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THE GERMAN RISK STUDY

Summary

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Preprint of the Summary of the German Risk Study

A first report on the new completed Phase A of the German Risk Study - a study carried out under contract to the Federal Minister of Research and Technology and investigating the incident-caused risk in nuclear power plants - was presented at the Federal Press Conference on August 14, 1979 in Bonn. Simultaneously, a photo-copied summary of the results was presented which, however, was already out-of-stock after a couple of days. Since printed copies of this summary will not be available from the Research Ministry before October 15, 1979, we were requested to produce a preprint in order to satisfy the most urgent needs until that date. The main volume of the German Risk Study, in which all the essential results are presented and explained in detail, will be available from the Research Ministry in the fall of 1979. The specialized volumes presenting all data required for repeating the work performed will be published by us by the end of 1979.

Cologne, August 1979

GESELLSCHAFT FÜR REAKTORSICHERHEIT (GRS) mbH
(Reactor Safety Company)

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1. INTRODUCTION

In nuclear engineering, safety considerations have always played an important role. Thus, nuclear power plants have to meet comprehensive safety requirements with regard to design, construction, operation and decommissioning. These requirements are a major element of the regulatory process under the German Atomic Energy Act and have been set forth in detail in the Safety Criteria for Nuclear Power Plants issued by the Federal Minister of the Interior, the Guidelines published by the Reactor Safety Commission and the relevant nuclear engineering codes.

On the basis of licensing applications, the responsible authorities will examine whether the necessary precautions have been taken in accordance with the state of the art in order to prevent damage which may result from the construction and operation of nuclear power plants. It will have to be demonstrated in this context that operational discharges of radioactive effluents will be kept as low as possible even if they are already below the acceptable limits. Accident-related radioactive releases have to be limited in such a way that neither personal injuries nor property damage will have to be expected in the environment as far as one can judge.

The safety review of nuclear power plants includes a comprehensive accident analysis. Safety-related requirements, in particular those for the design of the safety systems, are set up against the background of design basis accidents, i.e. the accidents involving the largest loads and thus most stringent requirements. The design of the safety systems is based on these requirements. The demonstration that these accidents will be coped with in terms of safety features includes at the same time the evidence that less serious accidents involving smaller loads than the design basis accidents will be coped with as well. As a result of the safety precautions which have been taken, accidents involving more serious consequences than the design basis accidents are considered to be so unlikely that they can be precluded as far as one can judge.

Experiences so far show that this safety concept has proven its worth. At a worldwide level there is now available the experience of some 1,500 reactor operation years over a period of about 25 years. During this time, neither deaths nor other health effects due to activity releases have occurred in the environment of nuclear power plants.

Outside the nuclear licensing process, there have been considerations to estimate the consequences of serious though extremely unlikely accidents beyond the design basis accidents. Well-known in this respect are US and British studies which deal with the consequences of arbitrarily assumed activity releases but do not include occurrence frequencies. These studies served to estimate maximum damage, clarify insurance aspects, derive safety requirements, etc. They are unsuited for an assessment of risks.

The US Reactor Safety Study, the "Rasmussen Report" (WASH-1400), was the first comprehensive study dealing with the determination of the accident-related risk of nuclear power plants. The Study was published in October 1975 after some three years of work. It was the first attempt to quantify the risks resulting from the operation of a great number of nuclear power plants in a single country. The Study enabled a systematic classification of all accidents in accordance with occurrence frequency and number of fatalities.

Although the US Reactor Safety Study had first been published as a draft version accompanied by a request for critical review and comment, and although the final version contains the modifications and amendments which were considered to be necessary, the Study has been reviewed and criticized frequently in the years to follow. Today's position of the Study can be characterized best by the results and recommendations of the Lewis Report.

The Lewis Report contains both criticism and appreciation of the US Reactor Safety Study. Critics point e.g. to the data

base which is still inadequate in several respects. Moreover, the uncertainties quoted for the results obtained are considered to be too small. On the other hand, however, there is explicit appreciation of the merits of the Rasmussen Report. In this context, the Lewis Report says, among other things:

- The Study was an essential step beyond earlier attempts to estimate the risks of nuclear power.

- The Study attained a far-reaching objectification of safety assessment, introduced a workable accident classification and presented a methodology for the quantitative determination of risks.

- The event tree/fault tree procedure, together with an adequate data base, has proved to be the best available tool for the quantification of the occurrence probability of accidents.

- The importance of late fatalities and property damage was recognized beside that of early fatalities.

2. PURPOSE AND TASKS

When the US Reactor Safety Study had been published, the question was raised in how far its results might be directly applicable to German conditions. Although the light-water reactor, as it is used in the United States and the Federal Republic of Germany, is predominantly the same type of reactor for the commercial generation of power, there are a number of points where the direct application of the American results to German conditions is impossible. Two aspects should be underlined in this context:

- The reference plants investigated in the US Study differ in numerous ways from the German plants as far as their engineered features are concerned. This applies primarily to

design and function of a number of important safety features.

- Site conditions as they exist in the Federal Republic of Germany also differ from those on which the US Study was based. For example, the mean population density in the Federal Republic of Germany is about ten times that of the United States. Even in the immediate vicinity of nuclear power plants, the mean population density in the Federal Republic of Germany is about three times that of the environment of American reactor sites.

To be able to judge the specific German situation, i.e. to evaluate the differences in engineered plant concepts and site conditions, separate investigations seemed to be necessary.

In the spring of 1976, the Federal Minister for Research and Technology awarded the contract for a separate German study. The contract was conceived in connection with and as part of the broad German Reactor Safety Research Program.

The Study is to be used to probe and utilize the possibilities of probability-oriented methods for safety evaluations. The existing methods of safety evaluation can thus be interpreted further and expanded into greater detail. In additon, the results of the Study are to be used to supply important hints and establish new priorities with regard to the further planning of research and development projects in the field of reactor safety. In a more restricted sense, the objectives of the Study can be summed up as follows:

- The collective risk involved in accidents at nuclear power plants is to be determined with due regard to German conditions.

- The results of the Study should allow comparisons with the US Reactor Safety Study in order to be able to evaluate differences in engineered plant features and site conditions.

The investigations of engineered plant features were carried out on the basis of a representative operating nuclear power plant with a 1,300 MW pressurized-water reactor. For risk evaluation purposes, all sites in the Federal Republic of Germany were considered where nuclear power plants with light-water reactors and an electric output of at least 600 MW were in operation, under construction and in the licensing process on July 1, 1977. This means that 19 sites and/or nuclear power plants with a total of 25 reactor units were considered.

In accordance with the far-reaching objectives, the German Risk Study was subdivided into two major phases (Phase A and Phase B). In Phase A, a great number of the basic assumptions and methods of the US Reactor Safety Study were taken over. Phase B, which is intended primarily for an intensification of individual problems, was to consider to a greater extent further developments in methods and recent results of reactor safety research.

The main contractor to prepare the German Risk Study is the "Gesellschaft für Reaktorsicherheit" whose executive managing director, Prof. Dr. A. Birkhofer, is responsible for the overall scientific management of the work carried out. The institutions contributing to the Study are listed in the Appendix.

3. METHODS OF INVESTIGATION

3.1 Accident Sequence

For risk analysis purposes, model concepts had to be developed of the processes taking place inside and outside the plant in case of an accident. The relevant considerations concentrated on such event sequences as involve greater activity releases from the plant and thus the possibility of fatalities in the environment.

At first, location and quantity of radioactive materials inside the plant were identified. Up to 95 % of the total activity in-

ventory are located in the reactor core and the reactor coolant loop. The remaining percentage is located mainly in the spent fuel elements in the spent fuel pool and to a lesser degree also in the reactor auxiliaries. Considering the existing safety features, no essential risk contribution as a result of a failure of these plant systems is to be anticipated. Therefore, the investigations concentrated on possible releases from the reactor core.

In the case of an intact plant, the radioactive materials produced in the reactor core are retained by several structures. Apart from the "inner structure" (crystal lattice of the fuel, fuel rod cladding tubes) which will practically retain the fission products at the place of production, there are the "outer structures" (reactor coolant loop, containment vessel). In case of a failure of the reactor coolant loop or of the containment, the activity releases will remain low as long as the fuel rod cladding tubes and the crystal lattice of the fuel can be kept more or less intact. Thus, major emphasis is placed on such events as will lead to a failure of inner structures. Later on, the possible consequences for the outer structures are investigated.

The overwhelming majority of radioactive materials can only be released if the fuel is overheated and in particular if the crystal lattice of the fuel disintegrates, i.e. if the fuel melts. However, even in the case of a complete fuel melting, various fractions would remain in the molten fuel, depending on the physical and chemical properties of the respective radioactive substances.

For the determination of risks, investigations thus concentrate on such events as will lead to a core meltdown. To simplify matters it is assumed in this context that accidents which will only lead to insufficient core cooling will always result in a complete reactor core meltdown. This procedure implies a tendency to overestimate the occurrence frequency of core meltdown accidents and thus supplies an upper risk estimate.

The melting fuel will also cause a failure of the core support structures. The molten fuel rods and the molten structural materials will drop into the lower semispherical part of the reactor pressure vessel. The residual heat release in the molten core will be sufficient to melt the bottom of the reactor pressure vessel and maybe even the concrete structures below.

By way of several different and partly simultaneous processes, the energy from the reactor core and from the molten core will get into the containment atmosphere where it will cause an increase in pressure and temperature.

Significant processes in this context are, among other things:

- the evaporation of the residual water in the reactor pressure vessel,
- the metal/water reaction between fuel rod cladding tubes and water/steam,
- the steam production resulting from the contact between molten core and sump water,
- the evaporation of the water released during the melting of the concrete.

As far as the impact on the environment is concerned, it is of importance to know whether, and if so when, the pressure inside the containment vessel will increase to such a level that a failure of the steel shell will occur. Although the activity content in the containment atmosphere will be reduced in the course of time as a result of condensation and deposition processes as well as radioactive decay, in particular of the short-lived radionuclides, the excess pressure in the containment vessel will at the same time cause the steam/gas mixture and radioactive materials to escape into the annulus and from the annulus into the environment if the containment has leaks.

The activity plume, which consists of the released mixture of vapors, gases and aerosols, will be carried away from the plant by the wind. In addition, the energy contained in the plume can produce a thermal lift. Although the activity plume is relatively confined at first, it will also spread at a right angle to

the wind direction as a result of turbulent dispersion. With an increasing distance an ever broader area will be covered by the plume. The concurrent dilution of the plume as well as the deposition and, possibly, washout of radioactive materials will reduce the activity concentration in the plume. The area covered by the plume will be subjected to radioactive contamination.

People staying in this area may be exposed to radiation through direct radiation and the inhalation of radioactive materials from the plume and from deposits as well as through the ingestion of radioactive materials in foodstuffs. The degree of radiation exposure, the number of persons exposed and thus the various kinds of possible consequences will depend not only on the anticipated activity concentrations but also on the feasibility of an implementation of emergency measures and their effectiveness.

3.2 Approach

The investigations carried out within the frame of the German Risk Study comprise the five different steps shown in Fig. 1.

I n i t i a t i n g E v e n t s – For the analysis of engineered plant features, the initiating events which may lead to an activity release to the environment are identified as to kind and frequency. Instead of performing an identification and analysis of all conceivable initiating events, it will be sufficient to treat a limited number of categories of initiating events which cover other initiating events as an envelope.

E v e n t T r e e a n d F a u l t T r e e A n a l y s i s – Starting out from an initiating event, different event trees will result depending on whether the required safety systems are available or not available. Event tree diagrams are set up in order to present an uncluttered view of the great number of possible event sequences. Fig. 2 is a schematic example of an event tree diagram starting out from a loss-of-coolant accident

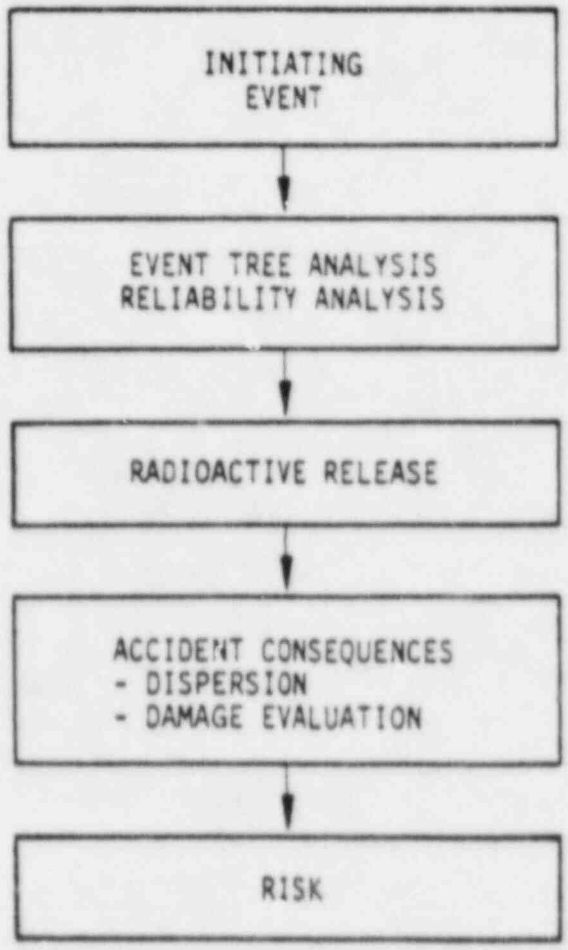


Fig. 1: Steps in the Risk Study

as the initiating event. Different event trees involving different consequences will result depending on whether the required safety functions such as power supply, emergency core cooling, etc. are available or considered not available. Fig. 2 shows such an event tree including the various activity releases from the containment vessel.

The event tree diagrams also incorporate probability data representing the frequencies of the event sequences which have been depicted. The occurrence frequencies of the initiating events and the non-availabilities of the existing safety features are needed for this purpose. The necessary reliability analyses are carried out with the aid of the fault tree analysis.

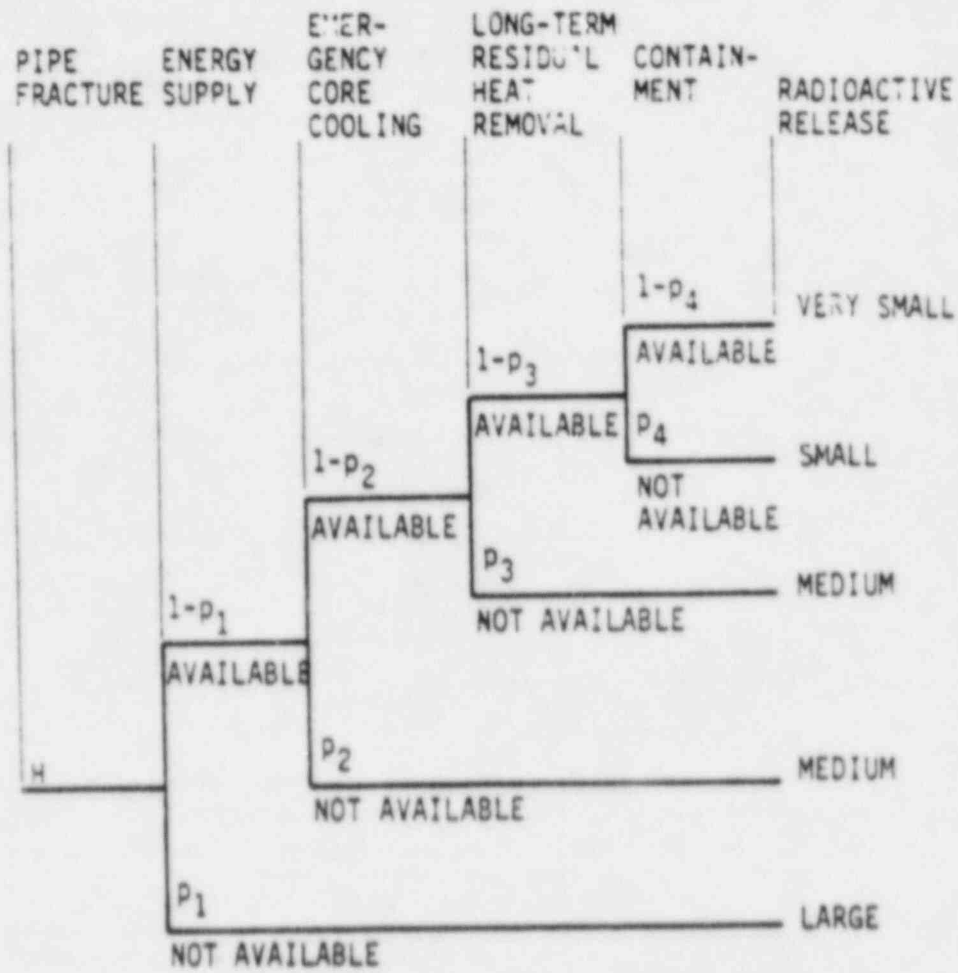


Fig. 2: Schematic event tree diagram
Loss of coolant accident

H: Occurrence frequency of initiating event per year
$p_{1,2...4}$: Non-Availability of safety features

Fig. 3 shows a schematic example of a fault tree for the undesirable event of a "Failure of Electric Power Supply". Starting out from this event all possible failure combinations of the system are followed up on the basis of the logic structure down to the level of elementary failures such as those of individual components. Only the first few steps of a fault tree analysis have been depicted. The electric power supply is considered to have failed if either the d.c. power supply for the supply of protective and control features or the a.c. power supply for the drive of the necessary safety-related components has failed. As soon as all possible failure combinations of a

system have been developed down to the level of elementary failures with the aid of the fault tree method, the failure probability of the system concerned can be calculated on the basis of the input data for the failure behavior of components and for other influencing variables (e.g. evaluation of a manual intervention).

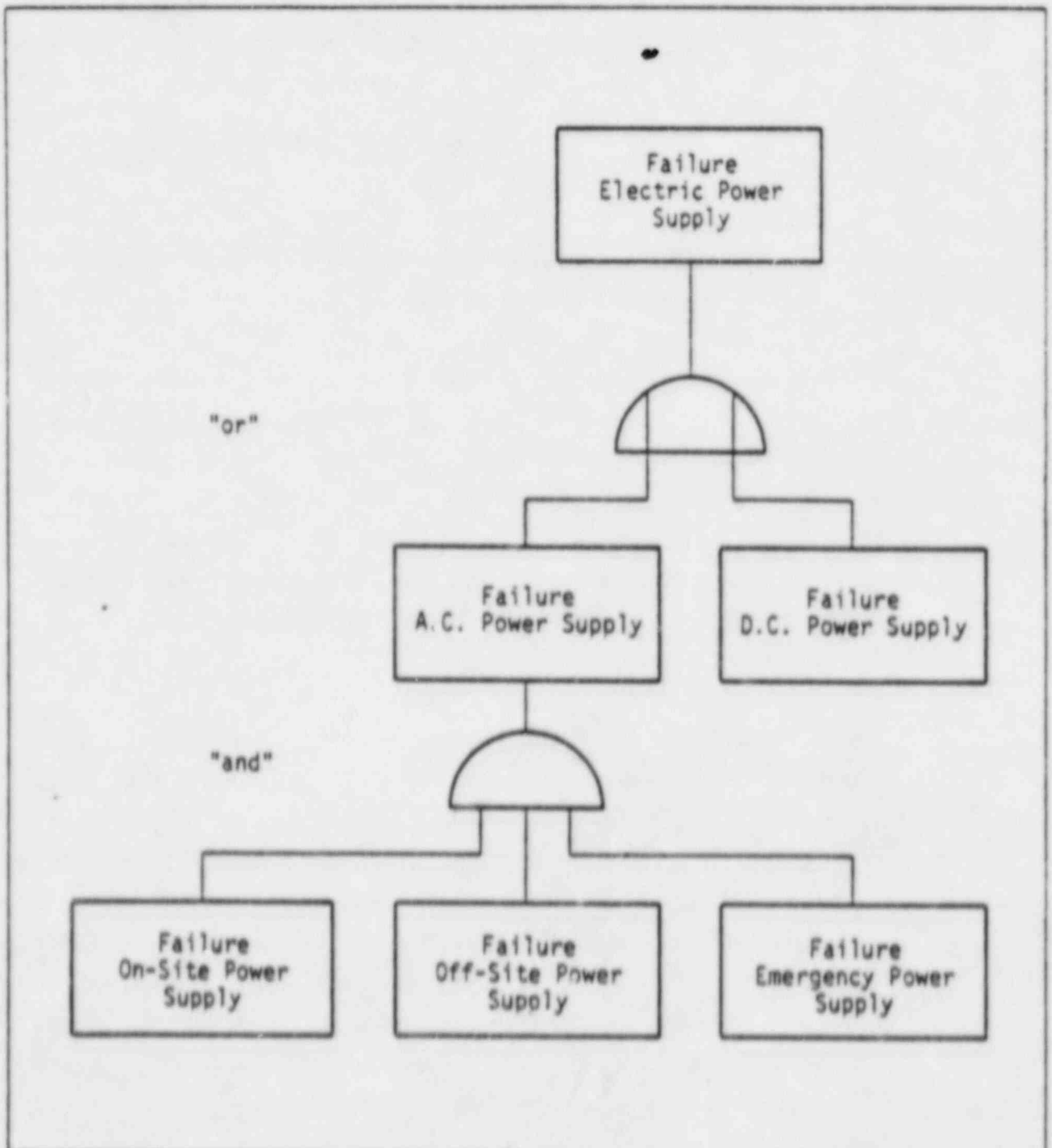


Fig. 3: Schematic fault tree
Loss of electric power supply

R a d i o a c t i v e R e l e a s e - For the determination of activity releases, the course of core meltdown accidents inside the plant is investigated, and kind and extent of the activity releases are calculated. This includes

- the processes involved in the meltdown of the reactor core and the behavior of the molten core,
- the behavior of the containment and its possible failure modes,
- the activity transport inside the containment and the release to the environment.

As several, basically different processes may lead to a containment failure, it is not only an investigation of the dynamic processes inside the containment but also a determination of the probabilities of the various failure modes that have to be done. The containment will fail if pipe or cable penetrations are not closed, if leaks turn up as a result of faulty materials or workmanship or if the stresses exceed the failure limits. In such a case, time and mode of the failure concerned are of importance. The result of an analysis of engineered plant features consists of mode (extent, location, time history, energy carried along) and frequency of activity releases to the environment. The releases resulting from accidents involving similar sequences are grouped into release categories.

A c c i d e n t C o n s e q u e n c e s - Accident consequences are also investigated in several steps. At first, the weather-dependent dispersion of the activity plume is calculated. This results in space-dependent and time-dependent activity concentrations in the environment of the plant. The resulting radiation exposures are calculated next, followed by an introduction of possible emergency measures to be taken as a function of these exposures. The doses reduced by these measures and the number of persons concerned are the basis for the determination of health effects which can be subdivided into early and late fatalities.

As far as the number of fatalities is concerned, several accidental factors are of importance which have a decisive influence on the dispersion of radioactive materials. Thus, the numbers of fatalities are associated with frequencies resulting from the probability of these accidental factors and the frequency of the release category in question.

R i s k – The number of fatalities and the associated frequency are the results required for risk statements. The results are summarized and evaluated in the following sections.

3.3 Individual Aspects

C o m p l e t e n e s s – At a nuclear power plant, numerous conceivable causes may lead to operational occurrences. Depending on the probability frequency, the possible event sequences may lead to an accident whose consequences will be coped with or to an accident which is so unlikely that no engineered plant features are required for it. This depends on the non-availability of the various safety features which is kept as low as possible though it cannot be reduced to zero. The number of possible event sequences is given by the resolving power of the description. Out of the great number of possible sequences, such sequences are selected as will include all other event sequences as an envelope.

To facilitate matters, the initiating events were grouped into categories in the German Risk Study. The selection of the more closely analyzed event sequences depends on the relevance of their anticipated risk contribution. While the nuclear licensing process disregards event trees involving a far-reaching or complete failure of safety systems, since they are extremely unlikely, it is these cases that are of particular interest in the risk analysis. They are the only cases which, via a core meltdown or serious damage to the core, can lead to an activity release which determines the risk. They are thus decisive for the accident risk which will remain in spite of all safety precautions.

The event tree analysis identified such sequences in a systematic way and evaluates them as to probability and consequences. In principle, it cannot be precluded in this context that possible accident causes or events influencing the accident sequence are overlooked. Furthermore, occurrence probabilities may be assumed wrongly. Although the systematic procedure makes such mistakes unlikely, a demonstration of absolute completeness is impossible. This is why great value is attached in the investigations to an optimal utilization of available analytical abilities, technological experience and familiarity with existing systems.

Data Base - A fundamental problem of the analysis is that reliability data are needed for all components. These data can be obtained from the statistical evaluation of operating experience or, if this is impossible in individual cases, can be derived from the knowledge of design requirements and application conditions. The less the experience that has been made with a certain component, the greater will be the uncertainty of the data. In many cases, data are used which have not been determined for the existing component but for similar designs and similar conditions.

An evaluation of the reliability data discussed in literature is not sufficient for the performance of analyses of engineered plant features, since often the applicability of these data cannot be sufficiently confirmed. Therefore, it is indispensable for the analysts to rely on the collection and recording of their own data. Such data are obtained to an ever increasing amount with the collection of reliability data at German power plants, including nuclear power plants, and with the systematic evaluation of operating experience at nuclear power plants.

Accident Simulation - Risk analyses will not only have to determine system reliabilities but will also have to simulate accident sequences by way of computer codes. On the one hand, this is necessary in order to establish the require-

ments for safety systems. On the other hand, it is necessary to simulate those accident sequences which result from the failure of safety systems. This accident simulation requires suitable computer codes describing the individual phases of the event sequence such as core meltdown, fission product transport inside the containment vessel, failure of the containment, dispersion of radioactive materials to the environment, radiation exposure, population concerned, emergency measures and health effects. Often the codes can only provide an approximate description of the actual processes. Thus, when preparing the codes an attempt is made to arrive at less favorable results in the calculations than are anticipated in reality. So far, it is hardly possible to quantify the faults which are thus introduced into the analysis and which will generally lead to an overestimation of the risk.

Common Mode Failures - If a common cause leads to the failure of more than one component, subsystem or system at the same time, this is called a common mode failure. Such failures can be particularly serious if they involve redundant systems, i.e. duplicated components, subsystems and systems to maintain one and the same function. There are several causes for these failures: faults in the design and manufacture of the relevant components, faults in their operation, external events, and the like. Common mode failures may be identified and considered in the light of operational experience or by a detailed system analysis. The frequency of their occurrence is estimated with the aid of mathematical models considering the interaction between the systems under review.

The German Risk Study considers functional failures of redundant systems within the frame of the fault tree analysis. Common mode failures of components can only be quantified if corresponding operational experience is available. If such experience is not available it will, as a rule, only be possible to carry out a qualitative evaluation. The analyses performed reveal that the effect of common mode failures on the probability that a certain accident will not be coped with is relatively low as compared with human errors.

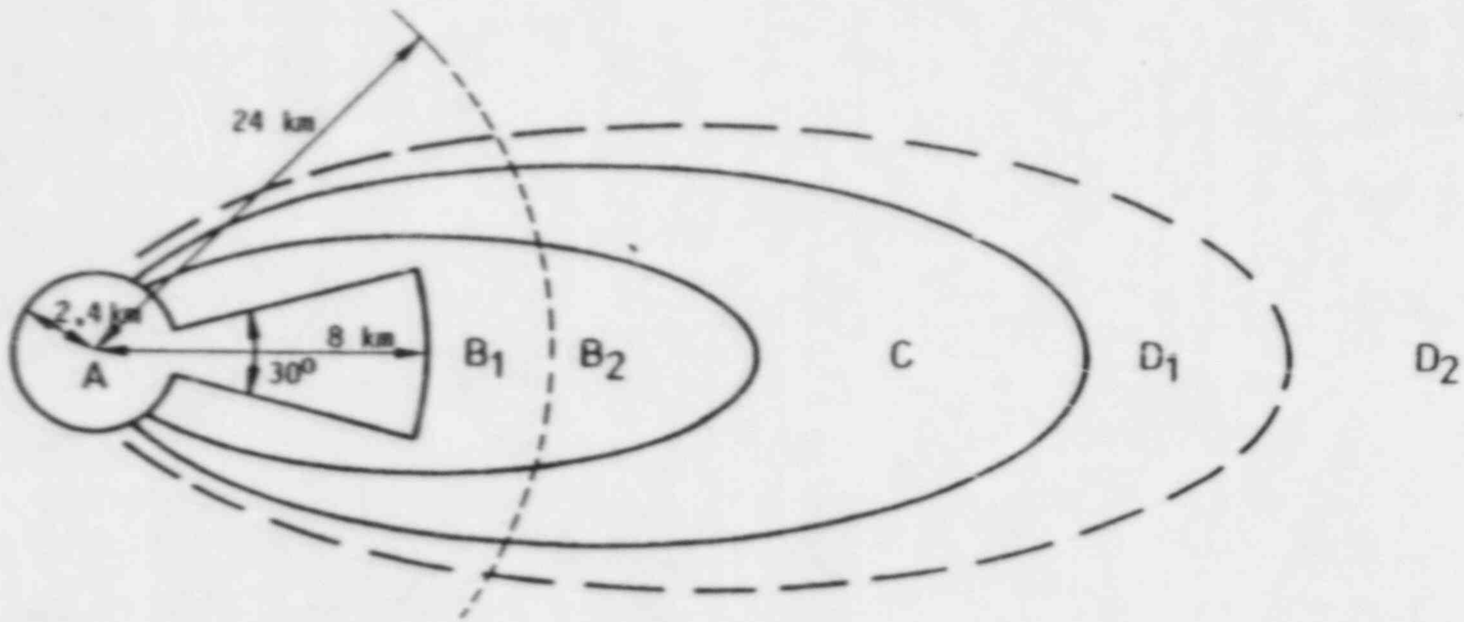
H u m a n   E r r o r – Human error may affect safety in any phase of the design, manufacture/construction and operation of nuclear power plants, during specified normal operation and under accident conditions. As long as comprehensive quality assurance, limit settings and the control of all important operational parameters have a preventive effect and reactor protection and safety features have a limiting effect, these possible effects can be restricted. Safety measures which are needed at short notice will be initiated automatically as far as possible. Scheduled manual intervention will have to consider human error although its quantitative evaluation is difficult. In general, the probability of maloperations is assumed to be the higher the less time there is available for an intervention and the more difficult the task to be performed is. Unscheduled interventions which may have a negative or a positive effect cannot yet be quantified. Therefore, they have not been taken into consideration in the investigations.

E m e r g e n c y   M e a s u r e s – On the basis of the basic recommendations for official emergency control in the environment of nuclear installations, the German Risk Study provides for the following protective actions and countermeasures:

- retreat into houses,
- evacuation,
- rapid relocation,
- relocation,
- decontamination and
- temporary restriction of the consumption of local agricultural products.

As in the US Reactor Safety Study, the intake of iodine tablets and the associated reduction of the thyroid dose are not taken into consideration.

The protective actions and countermeasures will depend on the doses determined without considering any shielding or decontamination. In this context, the following areas are defined for the environment of a nuclear power plant (Fig. 4):



<u>Fig. 4:</u> Regions for specific emergency measures

Region A is of constant size and identical for all release conditions.
The Regions B 1/2, C and D1 are defined by isodose limit values. In most cases, the dose values beyond A remain below the defined values of B 1/2. This means that the specifications of these regions and associated emergency measures do not apply. In many cases C and D1 do also not exist. No measures are provided in region D2.

- Area A: Circumference of 2.4 km and leeward 30° sector of 8 km - Irrespective of release category and environmental conditions, a retreat into houses after 2 hours and a stay in these houses of up to 8 hours followed by evacuation are postulated here.

- Areas B 1/2: Leeward areas defined by an isodose line (exceeding the dose to the bone marrow of 100 rad in 7 days) - Up to a distance of 24 km (B 1) the retreat into houses after 2 hours and a stay of at least 14 hours in these houses are postulated irrespective of release category and environmental conditions. No particular protective actions are postulated for those cases where the isodose line encompasses areas beyond 24 km (B 2). For areas B 1/2, rapid relocation will begin after 14 hours.

- Areas C: Leeward area defined by an isodose line (whole body dose of 250 rad in 30 years) - After 30 days a relocation action will be started which will cover a daily area of 5 km^2.

- Areas D 1/2: Leeward areas defined by an isodose line (whole body dose can be reduced to under 25 rad through decontamination or is under 25 rad in 30 years) - Normal human activities are postulated here.

The return of the relocated population is postulated at a time when the whole body dose is lower than 25 rad in 30 years. Supplementary dose-dependent measures include the restrictions in the ingestion of local agricultural products.

4. RESULTS OF INVESTIGATIONS

4.1 Analyses of Engineered Plant Features

A release of radioactive materials from the reactor core is only possible if the fuel rods are overheated in the course of an accident. Two major categories of accidents can be distin-

guished which may lead to a mismatch of heat production in and heat removal from the core:

- Loss-of-coolant accidents and
- transients.

With regard to this difference, which is made for historical reasons, any substantial deviation of operating parameters (e.g. power, pressure, temperature, flow rate) from the rated values that may lead to a mismatch of heat production in and heat removal from the reactor core is considered a transient unless such deviation is caused by leaks or breaks.

The investigation covers some 100 accident sequences which lead to activity releases. The frequency of core meltdown accidents is determined as $9 \cdot 10^{-5}$ or approximately 1 : 10,000 per year. Table 1 shows the results of the event tree analysis and reveals the contribution which is made to the frequency of core meltdown accidents by the various accidents which are not coped with. In connection with the most important initiating events, Fig. 5 shows the relative contributions which various failure modes in the safety features make to the frequency of core meltdowns.

By far the greatest contribution to a core meltdown is made by the small reactor coolant pipe break which is not coped with. One of the reasons for this contribution is the relatively great frequency to be postulated for small breaks. On the other hand, the ability to cope with small breaks is affected considerably by manual interventions which, in this case, involve a relatively great probability of a failure of the required systems. Thus the greatest contribution to the frequency of core meltdowns is caused by human error in the control of small breaks. The next most important contribution to the occurrence frequency of the core meltdown originates from a loss of the on-site and off-site power supply (emergency power case) whose occurrence frequency is relatively high. Such results are at the same time starting points for selected plant improvements. These consequences have already become obvious after the re-

Loss-of-Coolant Accident Transient	Frequency of initiating event h (Expectation value)	Conditional probability of the failure of required system function w (Expectation value)	Frequency of core meltdown accidents h . w (Expectation value)
Large reactor coolant pipe break	$2{,}7 \cdot 10^{-4}$	$1{,}7 \cdot 10^{-3}$	$5 \cdot 10^{-7}$
Medium reactor coolant pipe break	$8 \cdot 10^{-4}$	$2{,}3 \cdot 10^{-3}$	$2 \cdot 10^{-6}$
Small reactor coolant pipe break	$2{,}7 \cdot 10^{-3}$	$2{,}1 \cdot 10^{-2}$	$5{,}7 \cdot 10^{-5}$
Emergency power case	$1 \cdot 10^{-1}$	$1{,}3 \cdot 10^{-4}$	$1{,}3 \cdot 10^{-5}$
Failure of main feedwater supply	$8 \cdot 10^{-1}$	$4 \cdot 10^{-6}$	$3 \cdot 10^{-6}$
Emergency power case with small leak at pressurizer	$2{,}7 \cdot 10^{-4}$	$2{,}6 \cdot 10^{-2}$	$7 \cdot 10^{-6}$
Other transients with small leak at pressurizer	$1 \cdot 10^{-3}$	$2 \cdot 10^{-3}$	$2 \cdot 10^{-6}$
ATWS*)	$3 \cdot 10^{-5}$	$3 \cdot 10^{-2}$	$1 \cdot 10^{-6}$

*) Anticipated Transients Without Scram

Table 1: Summary of the results of the event tree analyses

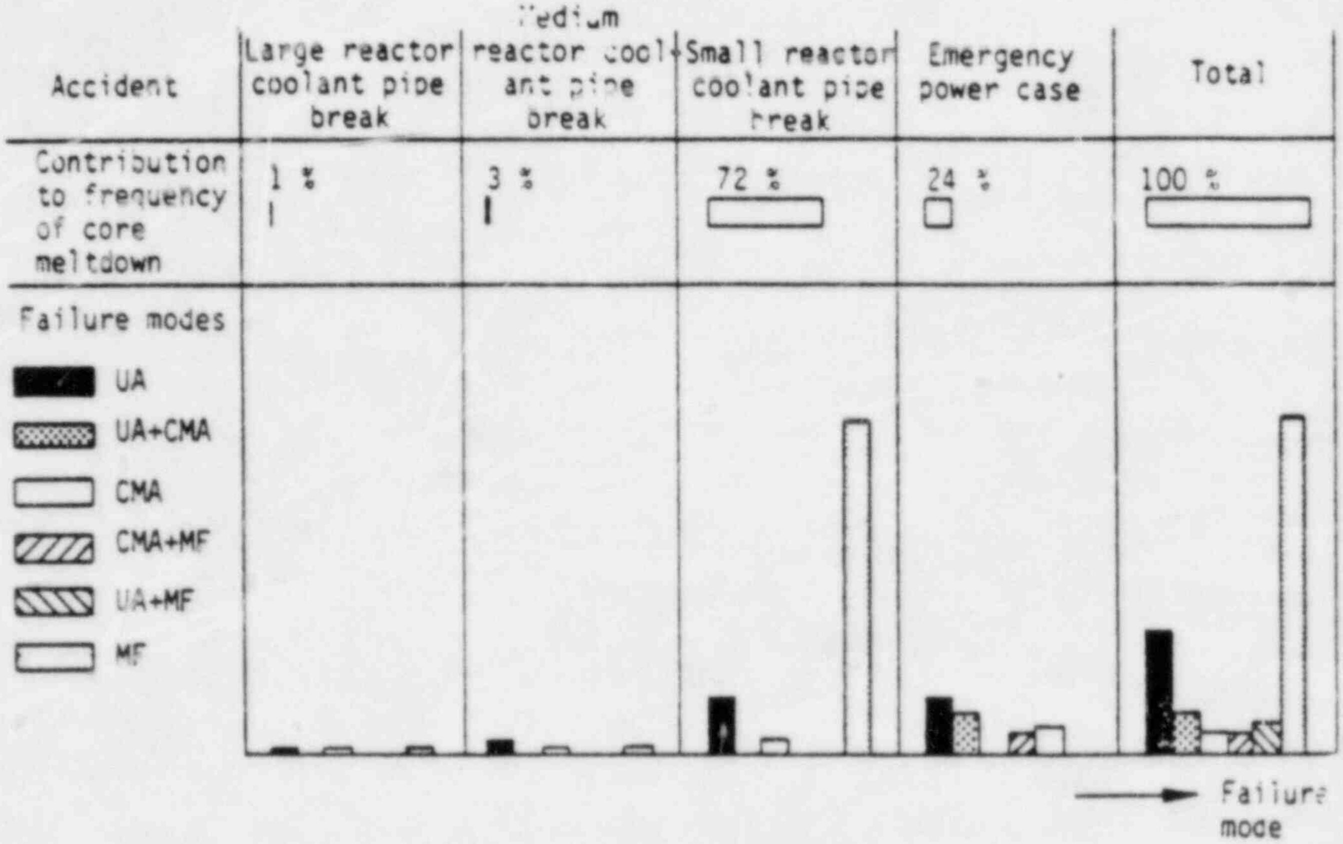


Fig. 5: Contribution of different failure modes to frequency of core meltdown

UA: Independent failure
CMA: Common-Mode-Failure
MF: Human error

porting of interim results in the course of the past three years. Major risk contributions identified in this context and originating from individual weak spots, in particular in the interfaces between control technology and process technology, have been reduced afterwards. Corresponding plant modifications could be taken into consideration until the first few months of 1978.

To determine the activity releases associated with a core meltdown accident, the possible failure modes of the containment have to be considered as well. The Study investigates the following failure modes:

- Containment leakage,
- Containment overpressure failure,

- Steam explosion in the reactor pressure vessel which leads to a destruction of the containment.

As their frequency depends on the event sequences concerned, the largest core meltdown frequency will not necessarily coincide with the greatest release frequency. All event sequences which have been investigated are grouped in one of a total of 8 release categories, depending on their release characteristics. Each release category is associated with a representative release which results from the most unfavorable values for the released fractions of the different radionuclide groups of the event sequences which have been grouped together. These release categories and the associated characteristic data are shown in Table 2.

The containment vessel has an important safety function for the nuclear power plant, as it will retain radioactive materials and thus enable the implementation of effective emergency measures in most of the event sequences. In those loss-of-coolant accidents which are coped with, the design limits of the containment vessel are not exceeded. The situation is different with regard to core meltdown accidents. If the molten core reaches the reactor sump, the reaction with the water in the sump, the generation of steam and the like will lead to a steady increase in pressure which will ultimate result in an overpressure failure of the containment. It is anticipated that such an overpressure failure does not occur before the end of one day. An earlier failure is only conceivable as a result of a steam explosion or in the case of leakage from the containment. In 93 % of all core meltdown accidents the release of activity is so restricted that early fatalities cannot be caused. As far as the remaining 7 % are concerned, environmental factors (weather conditions, population distribution) will lead to a further reduction of the frequency of fatalities so that no acute fatalities have to be anticipated in more than 99 % of all core meltdown accidents.

Release category No.	Description	Time of release h	Duration of release h	Release height m	Energy release 10^6 kJ/s	Frequency of release 1/a	Fraction of core inventory released: Xe-Kr	J_{org}	J_2-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
1	Core meltdown followed by steam explosion	1	1	30	540	$2\cdot10^{-6}$	1.0	$7.0\cdot10^{-3}$	$7.9\cdot10^{-1}$	$5.0\cdot10^{-1}$	$3.5\cdot10^{-1}$	$6.7\cdot10^{-2}$	$3.8\cdot10^{-1}$	$2.6\cdot10^{-3}$
2	Core meltdown, large leak in containment (Ø 300 mm)	1	3	10	15	$6\cdot10^{-7}$	1.0	$7.0\cdot10^{-3}$	$4.0\cdot10^{-1}$	$2.9\cdot10^{-1}$	$1.9\cdot10^{-1}$	$3.2\cdot10^{-2}$	$1.7\cdot10^{-2}$	$2.6\cdot10^{-3}$
3	Core meltdown, medium leak in containment (Ø 80 mm)	2	3	10	1	$6\cdot10^{-7}$	1.0	$7.0\cdot10^{-3}$	$6.3\cdot10^{-2}$	$4.4\cdot10^{-2}$	$4.0\cdot10^{-2}$	$4.9\cdot10^{-3}$	$3.3\cdot10^{-3}$	$5.2\cdot10^{-4}$
4	Core meltdown, small leak in containment (Ø 25 mm)	2	3	10	–	$3\cdot10^{-6}$	1.0	$7.0\cdot10^{-3}$	$1.5\cdot10^{-2}$	$5.1\cdot10^{-3}$	$5.0\cdot10^{-3}$	$5.7\cdot10^{-4}$	$4.0\cdot10^{-4}$	$6.5\cdot10^{-5}$
5 *)	Core meltdown, overpressure failure, failed filter system	0 1 25	1 1 1	10 10 10	– – 200	$2\cdot10^{-5}$	$2.0\cdot10^{-5}$ $2.3\cdot10^{-2}$ $9.8\cdot10^{-1}$	$1.8\cdot10^{-7}$ $1.6\cdot10^{-4}$ $6.8\cdot10^{-3}$	$1.8\cdot10^{-5}$ $9.6\cdot10^{-4}$ $9.6\cdot10^{-3}$	$4.7\cdot10^{-5}$ $6.7\cdot10^{-4}$ $4.5\cdot10^{-4}$	$3.6\cdot10^{-7}$ $6.7\cdot10^{-4}$ $7.7\cdot10^{-4}$	$5.5\cdot10^{-9}$ $8.0\cdot10^{-5}$ $4.7\cdot10^{-5}$	$5.5\cdot10^{-5}$ $5.3\cdot10^{-5}$	$8.8\cdot10^{-6}$ $9.5\cdot10^{-6}$
6 *)	Core meltdown, overpressure failure	0 1 25	1 1 1	100 100 10	– – 200	$7\cdot10^{-5}$	$2.0\cdot10^{-5}$ $2.3\cdot10^{-2}$ $9.8\cdot10^{-1}$	$1.8\cdot10^{-9}$ $1.6\cdot10^{-6}$ $6.8\cdot10^{-3}$	$1.8\cdot10^{-8}$ $9.6\cdot10^{-7}$ $9.6\cdot10^{-3}$	$4.7\cdot10^{-8}$ $6.7\cdot10^{-7}$ $4.5\cdot10^{-4}$	$3.6\cdot10^{-10}$ $6.7\cdot10^{-7}$ $7.7\cdot10^{-4}$	$5.5\cdot10^{-17}$ $8.0\cdot10^{-8}$ $4.7\cdot10^{-5}$	$5.5\cdot10^{-8}$ $5.3\cdot10^{-5}$	$8.8\cdot10^{-9}$ $9.5\cdot10^{-6}$
7	Design basis loss-of-coolant accident, large leak in the containment	0	1	10	9	$1\cdot10^{-4}$	$1.7\cdot10^{-2}$	$3.7\cdot10^{-5}$	$5.3\cdot10^{-3}$	$1.3\cdot10^{-2}$	$2.5\cdot10^{-5}$	$2.5\cdot10^{-7}$	0	0
8	Design basis loss-of-coolant accident	0	6	100	-	$1\cdot10^{-3}$	$4.6\cdot10^{-4}$	$1.0\cdot10^{-8}$	$1.2\cdot10^{-8}$	$2.1\cdot10^{-8}$	$4.3\cdot10^{-11}$	$4.1\cdot10^{-11}$	0	0

*) The released fractions of activity are specified for three intervals of time, because the release extends over a longer period of time.

<u>Table 2:</u> Categories of radioactive release

4.2 Determination of Accident Consequences

First, potential radiation doses are calculated on the basis of the calculated activity concentrations in the air and the associated soil contaminations. The various protective actions and countermeasures will be taken in accordance with these doses. Then the doses to be expected are determined with due regard to these dose-reducing measures.

The collective fatalities are determined in consideration of the population distributions at the 19 sites in question. The risk contributions of multi-unit plants are introduced in accordance with the number of units. The site-specific population data in accordance with sector and distance are taken as a basis for distances up to 80 km. For distances between 80 and 540 km, i.e. an area which covers mainly Central Europe, a characteristic population density of 250 inhabitants per km^2 is employed, since in this area local or regional differences will no longer influence the results of the investigations. Beyond this circumference and up to a distance of 2,500 km, a mean population density of 25 inhabitants per km^2 is postulated which is due to numerous sparsely populated areas and expanses of water. These distances were chosen by analogy with the US Reactor Safety Study.

The dose-effect correlation for early fatalities as used in the German Risk Study takes into consideration that the population includes groups having an increased radiation sensitivity which may be due e.g. to chronic infections, gastrointestinal diseases, wounds, burns, operations and pregnancies. For these groups, whose share of the total population may be 10 %, a mean lethal radiation dose of 340 rad was postulated. On the other hand, the medical profession has made such progress in the last few years, e.g. in the control of the side-effects of potent cytostatics which correspond to the effects of an acute radiation disease, that most people can even be saved after radiation exposures involving between 200 and 500 rad. The dose-effect correlation considering these two aspects begins at a dose

threshold of 100 rad, shows a 50 % mortality at 510 rad and reaches a 99 % mortality at 770 rad (Fig. 6).

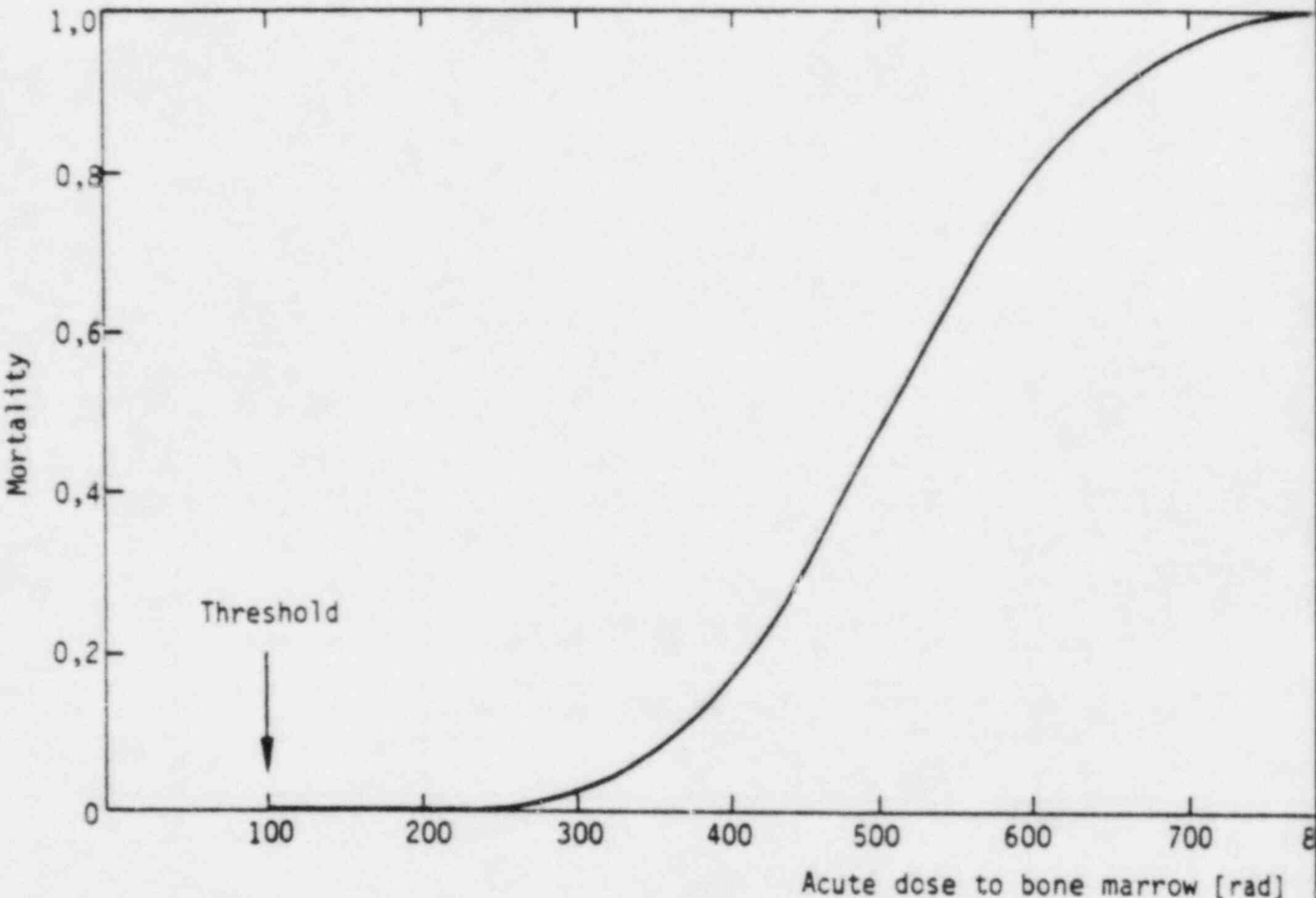


Fig. 6: Dose Effect Correlation for early fatalities
Mortality due to acute dose to bone marrow

Fig. 7 deals with early fatalities and shows the correlation between the number of fatalities and the occurrence frequency for 25 plants. The curve plotted in Fig. 7 is already a summarized representation of calculated results. The curve constitutes the cumulative frequency distribution[1]) which is an answer to the question as to whether the annual frequency of a given number of fatalities is attained or exceeded. The curve is obtained if, starting out from the maximum number of fatal-

[1]) *To facilitate easy understanding, the simplified term "frequency" will be used in the following instead of the mathematically correct "cumulative frequency distribution".*

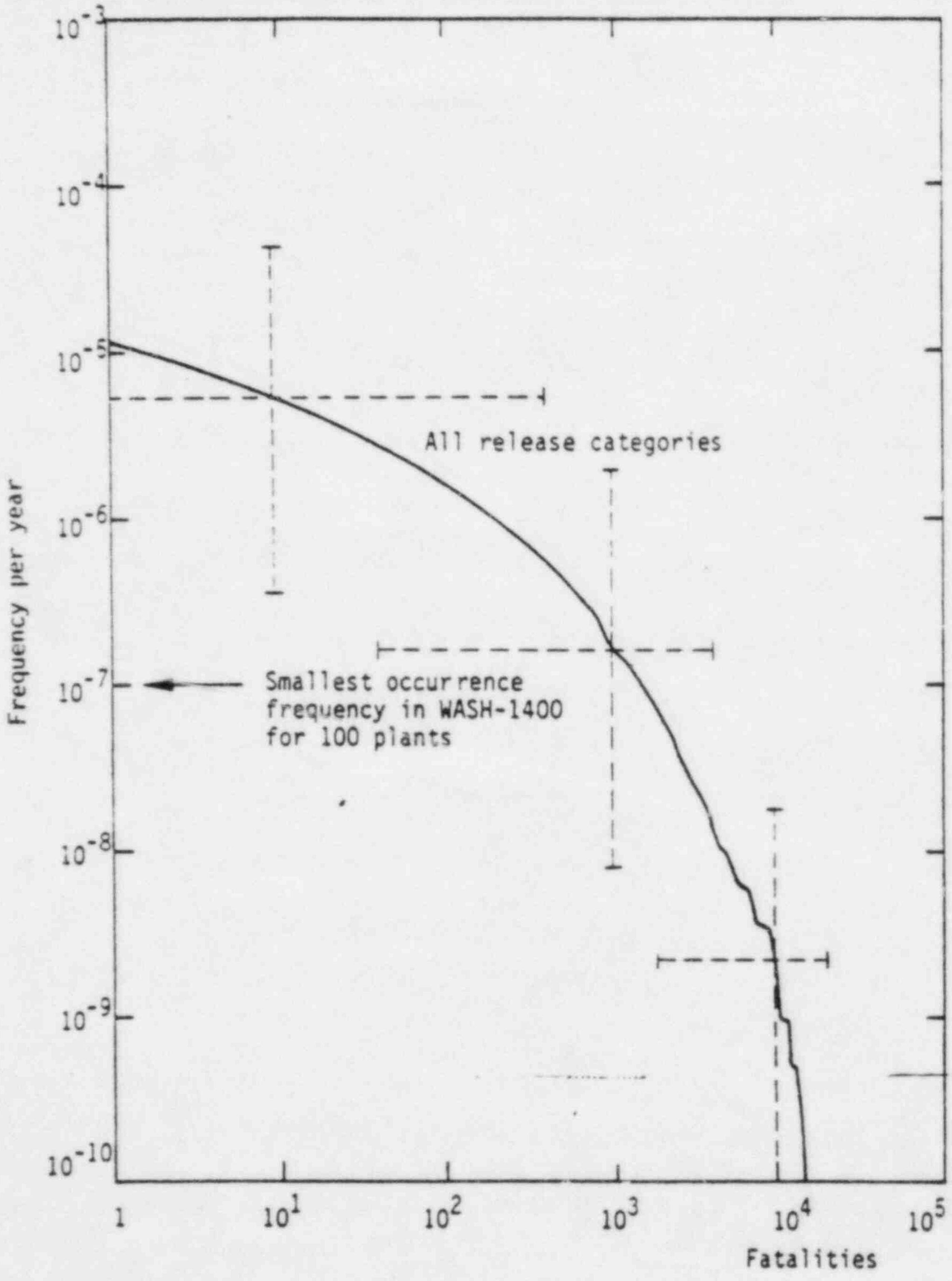


Fig. 7: Cumulative frequency distribution of early fatalities per year corresponding to 25 plants

The dashed bars indicate 90 % confidence limits.

ities that can occur, all occurrence frequencies are summed up which are associated with numbers of fatalities down to the given number of fatalities. Some points of the curve have been represented again in Table 3. All in all, the results obtained show that accident sequences which involve great numbers of fatalities are extremely rare. Thus accident sequences leading to 1,000 acute deaths or more are already associated with occurrence frequencies of under 1 : 1,000,000 per year.

Occurrence Frequency per year	Early Fatalities
1/ 100 000	2
1/ 1 000 000	200
1/ 10 000 000	1 400
1/ 100 000 000	4 400
1/1 000 000 000	11 000

Table 3: Occurrence frequency of early fatalities corresponding to 25 plants

Fig. 7 contains all results obtained including the maximum number of fatalities which could be determined in the calculations. The numeric value is 14,500 acute deaths (occurrence frequency obtained - 1 : 2,000,000,000 per year). In the numerous accident sequences which have been investigated, the maximum number of fatalities occurs in a sequence which is characterized by unfavorable conditions for release, weather conditions and population distribution. For smaller frequencies, no greater numbers of fatalities will be calculated either.

In general, it may be said that large numbers of early fatalities will only be determined if large activity releases coincide with a large number of persons in the sector concerned and if a high contamination of the soil is caused by rain in the vicinity. Acute deaths are obtained up to a maximum distance of approx. 20 km.

Based on the recommendations of the International Commission on Radiological Protection (1977), the German Risk Study uses for the determination of late fatalities a purely proportional dose-risk correlation which runs through the origin of the coordinates and corresponds to a risk factor of approx. 10^{-4}/rem (Fig. 8). As in this case the observations made with high doses

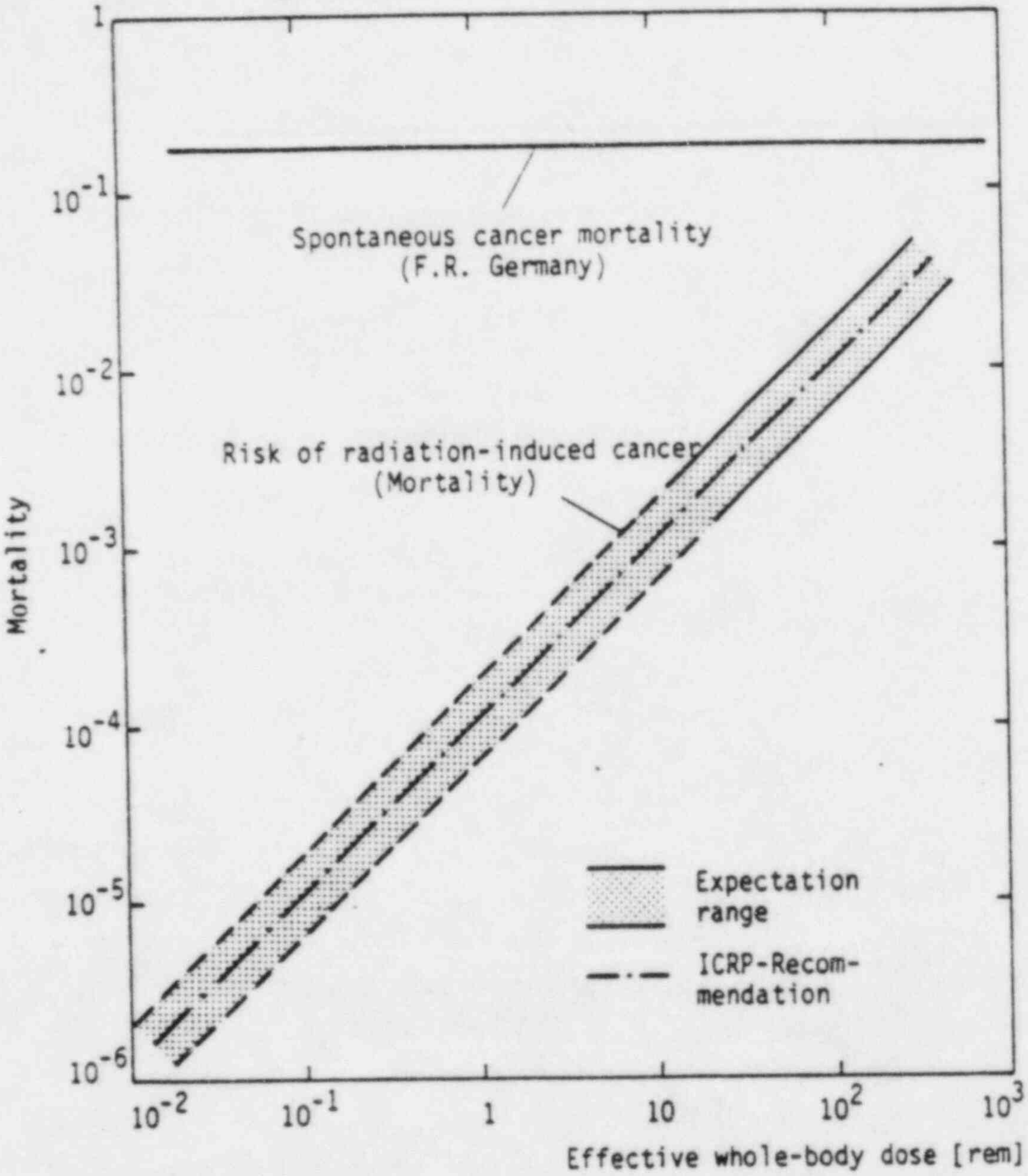


Fig. 8: Dose Risk Correlation for late fatalities

Mortality due to effective whole-body dose

For comparison the actual spontaneous cancer mortality in the German population is indicated.

were extrapolated to the risk associated with low doses, the real risk should have been overestimated rather than underestimated.

Fig. 9 shows the cumulative frequency distribution for late fatalities on the basis of 25 plants. Here, too, the course of the curve can be illustrated further by means of the following numeric examples (Table 4).

Occurrence Frequency per year	Late Fatalities
1/ 1 000	2 700
1/ 10 000	3 900
1/ 100 000	54 000
1/ 1 000 000	65 000
1/ 10 000 000	72 000
1/ 100 000 000	83 000
1/1 000 000 000	94 000

Table 4: Occurrence frequency for late effects corresponding to 25 plants

The explanations presented in connection with early fatalities will apply analogously to the maximum number of fatalities which has been calculated (approx. 104,000 deaths).

In this context, large numeric values for late fatalities result, among other things, from the consideration of a large number of persons subjected to small and very small radiation exposures. With the exception of the most serious accidents which are characterized by an early containment failure and large activity releases, the overwhelming majority of all calculated late fatalities result from an accident-related radiation exposure below 5 rem. This dose is more or less equal to the radiation exposure caused by natural background radiation in the course of 30 years. The late fatalities obtained would be spread over a period of several decades. Moreover, about one-half to

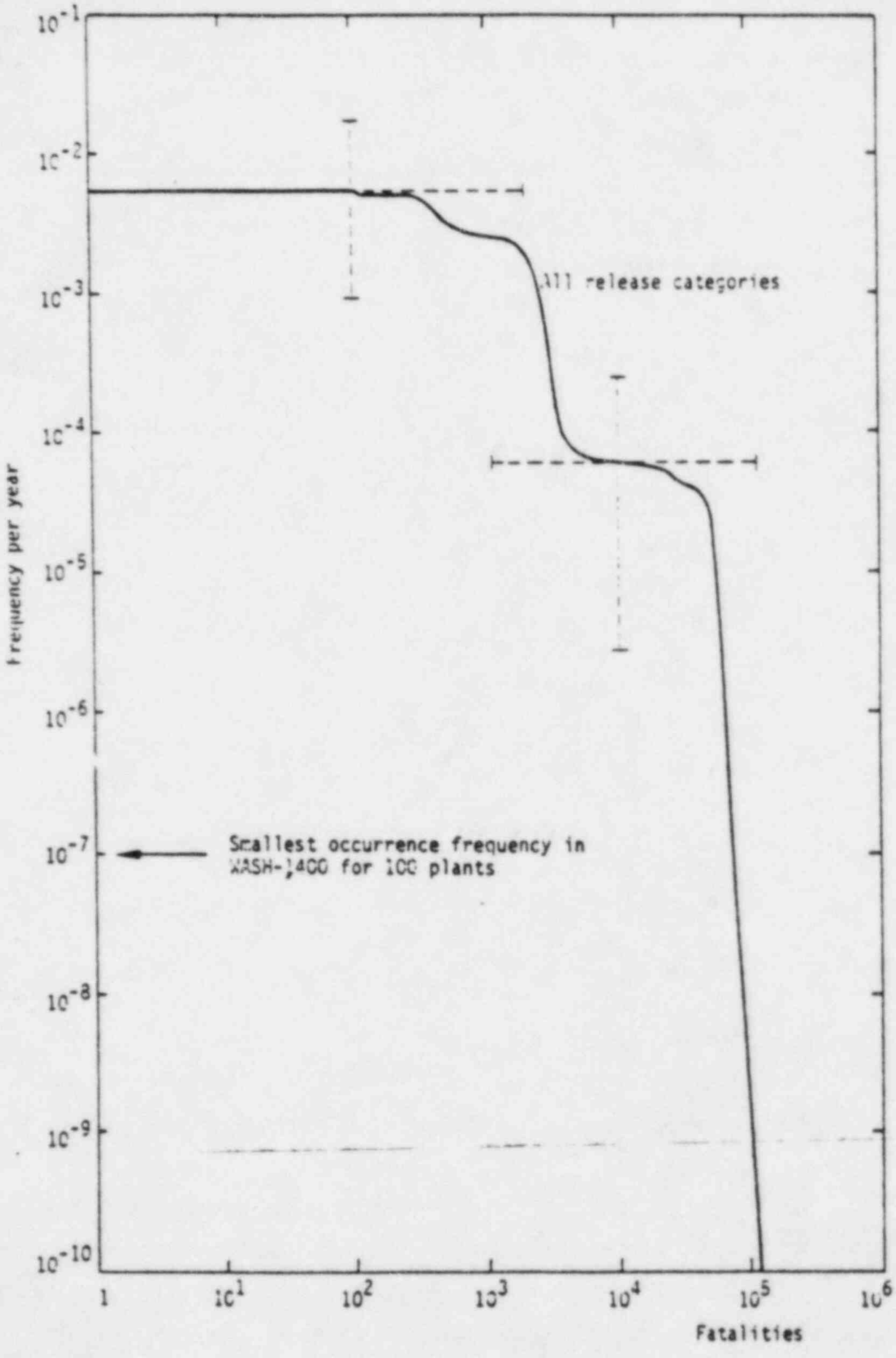


Fig. 9: Cumulative frequency distribution of late fatalities per year corresponding to 25 plants

Dashed lines indicate 90 % confidence limits.

them would be related to areas outside the Federal Republic of Germany. Accordingly, it can be concluded that the German population would be subjected to similar risks from nuclear power plants in foreign countries.

The computer code employed will supply large numbers of late fatalities whenever large activity releases are associated, as a result of special weather conditions in relatively large areas, with soil contaminations which are so small that the basic reference values for protective actions and countermeasures are not attained.

As it has not yet been possible to prove genetic radiation effects in population groups exposed to radiation, one has to use animal experiments for an estimate of this risk. The spectrum of possible health effects may range from minor deformations to serious illnesses. A categorization, which is necessary for comparison purposes, will not yet be possible with the required degree of reliability, not even if it is restricted to clinical relevance. Thus, only the genetically significant collective dose was quoted which can be interpreted in accordance with the current progress in the state of the art.

The cumulative frequency distribution for the genetically significant collective doses is represented by a curve which is basically similar to that of late fatalities. Thus, the relationship between occurrence frequency and collective doses is similar. With respect to the calculated maximum dose (approx. 420,000,000 man-rem), the explanations given in connection with early fatalities will apply analogously. Similarly, great numeric values will have to be anticipated here as well if great activity releases are associated, as a result of special weather conditions in relatively large areas, with soil contaminations which are so small that the basic reference values for protective actions and countermeasures are not attained.

The areas affected by the countermeasures (evacuation, rapid relocation and relocation) postulated in the German Risk Study

for the reduction of the number of fatalities are also investigated with respect to area size and number of persons (Fig. 4).

The conditions established in accordance with the basic recommendations for official emergency control provide that the immediate vicinity of the plant and the leeward sector (Area A) be evacuated in all types of accidents. This area has a given size of 33 km^2 and a maximum number of 66,000 and a mean number of 6,900 persons.

Outside Area A, radiation doses which would be so high as to suggest a rapid relocation to reduce the number of early fatalities, will only be encountered in 3 release categories and only in approx. 1 % of all core meltdown accidents. Thus, Areas B 1/2 may have a maximum size of 379 and a mean size of 14 km^2; the maximum number of persons may be approx. 1,000,000 and the mean number approx. 4,000.

In nearly all release categories, Areas C can be identified which cannot be decontaminated sufficiently so that a temporary relocation of the population is postulated in the calculational model in order to reduce late fatalities. The reference value that is set forth is a radiation dose that is higher than the mean natural background radiation received during the span of a life (12.5 rem). The corresponding areas have a maximum size of approx. 5,700 km^2 and a mean size of 11 km^2; the maximum number of persons concerned may be approx. 2,900,000 and the mean number approx. 2,900. Large areas and large numbers of persons are obtained almost exclusively for weather conditions which are characterized by rainfall. A dense population is a precondition in this context. On the other hand, large roof, concrete and asphalt surfaces in densely populated areas mean that a great amount of activity that is bound to the rainwater will flow into the sewer system. This effect is not taken into consideration.

4.3 Individual Aspects

R e a c t o r P r e s s u r e V e s s e l – There is no directly applicable operating experience for an assessment of the failure frequency of reactor pressure vessels in the German Risk Study. The application of the approx. 1,500 reactor operation years achieved with 206 reactor units at a worldwide level in late 1977 is not a suitable tool since this data base is still too small.

The failure frequency of conventional pressure vessels cannot be applied directly to reactor pressure vessels, as there are major differences with regard to design, manufacture and quality assurance. Their assessment permits the conclusion that the quality of reactor pressure vessels is far superior to that of conventional vessels. Pressure vessel experts agree that as a result of inherent causes there is no possibility for the sudden rupture of a reactor pressure vessel. Nevertheless, a failure frequency of 10^{-7} per reactor operation year was postulated for this accident cause in Phase A of the German Risk Study. This figure does not make any noticeable contribution to the total risk.

S t e a m E x p l o s i o n – While most activity releases are initiated by a late overpressure failure of the containment, a postulated steam explosion could lead to an early release as a result of the destruction of the containment vessel. The preconditions of such an accident sequence are:

- fragmentation of the molten core to particles of a size up to a few millimeters,
- even dispersion of the molten core fragments in the coolant,
- sufficient duration of these special heat transfer conditions and
- excessive loads on reactor pressure vessel and containment vessel.

The investigations reveal that a destruction of the containment vessel as a result of a steam explosion is very unlikely. The

physical conditions mentioned before would have to coincide in spite of the fact that each of them is already rather unlikely taken alone.

Because of the impossibility of clarifying the open problems in connection with the molten core, the fragmentation and the heat transfer at short notice, the statement for the occurrence frequency of a steam explosion as contained in the US Reactor Safety Study is taken over for Phase A. This results in the largest activity releases as compared with all other event sequences. Other early releases, though not specified in detail, are covered as well by this approach.

The loads determined for the steam explosion suggest that the reactor pressure vessel will not be destroyed, although an unequivocal proof to this effect has not yet been furnished. This is why the relevant questions in this context should be treated in greater depth in Phase B of the Study.

External Events - This heading covers initiating events such as earthquake, flooding, storm, lightning, aircraft impact, explosion pressure waves and the like. As, in consideration of their frequency, they can make a contribution to the total risk because of

- weak design spots,
- the exceeding of load assumptions and
- the failure of safety features,

they have to be included in the investigations to be carried out.

With regard to the earthquake analysis it is determined that the probability of exceeding the safe shutdown earthquake (which is considered in the design of the plant) is between 10^{-4} and 10^{-3} per year. The resulting accident sequences would necessitate comprehensive quantitative analyses. The decisive question in this context would be whether a loss of coolant would occur as a result of the earthquake. This would involve modified fail-

ure rates which are determined by structural failures or multiple component failures. The results obtained so far reveal that structural failures will not yet make any noticeable contribution to the total risk. However, in-depth investigations should be carried out in Phase B of the Study.

For the determination of the occurrence frequency of extreme river water levels there are no fully developed models available at present. Low water levels and the associated problems for the supply of cooling water will not endanger the plant. For extreme high water levels it cannot be excluded in the same way, that the function of safety features will be affected. Following the practice of the American Reactor Safety Study the German Risk Study assumes, that extreme high water levels give no dominant contribution to the frequency of core meltdown. This question should also be treated further in Phase B of the Study.

As a result of the structural design, tempests involving wind loads as they can be anticipated in the Federal Republic of Germany will not endanger the plant. Even in unfavorable conditions, they will cause no more than the emergency power case. However, the occurrence frequency of the emergency power case caused in this way is far smaller than the occurrence frequency of the emergency power case due to in-plant events. Lightnings will hit the plant relatively often. Although a quantitative analysis could not be performed because of a lack of data, rough considerations show that a lightning will not make any noticeable contribution to the total risk.

An occurrence frequency of 10^{-6} per year is derived for the impact of a fast flying military aircraft. Civil air traffic constitutes an even smaller hazard. Because of this small numeric value, detailed analyses can be waived. For the buildings concerned, upper limits for the consequential event of a core meltdown are estimated. As the frequency is smaller than $2 \cdot 10^{-7}$ per year, no noticeable contribution to the total risk will result.

The investigation of the effects of explosion pressure waves reveals that the occurrence frequency of design basis loads is already between 10^{-5} and 10^{-7} per year. A core meltdown as a result of load-dependent and/or independent accidental failures of systems is even less likely. In consideration of the pessimistic assumptions for the location of deflagration and for the gas cloud model no noticeable contribution to the total risk results either. Due to the safety distances the same applies to pressure waves caused by detonations or processes similar to detonations.

5. OTHER RISKS

Comparisons between accident-related risks of nuclear power plants and other natural and man-induced risks are problematic for a number of reasons. The two most important of these reasons are:

- The damaging effects of the different kinds of accidents are not the same. If e.g. only early fatalities are compared this is a random limitation of the material to be used for comparisons (deaths occurring after decades might be judged more serious than acute deaths).

- The comparisons of risks already performed involve, on the one hand, risks for which empirical data are available (e.g. traffic systems) and, on the other hand, risks which have been determined analytically (e.g. nuclear power plants). Empirical material will not consider the latest state of a technical development, whereas analytical material is subject to the imperfections of calculations and their limiting boundary conditions and is thus restricted with regard to its inherent significance.

Comparative risk analyses in various industries, such as for different facilities for the generation of energy, have so far only been attempted. In industrial chemistry, which is similar

to nuclear engineering in that it involves a great hazard potential, there are only a few investigations which might be used. In this context, reference may e.g. be made to the British Canvey Island Study which quotes, for greater occurrence frequencies, comparable numbers with regard to calculated acute deaths. Similar to nuclear engineering, the analysis shows how the frequency, though not the number of fatalities, can be reduced by means of additional safety measures.

The German Risk Study does not include comparisons with the risks of other industries, although their justification and their value are actually recognized if all boundary conditions are taken into consideration. However, what is directly comparable is the expectation values for late fatalities (deaths as a result of leukemia and cancer) resulting from accident-related releases of activity in consideration of 25 plants and the spontaneous deaths through leukemia and cancer in one and the same population. The expectation values are obtained by summing up the collective fatalities multiplied by the occurrence frequencies. The corresponding numeric values (approx. 10 calculated accident-related deaths per year as compared with 1,890,000 spontaneous deaths per year as a result of leukemia and cancer) differ by several powers of ten. A favorable ratio will result even if the natural background radiation is employed for the Dose-Risk Correlation used in the Study (approx. 10 calculated accident-related deaths per year as compared with 8,400 calculated deaths per year resulting from natural background radiation). The expectation values of genetically significant collective doses resulting from accident-related releases and from natural background radiation are similar (64,000 man-rem/year to 67,000,000 man-rem/year).

In Fig. 10, the expectation values of accident-related individual fatalities (late fatalities), standardized to 1 plant, have been plotted as a function of distance. This individual risk is at least 4 or 2 powers of ten below the total expectation values - assumed as independent of a particular site - for spontaneous death from leukemia and cancer or that share of the ex-

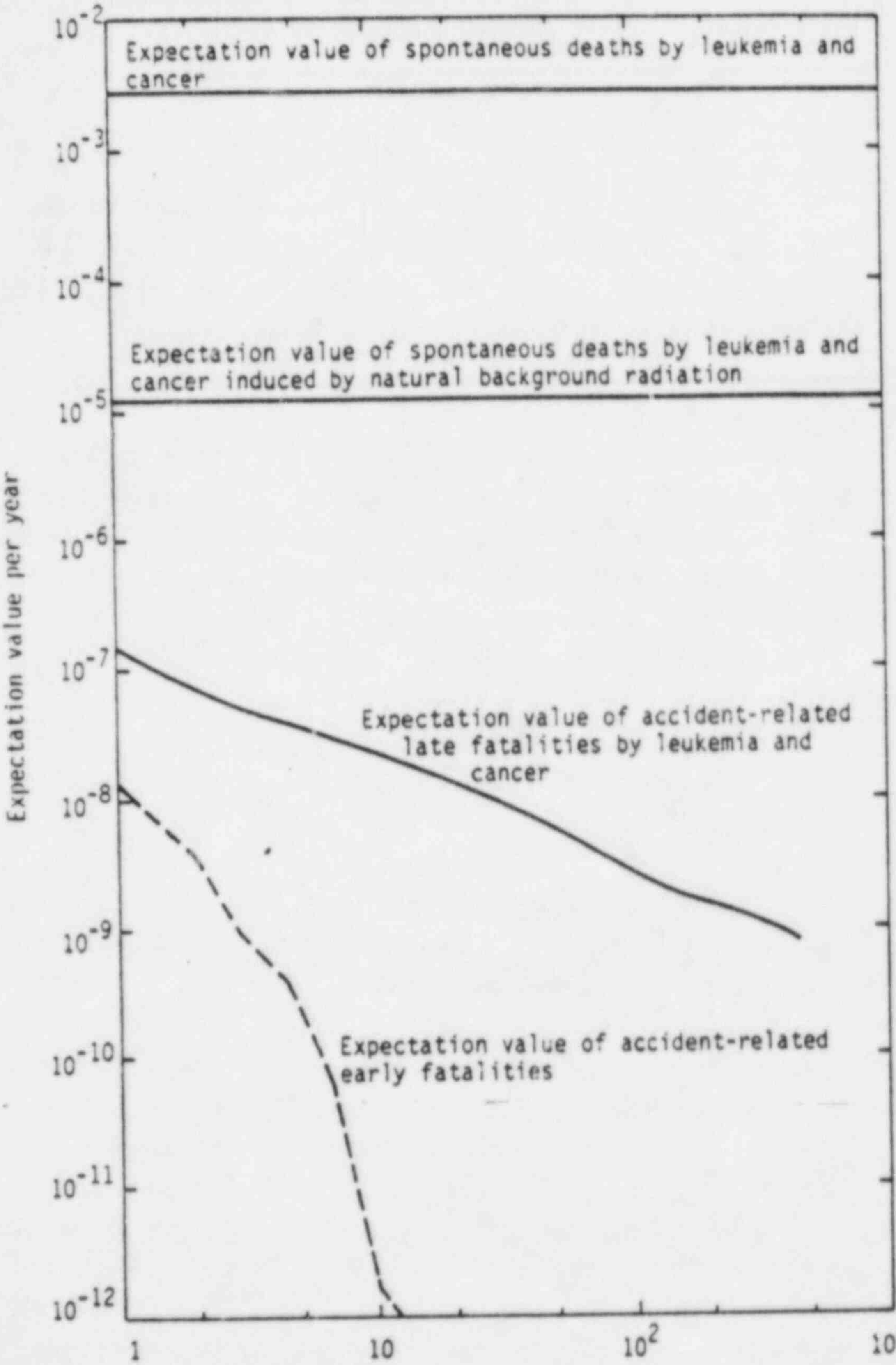


Fig. 10: Expectation values of accident-related individual fatalities due to the distance from the plant

For comparison: Expectation values of spontaneous deaths by leukemia and cancer

pectation values that is due to natural background radiation. The expectation values for accident-related individual fatalities (early fatalities) show that acute deaths are obtained up to a distance of no more than approx. 20 km.

6. COMPARISON WITH WASH-1400

In spite of different engineered plant features and different site conditions, the results of the German Risk Study more or less confirm those of the US Reactor Safety Study. This applies to both early fatalities (acute deaths) and late fatalities (deaths caused by leukemia and cancer). Fig. 11 shows a comparison for early fatalities, Fig. 12 the corresponding comparison for late fatalities. However, the comparisons are difficult to make as with regard to the American results only the median values are quoted and not the expectation values. The different unit sizes of the plants under review were not considered in this context. As far as early fatalities are concerned, the German figures are in general lower than the American figures, but they are higher than the American figures with regard to late fatalities. However, as the uncertainties of the German and American curves overlap to a great extent, no fundamental difference can be seen in the risks which have been identified.

It was not possible to perform the investigations completely analogous with the approach used in the US Reactor Safety Study with regard to all aspects involved. Essential deviations concern the following aspects:

- The differences between engineered plant features in American and German nuclear power plants led to different areas of major interest, in particular with regard to the reliability investigations. Thus the first few preliminary results of the event tree and fault tree analyses suggested that the investigations required for the assessment of the risk contribution of transients will have to be more detailed than originally scheduled.

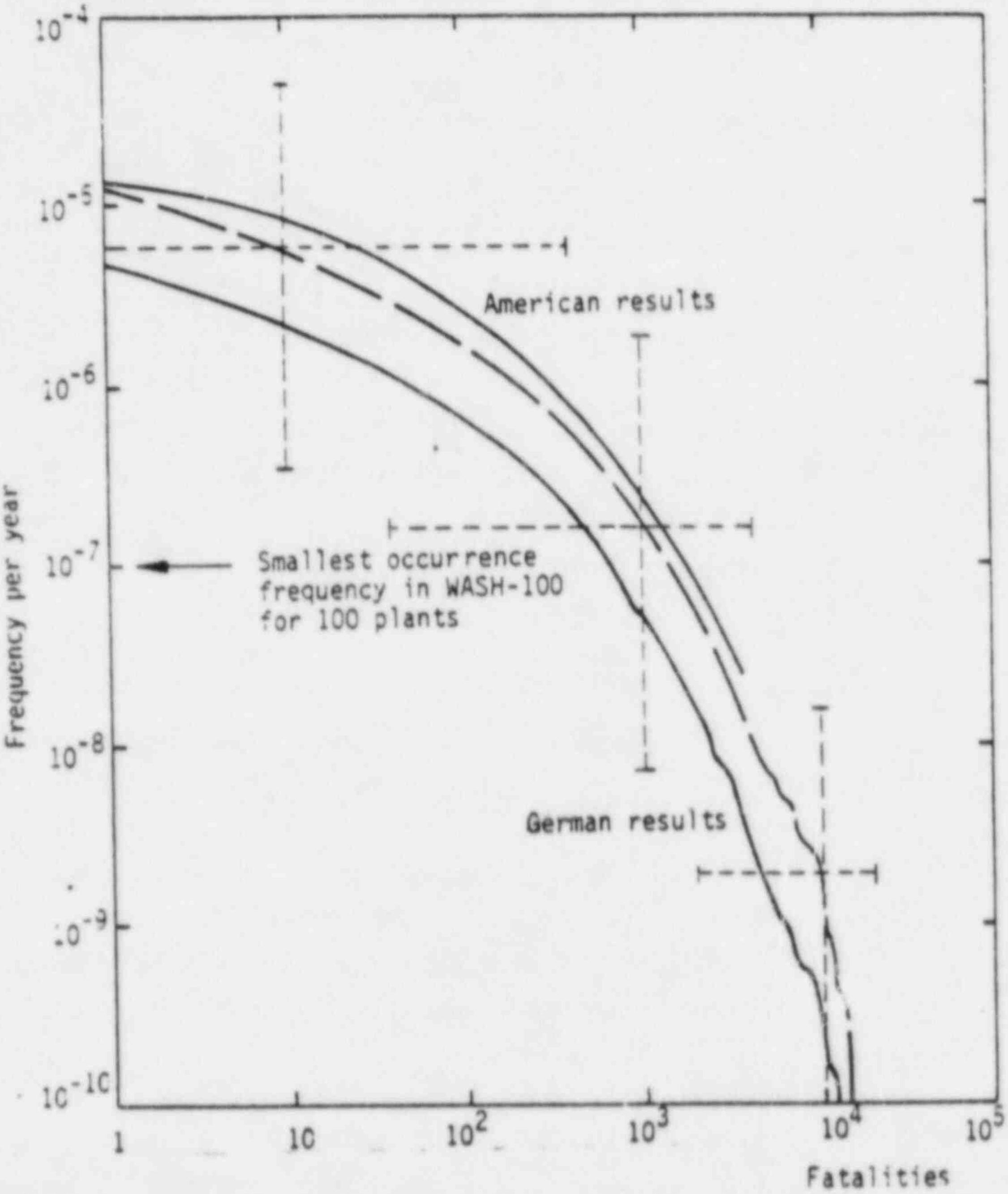


Fig. 11: Cumulative frequency distribution of early fatalities per year corresponding to 25 plants

The thick broken line with thin broken bars indicating 90 % confidence limits results from a different averaging method used in the German Risk Study.

For comparison: Median values of German Risk Study and American Reactor Safety Study

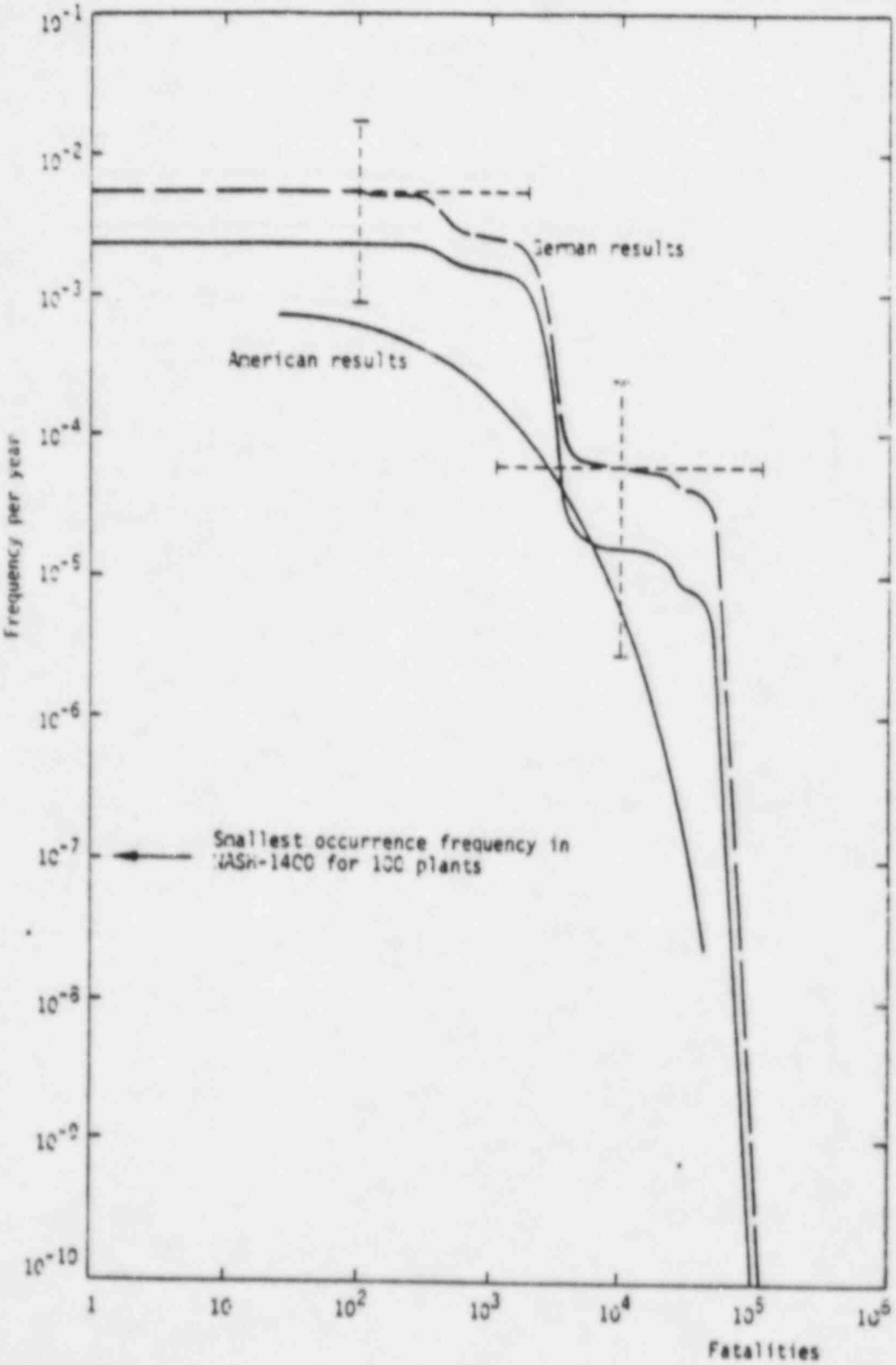


Fig. 12: Cumulative frequency distribution of late fatalities per year corresponding to 25 plants

The thick broken line with thin broken bars indicating 90 % confidence limits results from a different averaging method used in the German Risk Study

For comparison: Median values of German Risk Study and American Reactor Safety Study

- The accident consequence model used in the German Risk Study corresponds to that of the US Reactor Safety Study as far as its basic characteristics were concerned. In several individual aspects the calculational model had to be adapted to German site conditions. In particular, an independent model of protective actions and countermeasures was used which considers the basic recommendations for official emergency control in the Federal Republic of Germany.

- For the determination of early fatalities an independent Dose-Effect Correlation was derived with due regard to recent findings. However, this correlation does not lead to any noticeable difference as compared with the corresponding American correlation.

- For the evaluation of late fatalities, a purely proportional Dose-Risk Correlation, which is independent of any dose rate, is used together with the risk factors quoted by the International Commission on Radiological Protection (ICRP 26). The use of the American, instead of the German, correlation will considerably reduce the calculated number of late fatalities; thus, the maximum number of fatalities is reduced to about one-half.

Methods and objectives of risk investigations have been under discussion to an ever increasing degree in recent years. Suggestions which have been made in the debate concerning the US Reactor Safety Study have been taken into consideration to the extent permitted by Phase A.

7. EVALUATION OF RESULTS

Taken the present state of the art, the inherent significance of risk analyses is restricted. This applies also to the results of the German Risk Study whose evaluation will have to consider both these inevitable limitations and the basic conditions contained in the contract. The investigations put major

emphasis on the determination of the collective risk. Only a mean individual risk can be derived on this basis. It cannot be associated with any given site.

The Study deals only with the risk caused by incidents and accidents at nuclear power plants. Risks due to specified normal operation or other installations of the fuel cycle are not dealt with. Similarly, contributions originating from war, terrorism, sabotage and the like are not investigated either.

Although the design conditions of a typical plant are the basis of the analysis of engineered plant features, numerous data were introduced which have been obtained in other, more or less comparable, plants. The calculation of accident consequences was based on typical German site conditions. Thus the results cannot be applied to a certain plant but can only be used as a model.

The model character of the German Risk Study becomes obvious in the investigation of event sequences which are associated with activity releases. For this purpose, models are used which describe core meltdown, activity release and activity dispersion as well as biological radiation effects. The lack of detailed knowledge is compensated by simplifying and pessimistic assumptions, i.e. such as will also include the most unfavorable case.

The available methods and data are the reason why risk analyses are still associated with considerable uncertainties. As far as there is a workable basis an attempt was made to quantify the degrees of certainties. The uncertainties which cannot be quantified are covered by the pessimistic procedure. This means that the risk is rather overestimated than underestimated. Accordingly, subjective 90 % confidence limits have been plotted in Figs. 7, 9, 11 and 12. They say that as a result of the quantified uncertainties 5 % of the obtained values will be above, 90 % within and 5 % below the confidence limits quoted.

In many instances, the German Risk Study relies on existing investigations. Thus, for example, the simulation of accidents as it is carried out within the frame of the nuclear licensing process constitutes the basis for the establishment of minimum requirements for safety systems.

An essential part of the risk analysis for nuclear power plants is constituted by the investigation of engineered plant features, i.e. the determination of the probability that a core cooling failure will occur and lead to a core meltdown and, following a failure of the containment vessel, to a release of radioactive effluents to the environment. These investigations of engineered plant features will also include the availability of safety features. The investigations thus permit an analysis of weak spots in the systems under review.

Risk analyses will not only investigate the effect of certain components on the reliability of the systems. A calculation of the core meltdown frequency will also permit an estimate of the risk contribution of the individual systems and components. In addition, such analyses enable an objective evaluation of certain safety features. The German Risk Study confirms the results of the US Reactor Safety Study, i.e. that the containment vessel of a nuclear power plant will reduce the effects of serious accidents to a greater degree than originally anticipated.

Experience shows that problems will tend to turn up at interfaces between different systems and between different technical disciplines. As risk analyses, and in particular the systematic event tree analysis and reliability analysis, require a procedure involving more than a single system or discipline with respect to all important features of a plant, such problems will be identified with a greater degree of safety than in the customary safety assessment.

Reliability analyses of safety-related systems have already become part of the nuclear licensing process. They include in particular system comparisons and optimations as well as the establishment of testing and maintenance strategies.

In the safety-related design of nuclear power plants, a probability concept has always been used implicitly when, on the basis of existing engineering experience, a decision has to be made as to which accident sequences will have to be coped with by safety features. In many cases, decisions on protective requirements with regard to external events are made dependent on their occurrence frequency. Findings with respect to typical accident sequences may be used for the planning of emergency measures. They convey an idea of the periods of time during which protective actions are necessary and possible and, to a certain degree, enable an estimate of the effect of different protective actions and countermeasures. The results of risk analyses are apt to identify starting points where particularly effective further research and development studies concerning reactor safety can be initiated.

However, the possible applications of risk analyses are limited. Risk analyses use probabilities which are, in most cases, small or very small. Thus, the results, whose character is that of an estimate anyhow, will become even more uncertain with decreasing probability and thus an increasing number of fatalities. It has to be doubted whether the state of the art is sufficient, in the case of events involving probabilities of 1 : 1,000,000,000 per year, to determine somewhat reliable results which can serve as a basis for assessment and decision. Furthermore, it remains doubtful whether events involving the frequency mentioned before or an even smaller frequency can still be made part of realistic considerations at all.

The implementation of risk analyses will also lead to problems of a psychological nature which may counteract the purpose of risk analyses. Events which are impossible as far as one can humanly foresee will assume a real character as a result of a detailed analysis. Thus, people are made aware of possible hazards which in all probability will never result in fatalities and which do not play any role in the minds of most people. This may involve the paradoxical consequence that certain risks are demonstrated to be minimal but that fear of these

risks is increased by the very demonstration, whereas far greater risks which may not have been investigated in detail will not be taken not of at all.

8. NOTES

The results of Phase A of the German Risk Study have been compiled in a Main Volume containing a total of 9 Chapters (1. Objectives, Layout and Organization of the Study; 2. Fundamental Remarks on the Identification of Risks; 3. The Nuclear Power Plant; 4. Subject Matter and Method of the Risk Analysis; 5. Results of the Event Tree Analysis; 6. Release of Fission Products; 7. Accident Consequence Model; 8. Results and Inherent Significance of the Results; 9. Conclusions).

Because of its particular importance, the accident at Three Mile Island of March 1979 is dealt with in an Appendix to the Main Volume. The events are discussed in connection with the investigations carried out in the Study.

As supplements to the Main Volume, a number of specialized volumes are planned covering the following subjects: (1) Event Tree Analysis; (2) Reliability Analysis; (3) Reliability Data and Operational Experience; (4) External Events; (5) Investigations of Core Meltdown Accidents; (6) Determination of Fission Product Release; (7) Results of the Investigations of Engineered Plant Features; and (8) Accident Sequence Calculations and Risk Results. These specialized volumes are to contain the detailed documentations of the investigations so far carried out for the Study. Thus the interested reader will have a possibility of repeating and evaluating in detail results of the investigations beyond the present report.

The present report constitutes only the first part of the German Risk Study (Phase A). The second part (Phase B) is to provide an in-depth discussion of the individual problems with the participation of further institutions and persons. In this Phase,

the further development of methods and the latest state in reactor safety research should be taken into consideration to a greater extent.

Appendix

The Cologne-based "Gesellschaft für Reaktorsicherheit" is the main contractor for the German Risk Study to be performed on behalf of the Federal Minister for Research and Technology. Major contributors to this effort are:

- Gesellschaft für Reaktorsicherheit (GRS):
  Event tree analyses, fault tree analyses, description of core meltdown accidents, determination of activity releases

- Kernforschungszentrum Karlsruhe:
  Preparation of the accident consequence model, calculation of accident consequences

- Gesellschaft für Strahlen- und Umweltforschung:
  Establishment of the Dose-Effect and Dose-Risk Correlations

Moreover, the following participated in the solution of further particular aspects:

- Technische Universität Berlin, Institute for Nuclear Engineering:
  Determination of reliability data

- Technische Universität München, Department of Measuring Technology:
  Determination of reliability data for electronic components

- TÜV-Arbeitsgemeinschaft Kerntechnik West:
  Evaluation of the VdTÜV statistics of damage to conventional pressure vessels and boilers

- Universität Stuttgart, State Materials Testing Laboratory:
  Assessment of the reactor pressure vessel

- TÜV Norddeutschland:
  Evaluation of operational experience

- Ingenieurbüro F. Mayinger & Co., Barsinghausen:
  Investigations relating to the steam explosion

- König & Heunisch, Consulting Engineers, Frankfurt:
  Assessment of building structures in connection with the impact of earthquakes

- TÜV Rheinland, Institut für Unfallforschung:
  Studies relating to the model of protective actions and countermeasures

- Bonnenberg + Drescher Ingenieurgesellschaft mbH (B+D), Aldenhoven:
  Provision of population data