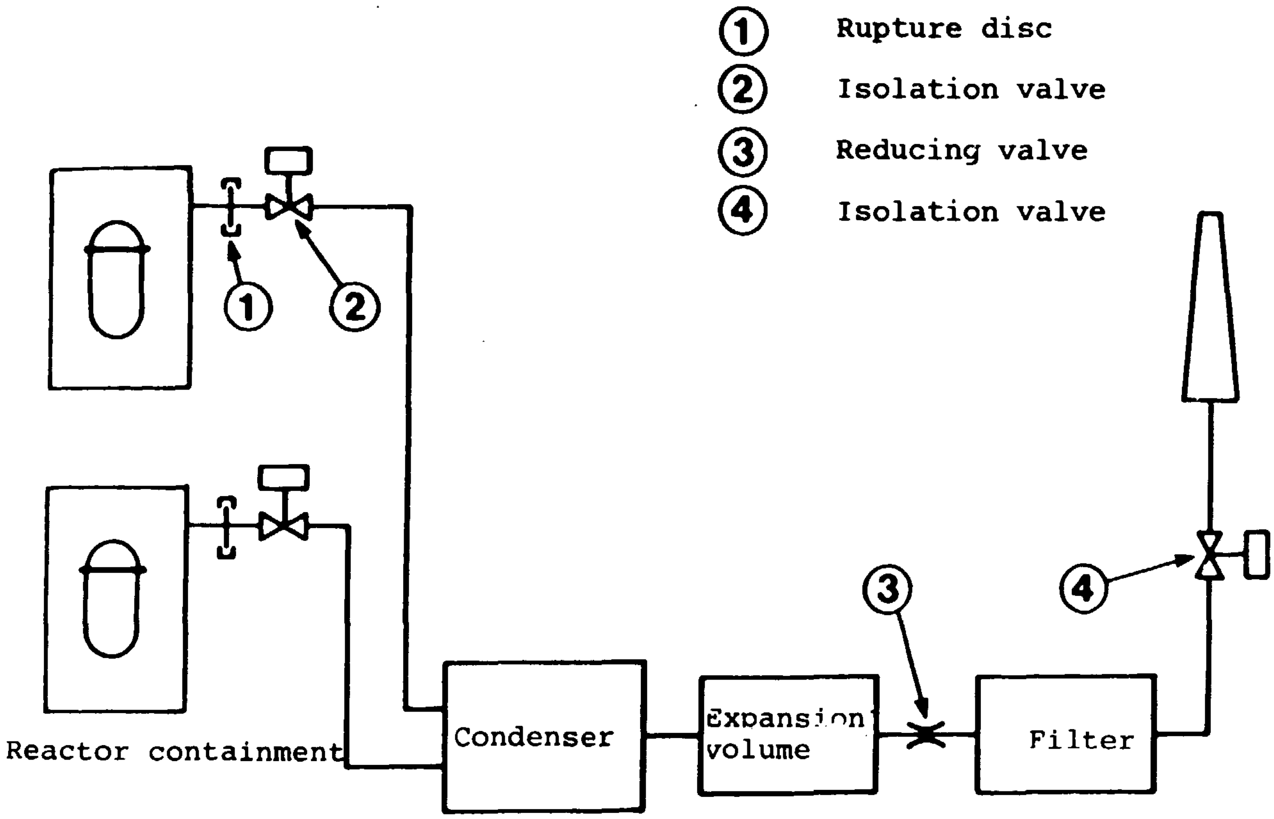


Figure 5.1

FILTRA - principle components.

FILTRA— FILTERED PRESSURE RELIEF OF A REACTOR CONTAINMENT

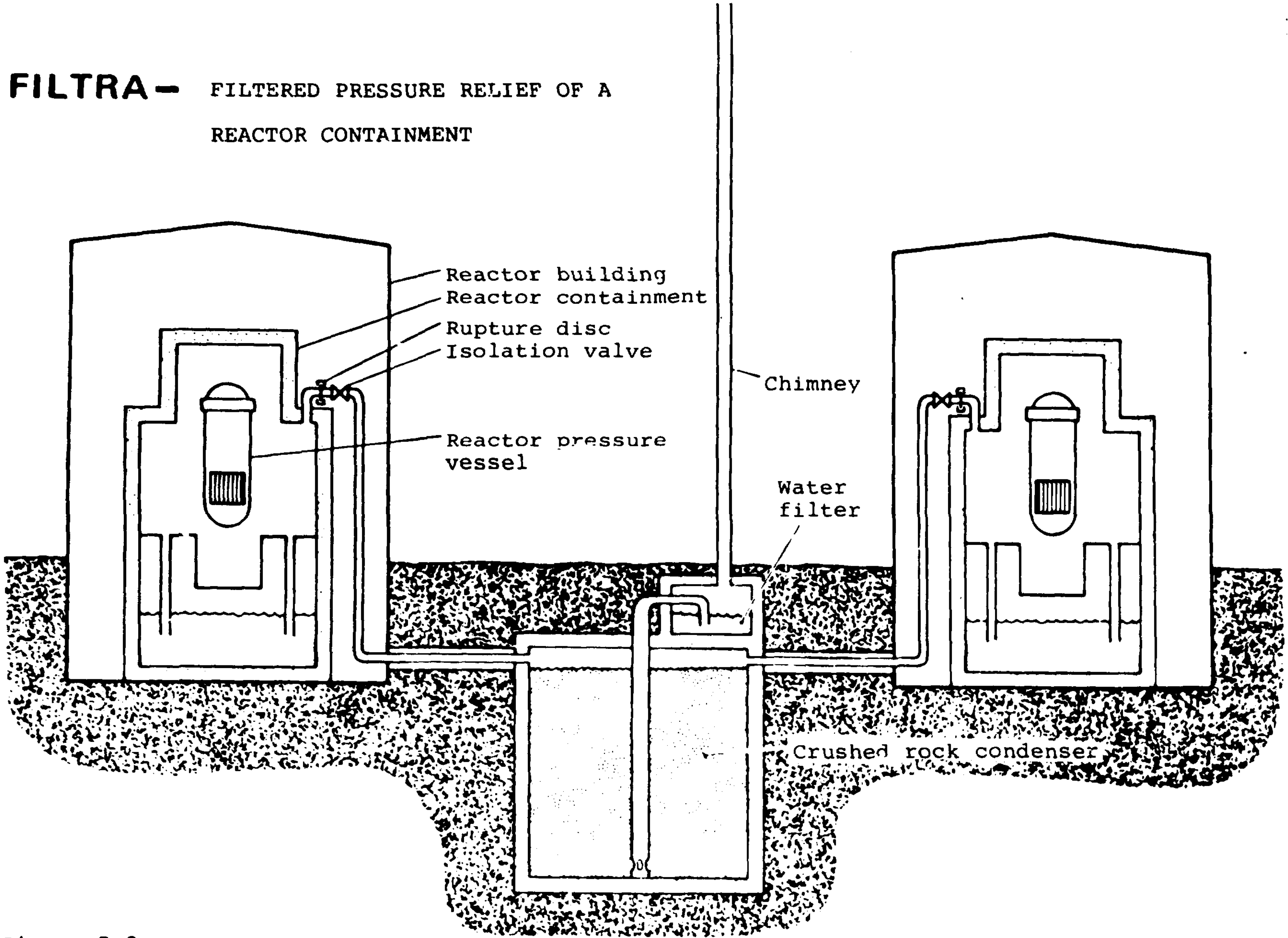
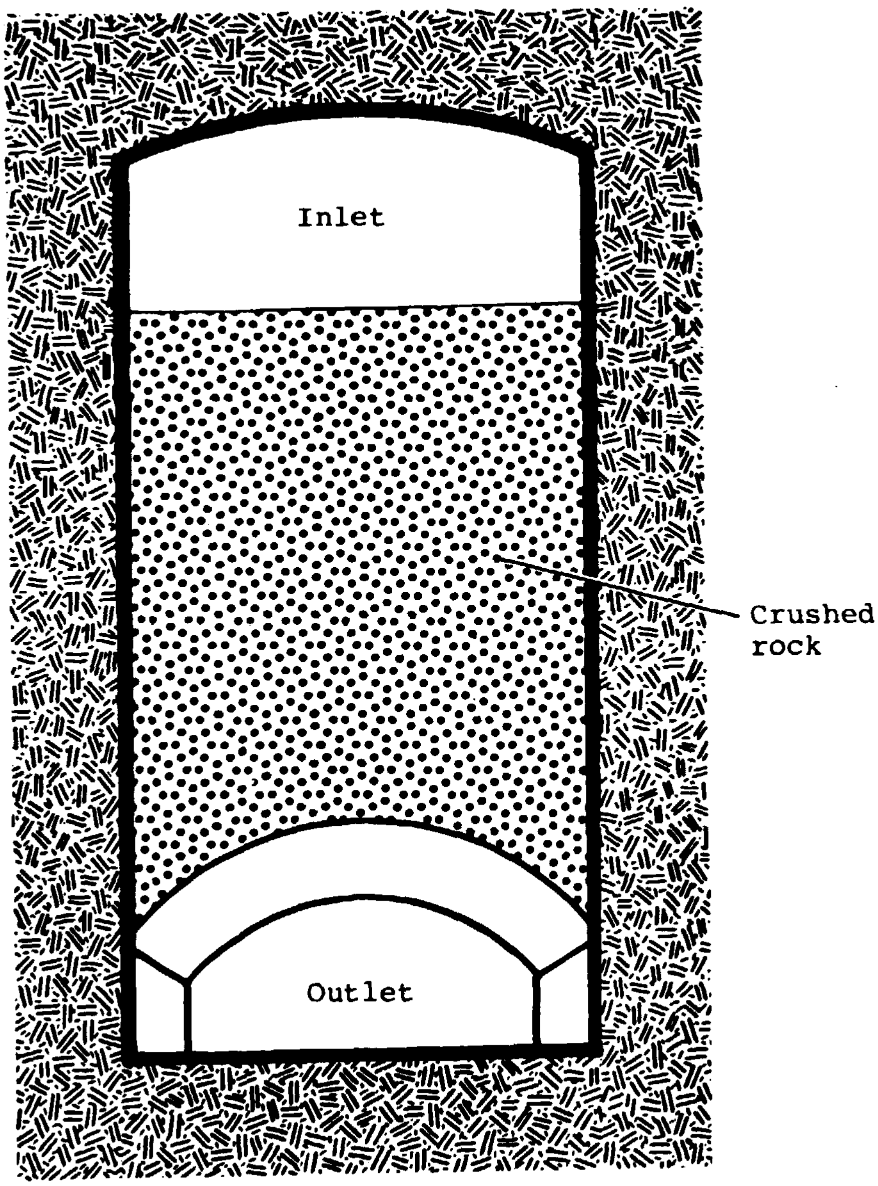


Figure 5.2



Figur 5.3

FILTRA - cross-section through the crushed rock condenser in a tunnel of rock.

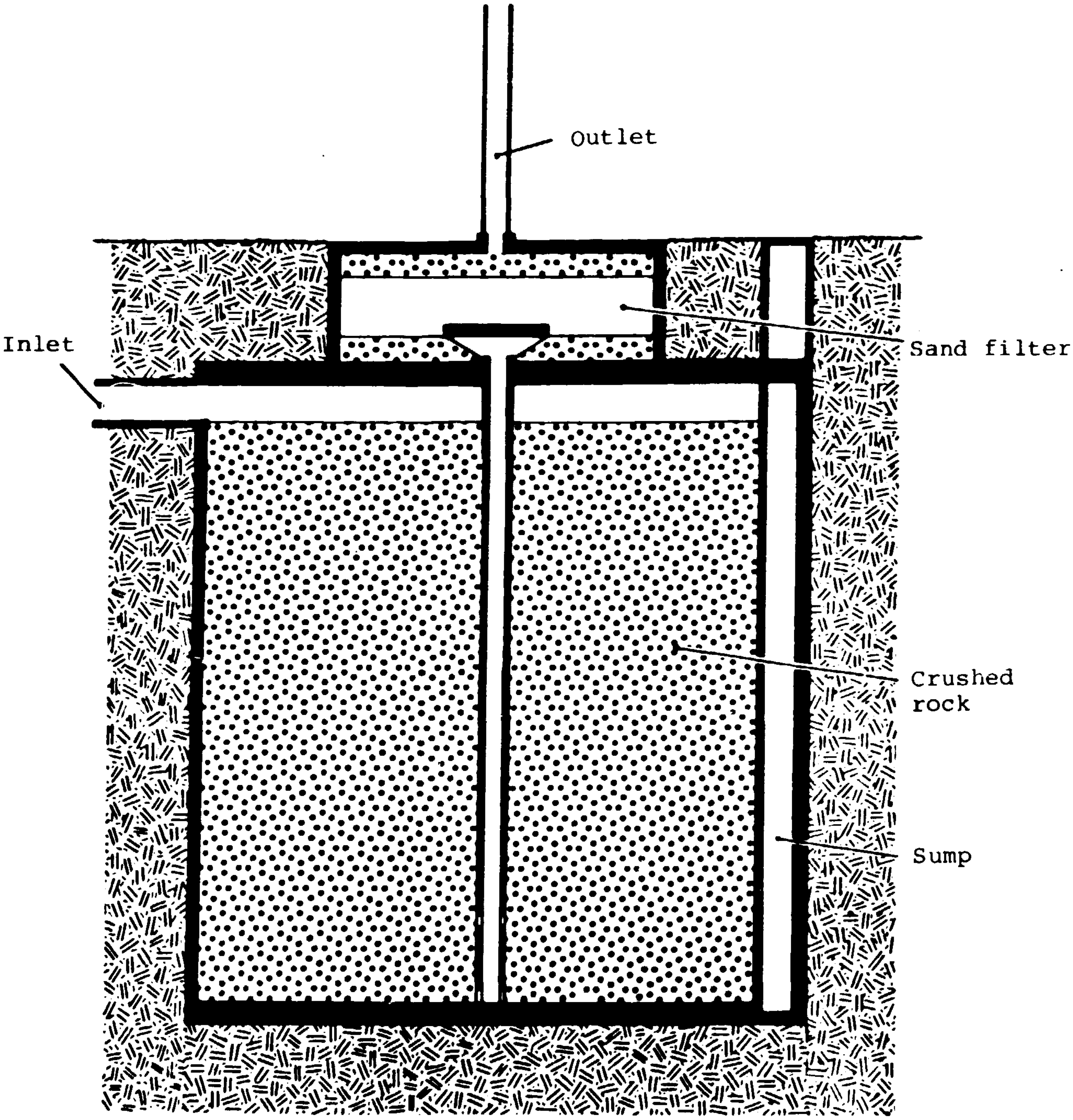


Figure 5.4

FILTRA - underground concrete cylinder for the crushed rock condenser and sand filter.

6. RELEASE OF RADIOACTIVITY TO THE ATMOSPHERE AS A RESULT OF PRESSURE RELIEF THROUGH THE FILTRATION PLANT

6.1 General

In this chapter an estimate is reported of how much of the radioactive elements which are released from the overheated fuel pass through the pressure relief channel to the filter chamber, how much is separated by the filtration plant, and how much passes through the filter and is discharged to the atmosphere through the chimney.

In this report the results of the first phase of the project are reported. This has concentrated on preparing the basis for an overall dimensioning of the filtration plant. In order to do that it is necessary to first estimate the amounts of material and heat which will be transported to the filter and the time scale for this. With regard to the transportation of radioactive gases and particles their bulk and radioactive content have been of the primary interest, followed by their tendency to deposit on surfaces in the reactor containment, pressure relief channel, crushed rock condenser and the filter.

Recently it has been pointed out, in the German reactor safety study and in the American debate, that the condensation of steam on particles, and the binding of iodine and cesium in solution in water, can result in a quicker and more effective separation than has previously been assumed in the assessments made in this report. The debate is referred to in Section 3.3 and will be evaluated in more detail in the next phase of the project.

6.2 Release from the fuel

The purpose of the filter is to separate and retain the radioactive substances, which can give large doses to the population and long term ground contamination after an accident. The filter chamber is dimensioned for the most extreme accident conditions, including core melt accident (se Chapter 4).

During this sequence of events radioactive substances are released from the fuel when the cladding fails (at approximately 1 000 - 1 300°C, gas release), when the uranium dioxide becomes overheated and melts (melt release, at approximately 1 700 - 2 500°C), and when the melt runs over the concrete floor and is penetrated by steam released from the concrete (vaporization release).

During this sequence gas borne smoke and solid particles, aerosols, leave the melt. They originate mostly from the fuel, structural materials and the concrete. The gas borne particles are estimated to total a couple of tons.

The radioactive substances constitute a small portion of the total amount of the particles. The long lived radioactive substances, which are of interest when considering ground contamination, are cesium and strontium and they weigh about 70 kg.

Another group of radioactive substances are the halogens iodine and bromine, which are to a large extent released from the heated fuel. The halogens are chemically very reactive and can therefore be chemically bound to the gas borne particles, which act as carriers for them.

The halogens are volatile and can exist as gases even at ambient temperatures. The gaseous halogens are relatively easy to separate in a filter. They can, for example, be captured in a crushed rock bed, a sand filter or a water pool.

Halogens can however also react with organic substances to form compounds which are difficult to filter (methyl iodide). In this report it is assumed that 1 % of the halogens are present in the organic form and which to a large extent passes through the filter to the surroundings.

A large fraction of the inert gases (krypton and xenon) is released from the fuel when it is heated above about 1 700°C. They are, as their name implies, nonreactive and are therefore relatively difficult to remove by filtering. It is assumed therefore that they pass through the filter and are released to the atmosphere. However the delay which occurs because of the volume and flow rate conditions of the process is taken into account.

6.3 Transportation to the filter

The filtration plant consists of a condenser followed by a filter. The separation calculations carried out to date have only considered the condenser. It has been shown that the physical processes in the condenser are strongly dependent upon the flow rate of the gases, and thus also on the condenser volume. For low flow rates the gases and particles are separated in the reactor containment and condenser by

- sedimentation
- diffusion
- water spray
- cleansing in the pool.

At higher flow rates the importance of sedimentation and diffusion decreases rapidly, but a certain amount of filtration occurs in the crushed rock condenser by impactation.

The above processes also occur in the reactor containment if the flow from it is sufficiently slow. Therefore the entire system must be considered as one unit, but this is only possible when using the computer program CORRAL (41), which only includes simplified models of the separation processes. CORRAL was developed for the American reactor safety study (1). The program has been revised in Studsvik so that more recent results from research in Germany and the USA can be incorporated, in which separate aerosol processes have been studied in detail. In particular the condensation of steam on the particles has been developed in the process. This program and the American program HAARM-3 also calculate particle coagulation. Both of these phenomena are decisive with respect to particle size, which in turn determines the rate of sedimantation and diffusion. The programs can however only consider one room or compartment, and the results have therefore been used as input to CORRAL. For dry particles NAUA and HAARM-3 give virtually the same results, that the mean particle diameter increases in about 10 minutes from 0.2 μm to 4 - 5 μm or more. In wet steam the mean diameter can rapidly be 14 - 30 μ. It is therefore important to be able to determine the thermodynamic conditions in the various rooms. In the calculations using CORRAL the particle sizes have been adjusted for these conditions.

CORRAL has been used in calculations of a system with six rooms, consisting of two rooms in the containment, one in the outlet channel and three in the condenser. The condenser volume was large. In the main calculation the follwoing assumptions were also included:

- 2 435 kg of particles were released during the melt accident, as well as 4.8 kg of iodine and 400 kg of inert gases.
- The pressure relief to the filtration plant started after 1.5 hours.
- None of the aerosols or iodine were removed by cleansing in the condensation pool within the containment.
- The filtration plant was isolated for at least 5.5 hours.

In this case about a third of the particles and half of the iodine were deposited in the containment. If cleansing in the containment pool occurs these fractions will be much larger. The most important result of the calculations was however the large deposition of particles and iodine in the crushed rock condenser. In the most unfavourable case: assuming that none of the steam condenses on the particles, the atmosphere in the condenser after 5.25 hours contained 0.01 % of the particles and 0.2 % of the iodine. This was of course based on the presumption that there was no release from the condenser during this period. One can obviously permit discharge from the condenser to the subsequent sand and water filters. This situation will be studied in more detail during the next stage of the project.

6.4 Separation in the filter

The filtration plant contains both the condenser and the filter. The condenser is a crushed rock bed of about 10 000 m^3 in volume. The filter will be either a sand bed or a water pool. In this section the filtration capabilities of two combinations will be discussed:

- a crushed rock condenser and a sand filter;
- a crushed rock condenser and a water pool.

Experiments have been carried out in Studsvik to study the capture of air borne droplets in crushed rock and sand beds. The results will be compared with those available in the literature and a model developed (31) to describe the capture for particles in crushed rock and sand beds due to:

- sedimentation
- impact
- diffusion.

Experiments have also been carried out to study the removal of iodine and methyl iodide by crushed rock and sand beds. The results show that iodine is removed to a certain extent by crushed rock beds, and relatively effectively by sand beds, but that only insignificant amounts of methyl iodide are removed (19)

The results of these studies concerning the retention of particles, iodine and methyl iodide by crushed rock and sand beds have been combined into a model for determining the filtration capability, which is a result of the first phase of the project. In the next phase the model will be used to dimension the crushed rock condenser and the sand filter.

During the next phase the filtration capability of the combination of a crushed rock condenser and a water pool will be studied. The retention of particles, iodine and methyl iodide in water pools is considered to be sufficiently well treated in the literature.

The aim of dimensioning the filtration plant is to show that it is capable of substantially reducing the release of particles and iodine, as well as showing that it will delay the release of inert gases and organically bound iodine. This implies that a release of class A could be reduced to a release of class B or lower, as defined in Table 3.2. The results of the studies carried out up to now indicate that it will be possible to achieve this goal with technically reasonable system combinations of crushed rock condensers, sand filters and/or a water pool.

6.5 Retention of radioactive substances in the filter

The radioactive substances, which are retained in the filter must be kept there for a sufficiently long time, so that the radioactivity decays. Half of the radioactivity has decayed after one half life, and after ten half lives almost all the radioactivity has decayed. Iodine is of particular interest here with its half life of 8.05 days and cesium with a half life of about 30 years. Radiocative iodine will therefore disappear within a few months whilst radioactive cesium will remain for a couple of hundred years.

The results of the estimates made to date of the chemical properties of these substances show that they are both soluble in water. Iodine will be present in water mainly in the hydrolized form, and to a lesser extent as elemental iodine. The hydrolized compounds of iodine are strongly bound to the water. Even if the water is heated to its boiling point the hydrolized iodine will remain in solution, and only a few per cent of the elemtal iodine will evaporate with the steam. Only if the water solution dries up and the iodine is oxidized can it be dissipated in the air borne form.

Cesium compound remain strongly bound to the water and do not evaporate with the steam even if the water boils. Cesium will be released only if the water evaporates and the residual salt is heated. The crushed rock and sand filter can be kept cool by keeping them covered with water. The iodine and cesium can thus be retained in the filtration plant for the necessary length of time.

6.6 Estimation of the release to the atmosphere

The material available, after the first phase of the project, shows that the release through the filration plant and chimney of radioactive particles and iodine to the atmosphere, for the accident cases studied, can be considerably reduced as compared to the situation without a filtration plant.

The inert gases and a small amount (about 19 %) of the iodine, mostly as methyl iodide, will be released to the atmosphere. The release of these substances can be reduced somewhat by delaying them, the extent of which depends upon the volume and flow rate conditions.

6.7 Assessment of leakage to the ground water

An accident sequence, which can result in a core meltdown, can also result in the release of radioactive substances to the ground water underneath the nuclear power plant, if the molten core subsequently penetrates the bottom slab of the containment. Alternatively the molten core could be cooled by the water which collects in the bottom of the reactor containment without melting its way through.

In both cases however it is prudent to assume that there will be a leakage of radioactive water from the reactor containment.

Some of the radioactive substances, which are captured in the filtration plant after pressure relief, can also leak to the ground water. Such a leakage is however deemed to be small compared with the leakage from the reactor containment after a serious accident, in particular if the containment has been damaged as a result of excess pressure (which is presumed in the accident sequences studied if pressure relief is not available). These comsiderations will be studied in more detail in the next phase of the project.

7. QUESTIONS SPECIFIC TO PWR

7.1 General

The accident sequences analysed in previous chapters and the pressure increases discussed were relevant for BWR, such as those in Barsebäck (1 and 2). For the other Swedish BWRs the sequence of events will be analogous in principle, although there may be quantitative differences according to reactor size in terms of thermal power output, and also due to differences with respect to the volumes and other design details of the containments.

For a PWR there are some fundamental differences compared with a BWR. These affect the sequence of events which determine the dimensions of the system for pressure relief via the filtration plant. The most important of these differences are:

a) The containment of a PWR is dry and has a much larger volume than a BWR.

b) The PWR containment is filled with air. Hydrogen fires, which give rise to rapid pressure increases, can occur if hydrogen is formed in the accident.

c) The primary circuit in a PWR is designed for significantly higher pressures than in a BWR.

c) The amount of fuel in the core of a PWR is appreciably less than in a BWR

In order to show how these differences affect the design of the pressure relief system and the filtration plant, the Swedish State Power Board has, in co-operation with the FILTRA project, started complementary studies using Ringhals 3 as the reference unit. A progress report from these studies has been prepared (40).

Ringhals 3 was chosen as the reference unit because it is (together with Ringhals 4) the largest of the reactors in Ringhals. The thermal power of Ringhals 3 (and 4) is about 15 % higher than that of Ringhals 2. (Ringhals 1 is a BWR the size of whi is roughly mid-way between Barsebäck 1, nd Ringhals 3, 4.) A comparison of the refe nce plants for both BWRs and PWRs is give n Table 7.1.

7. Sequence of events during an accident, pressure increases

Calculations of the sequence of events of an accident in Ringhals 3 has been commissioned by the FILTRA project from KWU, FRG. These calculations are not yet complete, and it is too early to draw any quantitative conclusions.

The preliminary results are reported in a progress report (40). The project has also had access to similar calculations for the American PWRs Zion and Indian Point. Some extracts from those results are presented in (40).

Based on the available material it is evident that there are, primarily, two fundamental phenonomena which can lead to rapid increases in pressure in the containment of a PWR in the event of a core meltdown. The first is the release of large quantities of hydrogen which is ignited and burns rapidly in the containment. The other is the formation of steam in a violent reaction between the molten reactor core and water in the reactor pressure vessel or containment.

7.2.1 Hydrogen burning

Hydrogen can be produced in several different reactions during core accidents. The most important is the reaction between water or steam and the zirconium alloy of the fuel cladding. The burning of such hydrogen in Ringhals 3 has been the subject of a special study which has been reported previously to the Swedish Nuclear Power Inspectorate.

The reactor core in Ringhals 3 contains 18.8 tons of zirconium which if all of it were to be oxidized would produce 830 kg of hydrogen. If this is homogeneously mixed with the air in the containment it is sufficient to result in deflagration. There is however insufficient hydrogen for a detonation to occur.

Calculations show that the containment can withstand a static pressure of 0.69 MPa. The calculations carried out show that only if more than 90 % of the zirconium is oxidized can the hydrogen burn, and under certain specific conditions contribute to the above mentioned pressure (0.69 MPa) being exceeded. Amongst other things the hydrogen cannot burn if the partial pressure of the steam exceeds about 0.2 MPa, corresponding to a total pressure of somewhat more than 0.3 MPa in the containment.

The effect of localized hydrogen detonations due to an inhomogeneous mixture of the hydrogen and air is currently being studied for Ringhals 3. The results obtained so far indicate that the containment can withstand relatively powerful localized detonations.

Hydrogen can also be generated in reactions between water and other metals than zirconium such as iron and chromium.

The conditions under which hydrogen will burn or detonate are different in different types of accident sequence. Thus, as already mentioned, the mixture of hydrogen-air-steam must fulfill certain conditions such as requiring that the steam concentration is not too high. This means that a violent hydrogen fire occuring simultaneously with a rapid formation of steam appears to be a relatively improbable phenomenon.

The following questions are crucial to the influence which hydrogen can have on the sequence of events during an accident:

- How quickly is hydrogen generated in the reactor pressure vessel and containment?
- How much hydrogen will leave the primary system before the core melts its way through the reactor pressure vessel?
- How is the hydrogen distributed in the containment?
- How and when dows the hydrogen burn?

These questions are the subject of considerable research efforts in several countries. More results from current and continuing programmes are required to enable an estimate to be made of how much pressure relief is necessary to prevent, or mitigate the effects of hydrogen fires.

7.2.2 Violent steam production

Studies carried out to date show rapid pressure increases due to sudden steam production, which is to be expected partly if the molten core were to fall into the water at the bottom of the reactor pressure vessel, or if it penetrates the pressure vessel and falls into the water which collects in the bottom of the containment.

Calculations show that it is the second of these two situations which results in the most severe strains on the containment.

During certain accident sequences in which the primary circuits is intact prior to penetration of the reactor pressure vessel, the steam production increases considerably because the emergency cooling water from the accumulators can be emptied at the same time as the pressure vessel fails. This case is specific to PWRs.

Calculations which have been made for American PWRs show that during some accident sequences considerable pressure increases occur in the primary circuit when the core falls into the water in the bottom of the reactor pressure vessel. There is thus a risk that the tubes in the steam generators will rupture, which in turn could result in the release of radioactive gases to the surroundings via the safety relief valves in the secondary circuit. Such a release cannot be prevented or reduced by relieving the pressure via a filter chamber.

The work in progress will provide an improved basis for quantitative estimates of the pressure transients in Ringhals 3 due to violent steam production.

7.3 Design considerations

Preliminary studies have shown that there are relatively good possibilities of connecting a pressure relief channel to the PWR plants in Ringhals. The principles of various alternatives are discussed in Chapter 3 of (40). Pressure relief openings with an area of 1 m^2 or more can be obtained by connecting to the equipment hatch on the cylindrical portion of the containment at ground level.

Table 7.1

Comparison of important parameters for the reference plants.

Type	Units	BWR	PWR
Plant		Barsebäck 1 (2)	Ringhals 3 (4)
Thermal power	MW	1 700	2 783
Operational pressure in primary circuit	MPa	7.0	15.5
Total volume of water within the containment	m^3	2 200	468
Water volume in condensation pool	m^3	2 000	-
Volume of free gas in containment	m^3	8 000	50 800
Quantity of UO_2	ton	89	82
Quantity of Zr	ton	34	18.8
Design pressure for containment	MPa	0.5	0.42
Static failure pressur for containment	MPa	~ 0.8	~ 0.69

7.4 Safety analysis

The introduction of pressure relief via a filtration plant can have som negative effects on the normal safety functions in the reactor plant. Some work has been started to analyse these questions for Ringhals 3. The studies are described in Chapter 4 of appendix (40) of the Swedish report.

The following summary has been made based upon the American studies.

A. Premature or inadvertant pressure relief in connection with an accident (which oterhwise falls within the design basis for the plant) may have the following consequences:

- the safety functions which are initiated by high pressure in the containment is delayed or fail to work - for example the containment spray.
- the time for the reflooding of the reactor pressure vessel following a large pipe failure (large LOCA) is prolonged. This results in a higher maximum cladding temperature.
- lower suction pressure for the pumps during the recirculation cooling phase.

B. Pressure relief of the containment can result in considerable water losses due to steam blow off. This may be detrimental to core cooling (or the molten core).

C. Pressure relief results in air (and other non-condensable gases) being blow off. this means that there is a risk for an under-pressure occuring if the containment cooling starts at a later point in time. Preliminary calcualtions show that for unfavourable cases the containment might not be able to endure the low pressure.

D. A direct effect of the filter or the pressure relief channel could occur because

- pressure waves from an explosion or similar phenomena reach the containment.

- air is admitted to a containment which contains hydrogen and thus constitutes a risk for a hydrogen fire.

8. PROGRAMME FOR FURTHER WORK

8.1 General

The plan of work includes, according to the project proposal, 16 tasks which will be studied further during the next year (phase 2). After that the results will be analysed and summarized in a final report during the following six months (phase 3).

The heat transfer and fluid flow experiments which will form the basis for designing the crushed rock condenser are an important task during phase 2. The measurements of pressure drop and drainage conditions during the flow of various mixtures of steam and gases are of particular interest. These parameters are strongly dependent on the degree of compaction of the crushed rock bed; the size and quality of the stones; the shape of the inlets, the mass flow rate per unit area; and other variables.

In addition filtration experiments will be carried out to form the basis for dimensioning the crushed rock and sand filters, so that they fulfill specified requirements for separation of particles and iodine.

In conjunction with the above tasks both Swedish and foreign consultants will be engaged to supply information concerning results from the current research programmes abroad.

8.2 Work planned for phase 2

The descriptions of the core meltdown sequence, and the penetration of the melt through the reactor pressure vessel, via the concrete floor or similar to the bottom of the containment, will be expanded. It is intended that the descriptions of the associated pressure increases and their rate should be based upon realistic physical processes, according to available estimations. The individual positioning of pipe penetrations, doors and drainage arrangements for the intermediate floor, etc, will be based upon the specific containment design, and will be considered in the analysis.

A critical assessment of the calculation models to describe these sequences of events will be carried out, with the help of both Swedish and foreign consultants.

Sensitivity studies will be performed to assess the effect on the results of essential assumptions and uncertainties in the models. Examples of such uncertainties and assumptions are the amount and rate at which steam is formed when the melt runs down into water pools of various sizes; the rate at which the melt progresses downwards or falls; the possibility of cooling the molten and solidified fragments on the bottom of the water pool; the possibility and extent of continuing attack on the melt the bottom of the water pool; the type and extent of gas production, in particular hydrogen when the core melts and when it attacks concrete.

The possible means to influence the rate at which the melt runs down, and the protection of important components in the containment will also be studied. The amount of hydrogen which forms depends upon the access to steam during the melting process in the reactor pressure vessel and whilst the molten core attacks the concrete.

A model will be developed to calculate the particle separation in crushed rock beds during the simultaneous condensation of steam (diffusiophoresis). The model will be based on information available in the literature. This material will be developed further on the basis of experimental work within the project.

Models will also be developed to calculate the separation of iodine, cesium iodide and ruthenium. The goal is to develop a complete calculational model for the separation of elemental iodine in crushed rock and sand filters as well as in water scrubbers. This development will be based on literature studies, draft models presently available, and experimental results. In this goal is included a literature study and analyses of the volatility of hypoidius acid and conversion of elemental iodine and methyl iodide. Based on the literature studies experiments will be performed to analyse the potential release of cesium iodide and the separation of ruthenium tetroxide in the filter system.

The release, transportation and separation of radioactive substances will be studied using a model, developed especially for the purpose, which estimates the release of radioactivity through the filter via the chimney to the atmosphere, for each accident sequence. The aim is to calculate the nuclide transportation through realistic FILTRA systems by simultaneously considering all the important transport mechanisms. This provides a possibility of optimising the plant with respect to effectivity, size and choice of filtration principles, which provides a basis for a better cost estimates.

8.3 Experimental work

8.3.1 Heat transfer fluid flow experiments planned in long crushed rock beds

Experiments concerning the condensation of steam in crushed rock beds have previously been performed on a small scale.

To study during conditions condensation in long crushed beds of high mass flow, an experimental programme will be carried out, in which the heat and flow conditions in long crushed rock beds will be examined with mass flow densities and steam/gas mixtures of relevance.

The experimental crushed rock bed contains full size stones but is scaled down perpendicular to the flow direction. The bed is 30 m long.

The parameters which will be studied are, amongst others:

- the effect of a pressure drop across crushed rock beds of varying degrees of compaction.
- the significance of the stone size and quality with respect to flow through the bed.
- the effect of the length of the bed with respect to the pressure drop.
- the temperature distribution in the bed.

The aims and experimental details are the subject of a separate report (38).

8.3.2 Hydrogen burning in gravel beds

Experiments concerning hydrogen burning carried out in phase 1 to study the pressure damping effect of crushed rock beds have given positive results. A limited continuation of the tests is necessary to study the possible effects of larger bed volumes and larger stone sizes.

This task is concerned with obtaining results from corresponding projects in the USA and FRG, as well as obtaining results from their current programmes.

8.3.3 Particle separation in a crushed rock bed with simultaneous condensation of steam

The work programme for theoretical studies of particle separation in condensed steam shows that there are considerable difficulties in determining the relationship between the many variables which affect the separation. Application of experimental results to other geometries, flow rates, particle properties, etc, cannot therefore, as it would appear at the present, be carried out with any great certainty. The experiments must therefore be as near as possible to reality. The most time consuming and expensive consequence of this is that a realistic test aerosol must be produced. For this typical materials from a reactor core materials must be heated to a high temperature in an atmosphere of steam, and must subsequently be diluted with air and steam, and aged.

Iodine can be expected to exist in particulate form, as cesium iodide. The distribution of the particle size should be investigated, since this can affect the importance of the separation of small particles.

A generator for enough air borne material (in the containment the concentration is expected to be about 100 g/m^3) with an associated heated ageing volume must be built.

8.3.4 Separation of iodine in a crushed rock bed

In phase 1 some preliminary tests have been made on the separation of iodine in sand beds (Brogårds sand 0.8 - 1.2 mm) and in a crushed rock bed (16 - 32 mm) (19).

Experience to date indicates that the sand filter can be used for separation of elemental iodine. Continued studies are necessary, however, partly at low, and partly at high temperatues, in dry and wet environments with the aim of quantifying iodine separation.

Experience also indicates that the separation of elemental iodine in crushed rock is measurable. It is however necessary to obtain results from complementary measurements before a quantitative separation can be taken into account.

8.3.5 Experimental programme

Laboratory experiments to determine the numerical values of the adsorption coefficient and kinetic constants for sand and crushed rock for the theoretical models.

Parametric studies will be carried out for the flow rate, bed length, iodine concentration, humidity, temperature, and grain size. Crushed material of granite (gneiss), chalk and sandstone will be tested. The experiments will be carried out in a 1.5 m high column and in the existing 6 m long crushed rock bed.

Studies will be made of the chemical sorption of iodine on various materials. Equilibrium conditions and the kinetics will be studied. Comparable parametric studies as under Pcint 1 will be carried out. The studies will be expanded to include special studies on for example slag, Leca and scrap metal.

Retention of elemental iodine and its conversion to methyl iodide at temperatures of about 105 to 110°C (slightly superheated steam).

The experiments will be performed in a special steam bed in the heat transfer laboratory.

Transfer to the gaseous phase and separation of cesium iodide and ruthenium at high temperatures in an oxidising environment.

The crushed rock bed will be doped with cesium iodide. The release of iodine will be measured during the tests at high temperatures and with hot air flowing through the bed.

The experiments will be carried out on a laboratory scale.

8.4 Survey of the work to be carried out during phase 3

The water borne leakage of radioactive substances from the reactor containment and filtration plant will result in individual and collective doses. These will be estimated taking into account the dispersion conditions for ground water in the neighbourhood of Barsebäck and Ringhals.

Through collaboration with authorities, research institutions and consultants in primarily the USA and FRG the results of current international research work will be collected and evaluated.

The results of the project work from phases 1 and 2 will be summarized and analysed in a safety report. This will present an estimate of the doses to the surroundings, and an estimate of the ground contamination after some accident sequences involving core meltdown and filtered pressure relief. These estimates will be compared with those obtained in the case in which there is no pressure relief and filtiation.

Preliminary estimates of the costs of introducing filtered pressure relief will be presented and estimated related to the estimated dose reductions.

The results will be summarized in a final report including appendices containing revised version of relevant technical reports.

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