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REACTOR SAFETY ANALYSIS OF THE BARSEBÄCK 2 BWR PLANT
A COMPARISON OF SOME RECENT STUDIES

Christian Gräslund

Abstract

Recently, two different reactor safety studies of the Barsebäck 2 BWR plant were concluded in Sweden. One was sponsored by the Swedish Nuclear Power Inspectorate and performed by Studsvik Energiteknik AB. The other study was sponsored by the governmental Energy Commission and performed by the American consultant firm MHB Technical Associates. Both studies have used the WASH-1400 method of analysis. In the Studsvik study the probability for core melt accidents have been calculated, as well as the impact of large releases of radioactivity. The MHB study, which is critical of the WASH-1400 report, covers in addition the question of release categories. The methodology and results of these studies are valuable tools for the Swedish safety authorities to use in their regulatory work.

Address: Swedish Nuclear Power Inspectorate
Box 27106
S-102 52 Stockholm
Sweden

Introduction

In the beginning of 1978 two different reactor safety studies of the Barsebäck 2 BWR plant were concluded:

- "Accident Study of Barsebäck 2"

- "Swedish Reactor Safety Study. Barsebäck Risk Assessment".

The first one, sponsored by the Swedish Nuclear Power Inspectorate, was performed by Studsvik Energiteknik AB with Olov Norinder as project manager. The second one was ordered by the governmental Energy Commission's Expert Group on Safety and the Environment from the American consultant firm MHB Technical Associates.

The Objectives of the Studies

The objective of the Studsvik study was to provide the Inspectorate with a basis for evaluating if the conclusions in the American report WASH-1400 are applicable to a Swedish reactor design at a specific Swedish installation site. The study was intended to illuminate:

1. Differences between the system construction in the reference plant of WASH-1400 (Peach Bottom 2) and Barsebäck 2. Possible weaknesses in the Swedish design.
2. Applicability of the methodology of WASH-1400 in the analysis of disturbances in nuclear power plants, in the planning of safety research, and in the considerations of backfitting measures.
3. Possible radiation doses to the environment in the event of large-scale accidents in Barsebäck 2.
4. Selection of parameters in WASH-1400. Additional differences between the calculational models used in the Studsvik study, the Rasmussen study, and the Danish Risø-356 study.

The Studsvik study was not intended to provide answers to questions regarding the safety of nuclear power in relation to other sources of energy. Therefore it does not deal with medical, social, and economic effects resulting from radioactive releases from conceivable accidents.

The Energy Commission was instructed by the Swedish Government to present material which could serve as a basis for a decision regarding the supply of energy in Sweden in the 80's. One task was the study of the risks to the environment and public health connected with each kind of energy considered. The Commission began its work in January 1977, and in March and June 1978 two reports were delivered to the Government.

The Expert Group on Safety and the Environment ordered a study parallel to that supported by the Inspectorate to obtain source material for their report to the Commission. In order to achieve as much independence as possible, the Expert Group chose the American firm MHB Technical Associates, whose members have a documented critical view on nuclear power. The comparison of the two studies was believed to give as thorough an evaluation of reactor safety in Sweden as possible.

The objective of the MHB study was to give a risk assessment of Barsebäck 2 as compared to the reference plant of WASH-1400. Consequently, it has a more far-reaching purpose than the Studsvik Study. The analysis in the MHB study embraces:

1. Possible failure modes and sequences leading to a core meltdown accident.
2. Different release mechanisms of radioactivity.
3. Health and environmental consequences of radioactive emissions from accidents.

Execution of Work

The Studsvik study was started in November 1976. It was divided into two parts, one dealing with the installation and one dealing with the consequences of radioactive emissions from accidents. The project group prepared several part-reports within these two problem areas, which were used by Olov Norinder as a basis for the summary report. So far only his report has been translated into English. A special reference group was organized for reviewing and guidance of the project. It consisted of representatives from the Inspectorate, the Radiation Protection Institute, the Swedish utilities, the Swedish reactor vendor (ASEA-ATOM), and Studsvik.

The MHB study was conducted with Science Applications Inc. as subcontractor as regards the probabilities for different accident sequences and release categories. The results were combined with the consequence model provided by Jan Beyea of the Center for Environmental Studies at Princeton University to give an assessment of risk for the Barsebäck plant. Beyea's work was presented in a separate study performed for the Energy Commission.

Description of Barsebäck 2

The Barsebäck 2 nuclear power plant was chosen as the subject of the studies for two reasons:

1. It is situated relatively close to the large population centers of Copenhagen, Malmö, and Lund (fig. 1).
2. There are two other plants of a total of six in Sweden of basically the same design, namely Barsebäck 1 and Oskarshamn 2 (fig. 2).

Barsebäck 2, which was commissioned in 1977, has a boiling water reactor manufactured by ASEA-ATOM. Its net output is 580 MWe. Preliminary to the accident analysis some of the safety-related systems will be described briefly.

The reactor's reactivity control system contains 109 cruciform control rod units, which in normal operating maneuvers are moved by an electro-mechanical screw transmission system. In case of a scram, a hydraulic system rapidly inserts all control rods. There is also a boron injection system which serves as a redundant and diverse reactivity control system. The logic of the reactor protective system is such that two out of three sensors must be activated in order to cause a scram.

During normal operation auxiliary power to the station is provided from either the external 400 kV or 130 kV lines. Emergency power is provided from either two diesel generators or two gas turbines. There is also a back-up battery system. The power is fed into two busbars which are separated throughout the station.

Decay power cooling is normally initiated by conducting steam from the reactor via the steam lines to the condenser and main coolant system of the turbine. It is also possible to use the blow-off system to conduct steam into the pressure suppression pool, from where the decay power is removed by the reactor containment cooling system. At reactor pressures below 1.2 MPa the cooling system for shutdown reactor is used.

Emergency core cooling systems include the depressurization blow-off system mentioned above, two redundant low pressure (below 1.8 MPa) core spray systems, and two redundant containment spray systems. For certain cases of small pipe breaks the main and auxiliary feedwater systems can be used for providing the reactor with make-up water.

Outline of the Studies

The analysis in the Studsvik report proceeds from the conclusion of the Rasmussen study that a normal transient is the most probable initiating cause for an event sequence leading to a meltdown of the core. The part of the report dealing with the installation is consequently mainly dedicated to the analysis of event trees beginning with a transient, and LOCA accidents are treated only summarily. As a basis for the accident analysis part-reports with the following objectives were made:

1. Compilation of conceivable accident causes.
2. Identification of event sequences with the greatest importance for the accident probability.
3. Evaluation of the reliability of the shutdown function.
4. Evaluation of the reliability of the decay power cooling.
5. A limited analysis of the core meltdown process and containment behaviour.

The available resources did not permit the undertaking of a complete new analysis corresponding to WASH-1400. In most cases the data of the Rasmusssen study had to be adopted. Concerning the course of events when the core melts, it was judged that there is not enough experimental data for improving on the analysis in WASH-1400. So the same release categories were used without alterations.

The source terms, used in the calculations of radioactive releases from hypothetical accidents in the Barsebäck 2 plant, were the categories BWR 1, BWR 2, and BWR 3 combined with the calculated fission product inventory of the reactor. The diffusion of radioactivity was computed using weather statistics recorded at Risø not too far from Barsebäck. A special plume rise model was used. Values for the deposition velocity were chosen corresponding to the actual weather situation. Dose distributions were calculated for cases which were thought to be of particular interest. The densely populated areas around Barsebäck received special attention. This is contrary to WASH-1400 in which an average population distribution is used in the analysis. Extensive comparative calculations were made in order to bring about a better understanding of the differences of the models and the choices of parameters in the Studsvik study, WASH-1400, and the Danish report Risø-356.

In the MHB study the WASH-1400 approach was also used. However, transients and pipe ruptures were given the same attention when assessing the initiating causes of serious accidents. Event sequences leading to core meltdown accidents were then defined. Possible containment failure modes according to WASH-1400 were assessed, in contrast to the Studsvik study. After quantification of the event trees, the probabilities for release categories BWR 1 through 5, as defined by WASH-1400, were calculated.

The considerations in the MHB study of the consequences of radioactive releases were based on simulated accident scenarios. Weather conditions were selected according to a Pasquill stability class matching local weather. Plume rise height and deposition velocity were chosen randomly from a wide range of values. The influences of evacuation and medical treatment were included when early and late health effects were calculated for the population around Barsebäck. Consequences of land and sea contamination were given some attention, too.

Finally, an overall risk assessment was made expressing, in the same manner as WASH-1400, the probability of various health effects as a function of their number.

Results of the Studies

In the two reports the following basic assumption is made: catastrophic emissions of radioactivity from the reactor to the environment can only take place in conjunction with a meltdown of the fuel and subsequent damage to the containment.

The Studsvik project group made an inventory of possible causes of serious accident sequences at Barsebäck 2. No unique causes, not mentioned in WASH-1400, were discovered for the Swedish reactor. The following causes were considered in the work:

1. Pipe rupture in the nuclear power installation.
2. Rupture of the reactor vessel.
3. Probable reactor transients.
4. Improbable reactor transients.
4. Airplane crash.
6. Missile from turbine.

The frequency of pipe rupture was obtained from the Rasmussen report. In the case of a large-scale pipe rupture the value is 10^{-4} per reactor year. This leads to a probability for unsatisfactory core cooling of $2 \cdot 10^{-7}$ per reactor year. Correspondingly, the total probability resulting from pipe rupture was calculated as $1.2 \cdot 10^{-6}$ per reactor year.

Regarding pressure vessel rupture in Barsebäck 2, the values of WASH-1400 were used. There the frequency is set at 10^{-6} per reactor year. Assuming that one tenth of such events leads to a core meltdown, the resulting probability in Barsebäck 2 would be 10^{-7} per reactor year.

In the Barsebäck 2 plant approximately 10 probable transients were assumed to occur per year. The event tree for unsatisfactory core cooling is shown in figure 3. With transients as initiators the probability of a core meltdown is found to be $2.3 \cdot 10^{-5}$ per reactor year. More than half of this number results from failure of auxiliary electrical power. The estimate of the reliability of this system is conservative. Due to the limited resources available, however, a thorough enough investigation has not been made to justify a better value for the reliability of the auxiliary power.

In one of the part-reports of the Studsvik study the probability of an airplane crash was found to be 10^{-6} per year for the Barsebäck 2 plant. The ensuing probability of a core meltdown accident was then calculated as 10^{-8} per reactor year.

The probability of core meltdown accidents initiated by probable transients in Peach Bottom 2 was obtained by summation of the figures in the appropriate section of WASH-1400. The value thus arrived at is $2.9 \cdot 10^{-5}$ per reactor year. The corresponding value for an accident due to pipe rupture was found to be $1 \cdot 10^{-6}$. It is therefore concluded that, within the error margins of the presented numbers, the probability of a serious accident is approximately the same in the Swedish plant and the American plant.

Extensive calculations were made at Studsvik of doses resulting from radioactive emissions. All weather situations with the wind directed towards the population center considered were chosen from the statistical material available. Doses from the plume and from ground fallout as well as inhalation doses were then calculated. Two examples of the resulting values will be quoted: for Copenhagen, at a distance of 20 kilometers, the individual bone marrow dose indoors after 24 hours' exposure time would range from 0.3 to 45 rems for release category BWR 1 and from 0.2 to 25 rems for release category BWR 2. It should be noted that doses from the ground fallout was very dominant in the most severe cases. This was particularly true in connection with heavy precipitation and was most noticeable in the near zone.

As was mentioned earlier, assessment of health effects was considered to lie outside the scope of the Studsvik study. Some tentative results were obtained, however, using a standard model and taking into account the population distribution in the region. It was then found that in the most severe cases an exposure time of 24 hours would result in prompt fatalities in the near zone only.

In WASH-1400 the results are presented in a way which makes it difficult to evaluate to what extent they are affected by the modifications in the Studsvik study. A more direct comparison may be made between the Studsvik study and the Danish study Risø-356, in which, however, only the direction Barsebäck-Copenhagen was studied for weather categories according to Pasquill. The results of the two studies were found to be approximately in agreement.

In the MHB study much of the failure data and some of the fault tree analysis of WASH-1400 were either adapted or used directly to determine the probability of radiation release categories. The causes for serious accident sequences considered by MHB are essentially the same as those considered by the Studsvik group. External causes, such as airplane crashes, are discussed but do not enter into the final analysis.

The frequency of pipe rupture is evaluated by MHB taking into account the design differences between Barsebäck 2 and the Rasmussen reference plant as well as recent operating experience. The probability for large-scale pipe rupture is then given as $7.7 \cdot 10^{-4}$ per reactor year. The WASH-1400 figure is 10^{-4}. For small and very small LOCA's the figures are $2.5 \cdot 10^{-3}$ and $3.0 \cdot 10^{-3}$ respectively. The values according to WASH-1400 are $3 \cdot 10^{-4}$ and 10^{-3}.

The other main initiating cause considered by MHB is transients. After analysing Swedish data an average value of 20 transients per year was obtained.

The event trees used by MHB are somewhat different from those used by the Studsvik group but give equivalent information. They lead to probabilities for the release categories BWR 1 to 5. The total probability per year of a core meltdown is given as $1 \cdot 10^{-4}$ with an upper and lower bound of $8 \cdot 10^{-4}$ and $4 \cdot 10^{-5}$ respectively. Approximately one half of the value $1 \cdot 10^{-4}$ is due to accident sequences initiated by transients.

In their assessment of containment failure modes MHB have advocated a substantial shift of probabilities compared to the Rasmussen report. For overpressure failures the direct release mode was judged much more likely than the indirect one. Furthermore steam explosion failures emerge as more important. The resulting values compared to WASH-1400 are:

BWR	1	2	3	4	5
WASH-1400	$1 \cdot 10^{-6}$	$6 \cdot 10^{-6}$	$2 \cdot 10^{-5}$	$2 \cdot 10^{-6}$	$1 \cdot 10^{-4}$
MHB	$4 \cdot 10^{-6}$	$1 \cdot 10^{-4}$	$2 \cdot 10^{-5}$	$1 \cdot 10^{-7}$	$8 \cdot 10^{-4}$

The computations of the potential consequences of accidents at Barsebäck are presented by MHB in families of curves in diagrams similar to those of WASH-1400. Hence they are not easily summarized to give quantitative statements. In general, however, the MHB study presents much more severe consequences than WASH-1400. By way of example, given the deposition and plume rise approach mentioned earlier, assuming only minimal health treatment being available and wind direction being towards Copenhagen, then a 24 h exposure time would result in thousands of prompt fatalities on the 1 % probability level for BWR 1. On the other hand, if release category BWR 2 is postulated, the same assumptions would result in no prompt fatalities on the same probability level. Furthermore, according to MHB latent cancers are likely to be the most important consequences of catastrophic reactor accidents. The importance of effective evacuation in case of an accident is stressed in the MHB study. It is also pointed out the problems with land and sea contamination which would arise in the case of a serious accident.

The calculated probabilities for the various release categories have been combined with the consequence estimates to yield the risk assessments. According to MHB the Rasmussen report has underestimated the prompt fatality risk. The MHB study also shows a greater risk for long-term cancer deaths than does WASH-1400.

Comparison and Discussion of Results

The probability of a core meltdown as given by MHB is 4 times the value found by the Studsvik group. The latter value is approximately in agreement with that of WASH-1400, as has previously been pointed out. Part of the discrepancy is due to differences in methodology. When these are left out a factor of two remains with which MHB gives a higher probability. Event sequences with either transients or pipe rupture as initiating cause contribute approximately equally to the probability value.

The Studsvik report gives a value of $2.3 \cdot 10^{-5}$ per year for the probability of core meltdown accidents with transients as initiators. The MHB study gives approximately the same number although it assumes twice as many transients per year. This means a calculated value for the reliability which is higher than that obtained in the Studsvik study. It seems that the reliability of the shutdown system is overestimated in the MHB report. Furthermore, complete loss of AC power is not treated as conservatively as in the Studsvik study and a lower probability is obtained.

The difference between the values of the probability of a core meltdown accident in the two studies is then obviously due to the difference in assessing the contribution from pipe rupture. The Studsvik report assumes the WASH-1400 value. MHB, on the other hand, try to establish a value, which in the case of large LOCA's is 8 times as high. There has been much discussion of this raised value within the Inspectorate. Although the value as given by WASH-1400 is rather uncertain, we believe that the foundation for an argument to use a higher value for Barsebäck 2 than for Peach Bottom 2 is weak. This opinion of the Inspectorate is based on:

1. The recirculation system of Barsebäck 2 can not be considered more unfavourable than that of Peach Bottom 2.
2. Stainless steel in Swedish reactors has lower coal contents than that of American reactors.
3. The problem with thermal stresses in American reactors has not arisen in Swedish reactors.

The contribution from pipe rupture initiated sequences to the probability of a core meltdown is further raised in the MHB report by the assumption of a higher value of the probability of failure of the emergency core cooling function (ECF) than that given by WASH-1400. This number as used in the Studsvik study is $3 \cdot 10^{-4}$. MHB claim that design differences in Barsebäck 2 and Peach Bottom 2 warrant the higher value of 10^{-2}. Although the uncertainty of this number is great in the Rasmussen report, the attempt of MHB to raise it is not altogether convincing. A great effort has been put into the design of the emergency core cooling system of Barsebäck 2 to make it withstand the forces postulated in conjunction with a LOCA.

The values of the probabilities of the various release categories derived by WASH-1400 and by MHB are quite different. It should be noted how sensitive the number of health effects is to the choice of category. This is apparent from the calculations done by Beyea.

As has been pointed out before, the health effects in the worst case according to MHB greatly exceed those of the Studsvik report. In the case of prompt fatalities this is to a great extent due to the method of selecting the combination of parameter values for plume rise and deposition velocity. There is some controversy about Beyea's approach as to whether it always gives physically possible combinations. The higher values of latent fatality risk as quoted by MHB ensue partly from the assumptions of ineffective evacuation and minimal health treatment.

Conclusions

Much effort has been made to compare and combine the results of WASH-1400, the Studsvik report, and the MHB report. Some general conclusions have been arrived at:

1. Although there are differences in the system design of Barsebäck 2 and Peach Bottom 2, the safety levels of the two plants are approximately equal. The probability of core meltdown accidents as calculated by the Studsvik group is only lower by a factor of 2 than that of MHB, despite the conservative and critical approach of MHB. The reasons for the discrepancy are well known.

2. The methodology of WASH-1400 is a useful tool for evaluating the effectiveness of backfitting measures. However, a thorough understanding of the factual design which is being analysed is needed in the application of the figures obtained.

3. With certain combinations of extremely pessimistic assumptions cases of meltdown accidents with very low probability but with unacceptable consequences may be constructed. It must be emphasized, however, that a core meltdown accident in Barsebäck 2 would always be a very serious event. Vast areas of land and a great number of people would be affected. At present there exists an organization in Sweden to deal with such accidents. Whether the measures which are being prepared would be sufficient in case of an accident in a nuclear plant is being looked into. Work in this context is being done by the Radiation Protection Institute using, among other information, the results from the Studsvik study.

4. Techniques for calculating doses from radioactive releases in conjunction with core meltdown accidents in nuclear power plants have been developed. The significance of the various parameters being used in WASH-1400, the MHB study, and the Studsvik study has been clarified.

Acknowledgement

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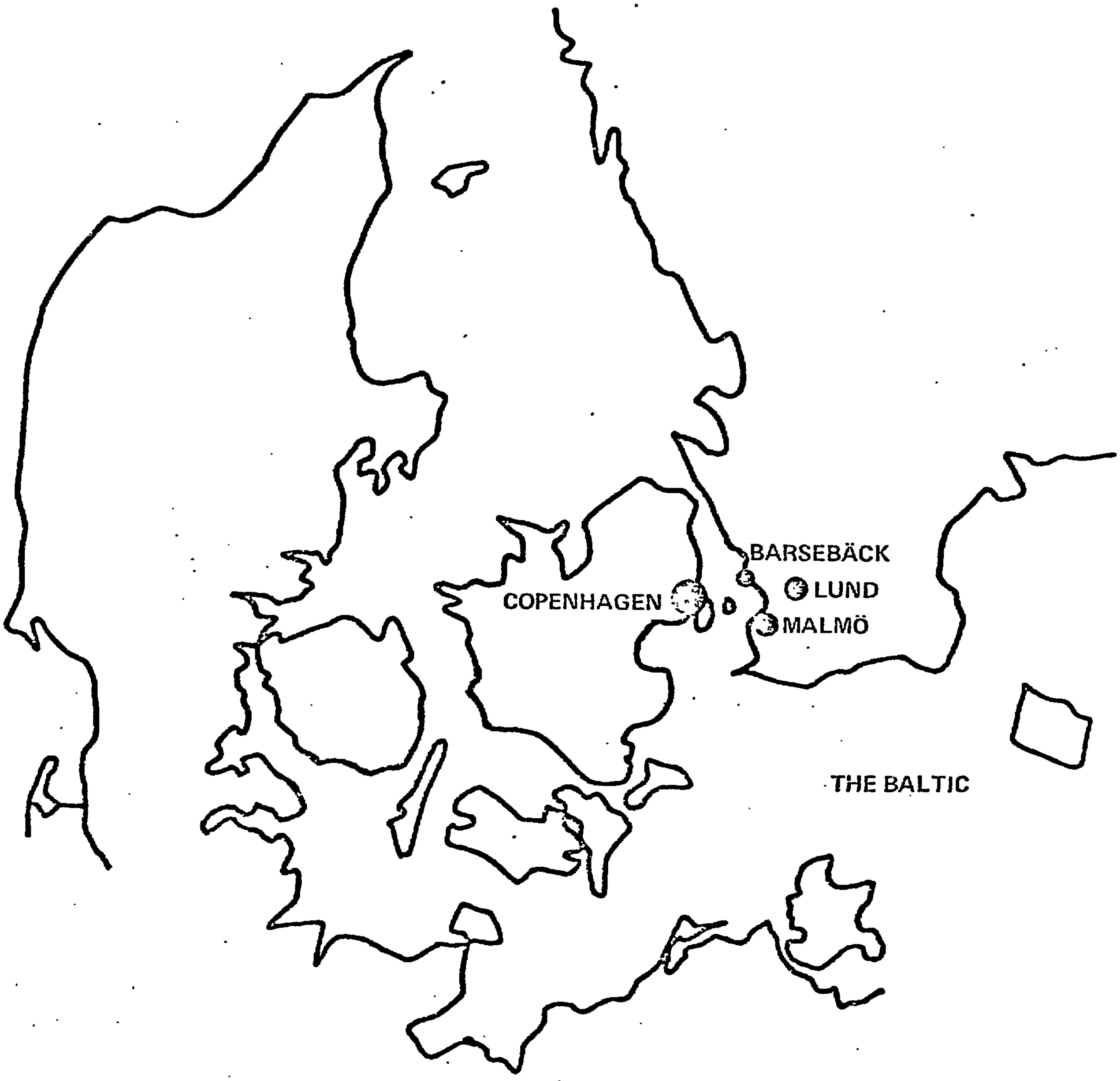
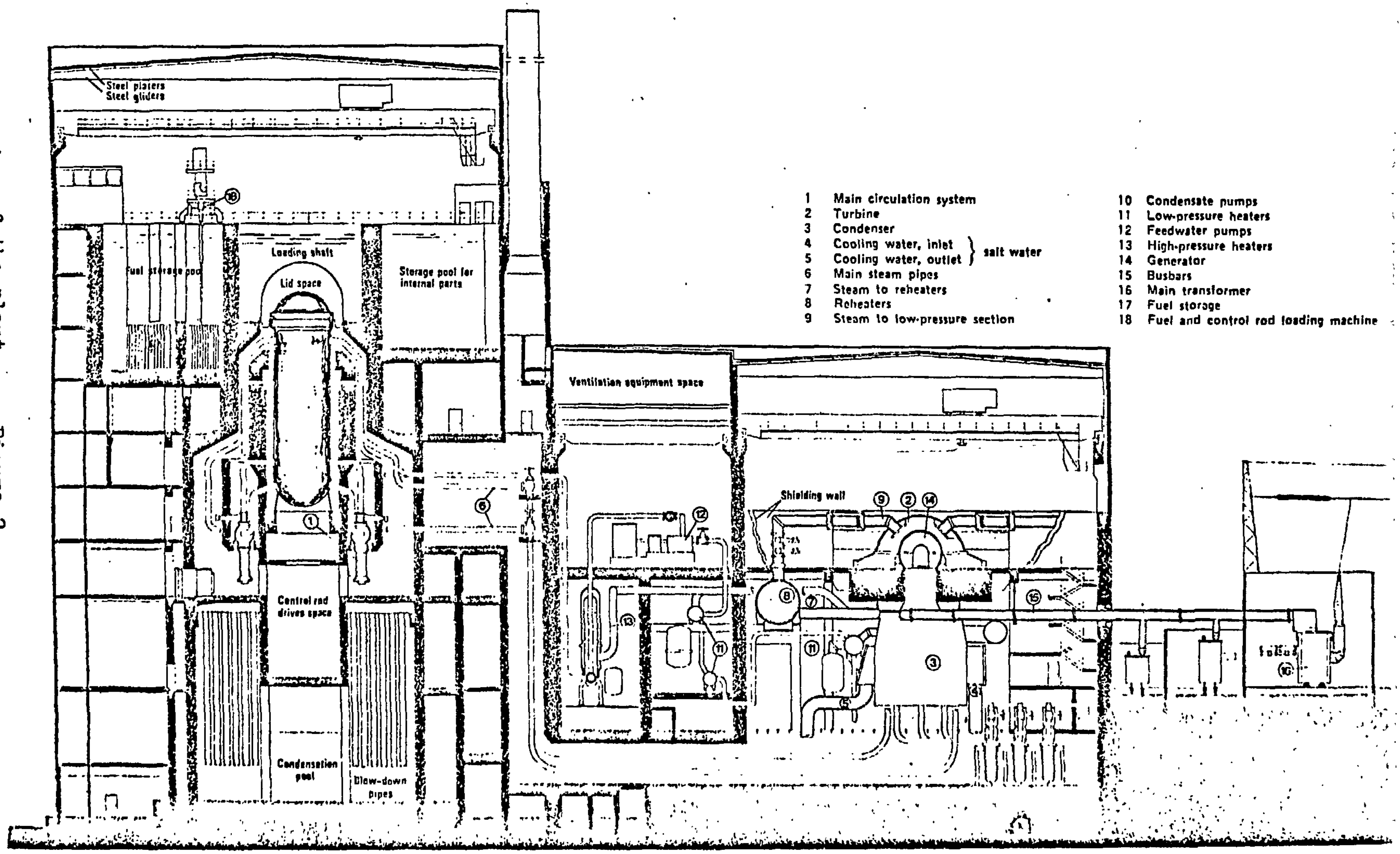
   P. Hedemann-Jensen et al. Risø Report No. 356 Oct. 1977.

Note: No. 2 can be obtained from the Swedish Nuclear Power Inspectorate. Nos. 3 and 4 can be obtained from Liber Förlag, S-162 89 Vällingby, Sweden.



Location of Barsebäck.

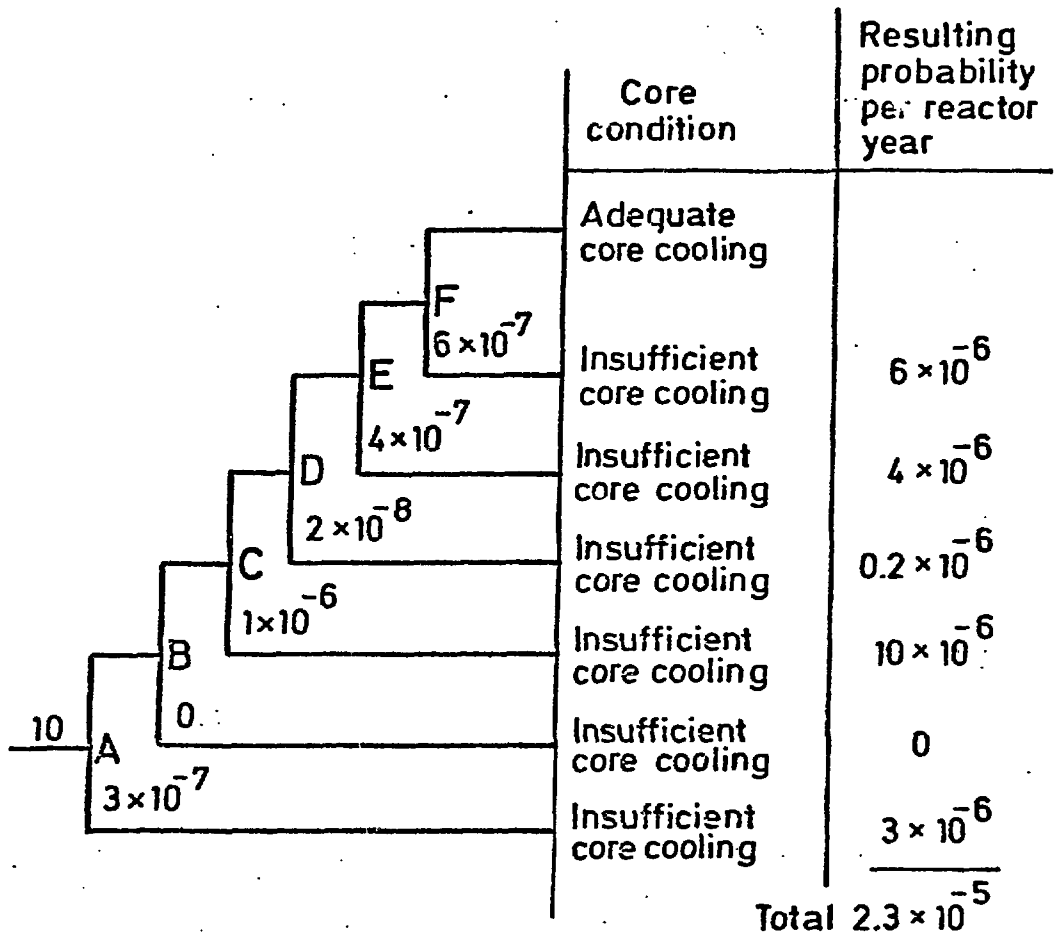
Figure 1.



Side-view of the plant. Figure 2.

Initiating event:

Anticipated transient
(10 per annum)



System function

- A Nuclear shut-down (scram)
- B Pressure relief and safety valves
- C Auxiliary power within 20 minutes
- D Make-up water supply
- E Long-run auxiliary power
- F Decay power given off to environment (sea)

Functional event tree for anticipated transients

Figure 3. [redacted]