The German Risk Study for Nuclear Power Plants

by A. Birkhofer

In August 1979 results of the "German Risk Study for Nuclear Power Plants" were published. The Main Report, in which approach and results of the study are documented, has been available since the end of 1979.

It was the charter of the study – which was performed on behalf of the Minister of Research and Technology of the Federal Republic of Germany – to apply as far as possible the methods of the US Reactor Safety Study (WASH-1400) to German plant and site conditions.

A direct transfer of the results was not deemed justified, mainly for the following reasons:

- There is quite a number of differences between the design of the reference plants of WASH-1400 (Surry-1, Peach Bottom-2) and German nuclear power plants.
- The mean population density in the Federal Republic of Germany is more than ten times that of the United States. In the vicinity of nuclear power plants the ratio is about 3:1.

The Gesellschaft fur Reaktorsicherheit was the main contractor and performed most of the plant analysis The Kernforschungszentrum Karlsruhe performed the calculation of accident consequences In matters of health effects of radiation, assistance came from the Gesellschaft fur Strahlen- und Umweltforschung. Further institutions have contributed to special problems

As reference plant for the technical part of the analysis, Biblis-B, a KWU-type* PWR** with 3750 MW thermal power, which started commercial operation in March 1976, was used

To calculate the collective risk resulting from reactor accidents, a total of 25 plants at 19 different sites in the Federal Republic of Germany were considered. This included all plants with 600 MW or more electrical output, which were in operation, under construction or in licensing process by July 1, 1977. As an approximation to the real situation, it has been assumed that all 25 plants are technically identical to the reference plant.

* Kraftwerk Union AG.
** Pressurized light-water reactor.

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Methods and Results of the Study

Each risk can be characterized by probability and extent of potential damages. With respect to nuclear power plants, neither of these components can be determined from direct experience. In more than 25 years of reactor operation, there have been many operational incidents and also a number of serious accidents. However, no one has been killed or injured outside a plant by a nuclear accident.

Therefore, the only way to estimate the risk, which remains in spite of extensive safety measures, is by analytical means.

The safety concept applied in nuclear power plants ensures that accidents do not cause a dangerous release of radioactive material into the environment as long as a minimum number of the redundant engineered safety systems are properly operating. Hence, a risk analysis has deliberately to assume failures of safety systems, since only those events may lead to fission product release and therefore contribute to the risk.

A rough survey of the radioactive inventory of a nuclear power plant indicates, that – on time average – fission products in the reactor core constitute about 95 per cent of the total inventory. Therefore, it is justified to consider mainly such events which could lead to serious releases of fission products from the core.

For an estimation of risk, releases can be neglected as long as damages to the core and especially a meltdown of the core are prevented Therefore risk analysis has to deal with severe overheating of the core, the prevention of which is a central task of reactor safety.

"Initiating events", potentially leading to core damage by insufficient cooling, usually are grouped into two types:

(i) loss-of-coolant accidents, initiated by a leak or a break in the reactor coolant system;

(ii) transients, leading to an imbalance between the heat generated in the core and the heat removed from the core, caused by events different from loss-of-coolant accidents.

It is most probable that after an initiating event the plant will be brought to a safe state by means of control and safety systems. However, if systems essential to maintain sufficient core cooling failed, overheating and finally meltdown of the core would result. Depending on the specific initiating event, this may require successive failures of a number of different systems. Considering operational transients, the functioning of safety systems would generally be necessary only if control systems fail. More severe events, however, require immediate functioning of safety systems.

In order to record clearly the possible event sequences induced by an initiating event, "event trees" have been established. The trunk of an event tree stands for the initiating event. The tree branches at points where the event is influenced by success or failure of the various systems. Consequently, the branches of the tree represent the possible accident sequences. The frequency of occurrence of the respective accident sequence is given by the frequency of the initiating event and by the probability of success or of failure of the systems involved.

TABLE 1: Summary of the Results of Event Tree Analysis

Accident Initiating Event	Probability of Occurrence of the Initiating Event per Reactor Year (P_1)	Failure Probability of Required Safety Functions (P_2)	Probability of Occurrence of Core Melt per Reactor Year ($P_3 = P_1 \times P_2$)
Large LOCA[1]	2.7×10^{-4}	1.7×10^{-3}	5×10^{-7}
Medium LOCA[1]	8×10^{-4}	2.3×10^{-3}	2×10^{-6}
Small LOCA[1]	2.7×10^{-3}	2.1×10^{-2}	$5\ 7 \times 10^{-5}$
Loss of Off-Site Power	1×10^{-1}	1.3×10^{-4}	1.3×10^{-5}
Loss of Main Feedwater Supply	8×10^{-1}	4×10^{-6}	3×10^{-6}
Emergency Power Case with Small Leak at Pressurizer	2.7×10^{-4}	2.6×10^{-2}	7×10^{-6}
Other Transients with Small Leak at Pressurizer	1×10^{-3}	2×10^{-3}	2×10^{-2}
ATWS-Events[2]	3×10^{-5}	3×10^{-2}	1×10^{-6}

[1] Loss-of-Coolant Accident.
[2] Anticipated Transients Without Scram.

The frequency of transients is estimated mostly on the basis of operation experience. Data have been obtained from literature for loss-of-coolant accidents.

Safety systems in nuclear power plants are designed to perform with high reliability. As a consequence, the probabilities of failure of those systems are generally not known from direct experience. Therefore, they have been calculated by means of fault tree analyses. In a fault tree the functional interaction of system components is translated into a logical structure. According to this structure the probability of system failure can be calculated, starting from the unavailability of components. In principle, the influence of human actions and of external impacts can also be taken into account.

Reliability analysis has to assess which part of a redundant system would be sufficient to perform the required function. For this purpose the study relied upon the safety analysis of the licensing procedure. A redundant system has been assumed to fail totally, if fewer sub-systems are available than have been considered for the safety analysis. There are good reasons to believe that, in a real situation, degraded system would maintain core integrity.

It has already been mentioned that for risk analysis mainly those events have to be traced which lead to meltdown of the reactor core. Only in this case could a large amount of fission products be released.

To calculate core melt frequency the study has considered about 70 accident sequences in some detail. Summing up all relevant contributions, an overall core melt-frequency of about 9×10^{-5} per year has been calculated. Table 1 summarizes the results of event tree and fault tree analyses. Figure 1 shows the relative influence of different initiating events on the overall core melt-frequency.

A loss of main coolant through a small leak in a reactor coolant pipe dominates all other contributions, mainly for the following reasons:

- Small leaks may occur more frequently than medium or large breaks.
- The secondary system is necessary to remove the decay heat and to cool down the reactor. In the reference plant, the function of this system has to be initiated and controlled by the operators. This reduces the system reliability significantly.

The second largest contribution results from transients, the loss of offsite power playing an important role.

For many transients an increase of the primary system pressure has to be expected. The activation of pressurizer relief, or safety valves, could become necessary to protect the system from undue overpressure. In this case, a transient can develop into a small leak if a valve failed to close after it had opened. This sequence played a dominant role at the time interim results of the study were published in November 1977. Plant improvements reduced its probability considerably.

It is not surprising that the contingency of a large break is quite small. This accident has been studied very extensively for many years as a basis design accident. Engineered safety systems have been optimized to cope with this kind of accident.

After the release of fission products from the core, the deposition processes in the containment atmosphere as well as the state of containment integrity determine the amount of fission product release into the open atmosphere.

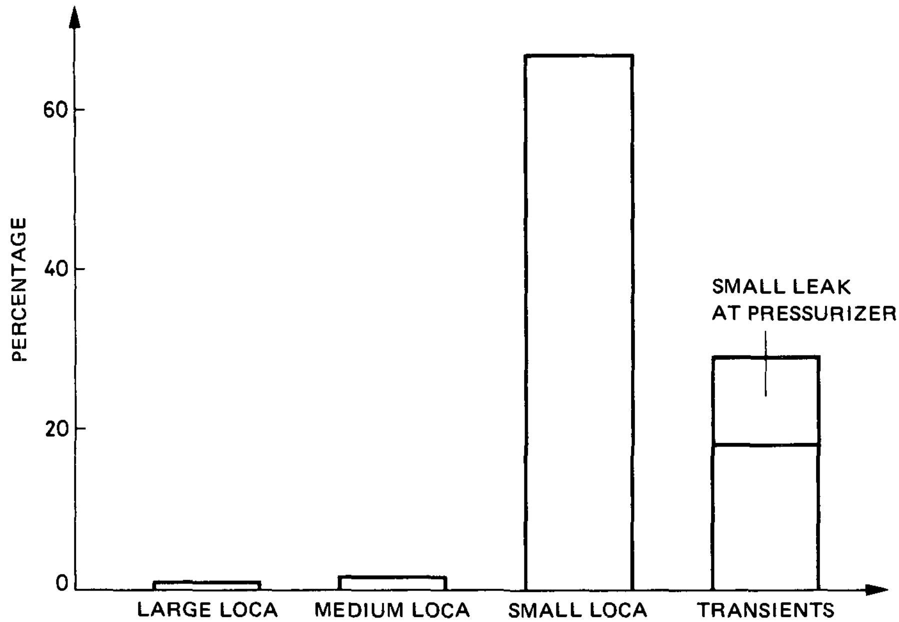

Figure 1. Relative contribution of various accident initiating events to the probability of core melt.

During meltdown, molten material would fall down into the lower plenum of the pressure vessel. Decay heat generated in the molten fuel is assumed to be sufficient to melt through the lower head of the reactor pressure vessel and even to melt into the concrete structures of the reactor pit and the containment foundation. Melt-through of the reactor building foundation has been calculated to occur about 100 hours after accident initiation.

During meltdown and melt-through, not only parts of the radioactive core inventory are released, but also large amounts of steam and hydrogen are generated, resulting in an increase of containment pressure and temperature. It has been calculated that, about 25 hours after accident initiation, the containment will fail because of overpressure.

Prior to this "late overpressure failure", containment integrity may be lost due to failure of containment isolation or by a "steam explosion". When the molten core falls into water in the lower plenum, a rapid evaporation of the water takes place. However, an energy release sufficiently large to endanger primary system integrity would require very fine fragmentation of large parts of the molten core.

By combining results of core melt analysis and containment behaviour analysis, amount and frequency of fission product release from the plant to the atmosphere are obtained.

To simplify the further analysis, accident sequences resulting in the same containment failure mode are grouped together into one of several release categories.

TABLE 2: Times of Release and Probabilities of the Release Categories

Release Category Number	Description	Time of Release (hours)	Probability[1] per Reactor Year (mean)
1	Core melt, steam explosion	1	2×10^{-6}
2	Core melt, large containment leak (300 mm ϕ)	1	6×10^{-7}
3	Core melt, medium containment leak (80 mm ϕ)	2	6×10^{-7}
4	Core melt, small containment leak (25 mm ϕ), "Late containment overpressure failure"	2	3×10^{-6}
5	Core melt, "Late containment overpressure failure", failure of filter systems	25	2×10^{-5}
6	Core melt, "Late containment overpressure failure"	25	7×10^{-5}
7	Design basis accident, large containment leak (300 mm ϕ)[2]	0	1×10^{-4}
8	Design basis accident[2]	0	1×10^{-3}

[1] Probabilities are calculated including 10 per cent contributions from adjacent release categories
[2] No core meltdown accidents

Categories 1 through 6 comprise core melt accidents. The probability is about 95 per cent that a core meltdown will be followed by a "late overpressure failure" of the containment. These events are included in the categories 5 and 6. In category 5, additional failure of filter systems prior to "late overpressure failure" of the containment is assumed. Categories 2 through 4 comprise core melt accidents with failure of containment isolation, openings ranging from a large leak (300 mm equiv. diameter) to a small leak (25 mm equiv. diameter) In these cases, fission product releases, particularly for a large leak, are significantly higher than those shown in categories 5 and 6. Category 1 contains the most severe releases. It has been assumed that reactor pressure vessel and containment are seriously damaged by a steam explosion after core meltdown. The present status of analytical and experimental

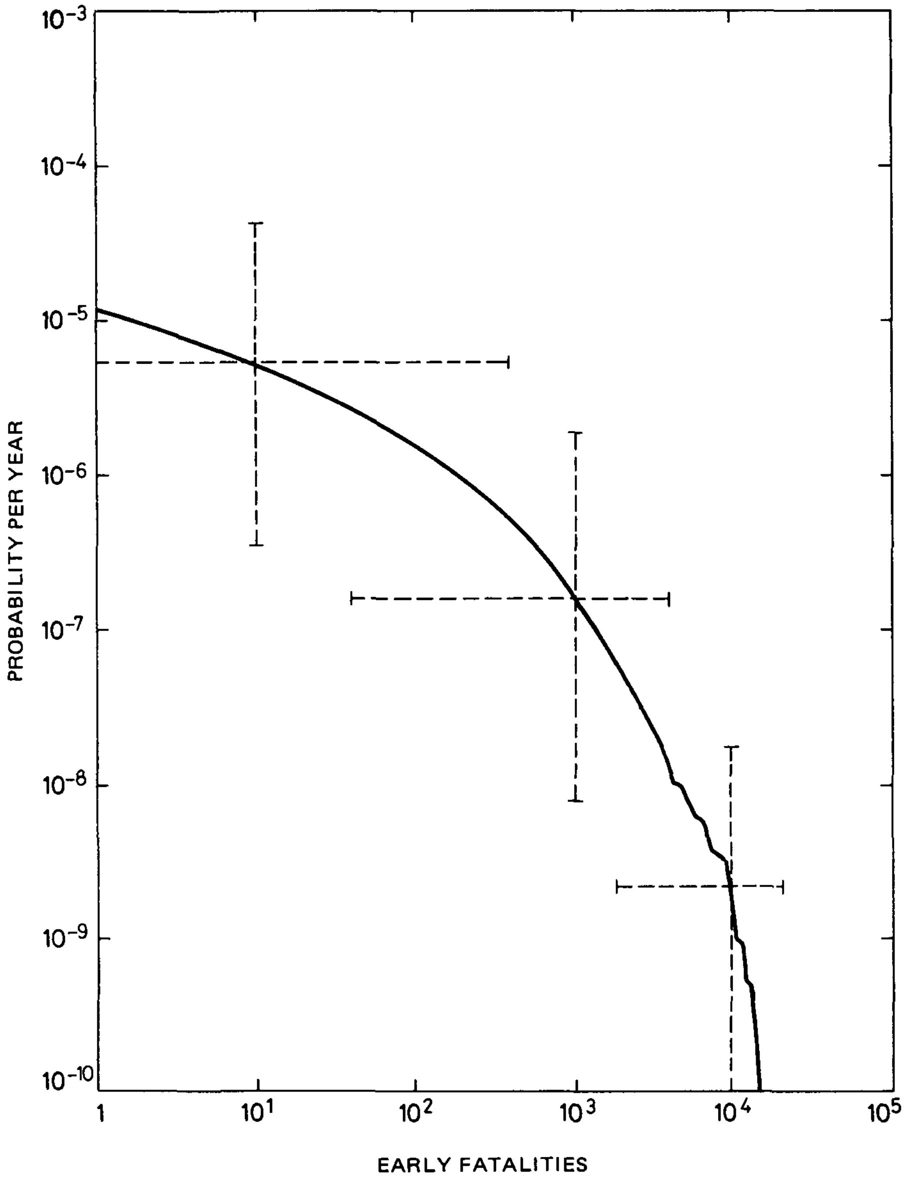

Figure 2. **Complementary cumulative distribution function for early fatalities per year for 25 plants (the dashed bars indicate 90% confidence limits).**

evidence shows that such an event is extremely unlikely. As a very cautious assumption, similar to WASH-1400, a two per cent probability has been assigned to the event that a core meltdown leads to a steam explosion, destroying the containment integrity.

In addition, the study has analysed loss-of-coolant accidents, assuming sufficient core cooling by the emergency core cooling systems. These events are grouped into **categories 7 and 8.** Fission products are released only because of postulated cladding failures. For category 7, failure of containment isolation by a large leak is assumed Subsequently to the technical part of the analysis, the dispersion of fission products by atmospheric transport and diffusion has been analysed. Weather conditions have been assumed according to records of actual data applying for different meteorological areas. Finally, the resulting radiation exposures, health effects from radiation exposure and — according to the population data — the number of individuals affected by health damage, have been calculated. For this, emergency procedures were taken into account based on governmental recommendations existing in the Federal Republic of Germany.

Main Results of the Consequence Calculations

In the following, the main results of the consequence calculations are compiled:

Figure 2 shows the correlation between number and frequency of acute fatalities which could be caused by radiation exposure to the public after a core melt accident With 25 plants in operation, a frequency of about 10^{-5} per annum that acute fatalities are caused has been estimated. It can be concluded from the figure that large-consequence events are extremely unlikely.

As for the results, the Study has made an attempt to quantify confidence intervals which are shown at selected points.

The very low frequencies calculated by the Study result from the product of several factors (Fig. 3).

Taking 25 plants into consideration, calculations show a core melt-frequency of 1 to 400 per annum. Given a core meltdown, fission product release to the open atmosphere is, in most cases, limited very effectively by the containement. There is only a chance of 1 out of 16 that potentially lethal doses would appear after severe containment failure. In this case, the consequences would depend on weather conditions and population distribution. The chance is 1 out of 10 that acute fatalities might occur after severe containment failure.

Altogether, the probability is more than 99 per cent that a core melt accident would not cause acute fatalities. A great number of fatalities could occur only if adverse weather conditions coincide with unfavourable site characteristics and the most severe accidents. This results in a very low probability of large-consequence-events.

The situation is different with respect to late health effects. Considerable numbers of late fatalities are calculated also for less severe accidents (Fig. 4). It has to be borne in mind that a linear dose-risk relationship has been used in the Study. That means that even the smallest radiation exposure is assumed to cause an increase of cancer risk.

Late health effects caused by radiation would appear after a delay of 10 to 20 years and be spread over large areas and over several decades.

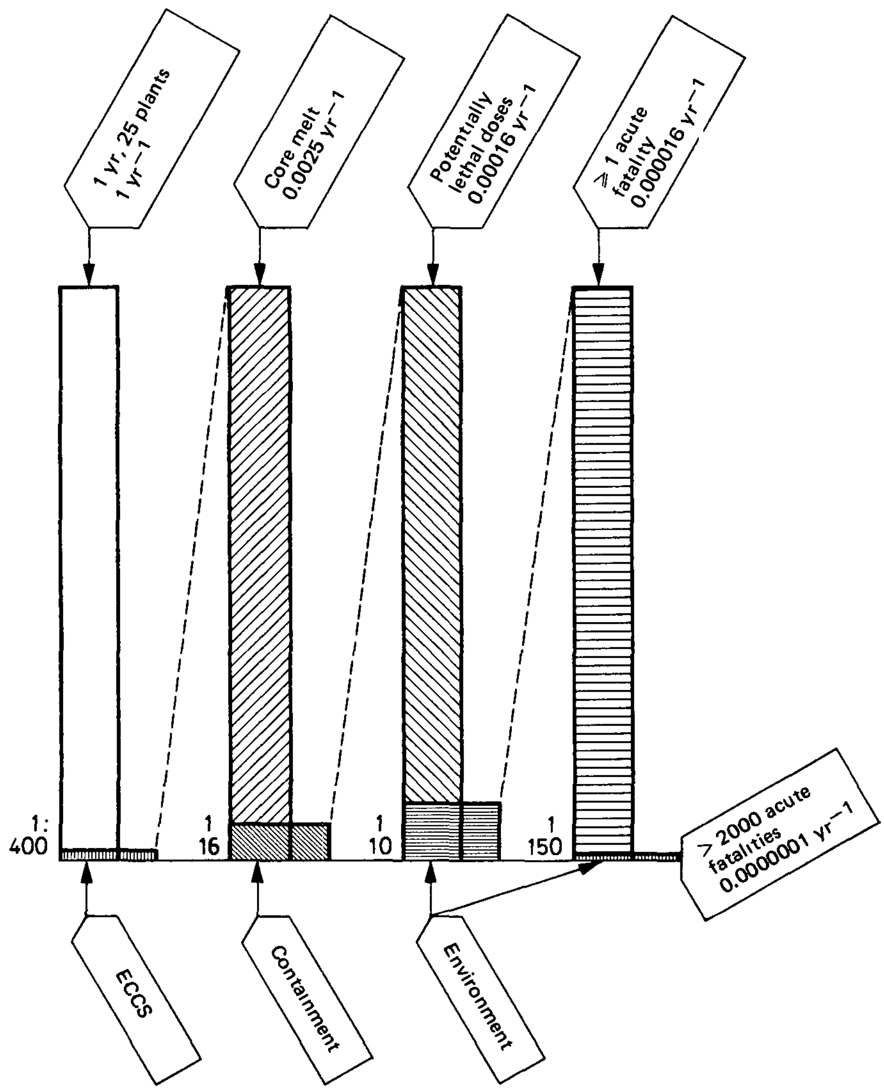

Figure 3. Frequency of accident consequences (per year, 25 plants).

On average, about half of the effects might occur outside the Federal Republic of Germany. This emphasizes the international importance of reactor safety.

It seems appropriate to put the number of late effects calculated by the Study into perspective. Applying the linear dose-risk relationship used in the Study, it can be calculated that about half a per cent of all cancer fatalities are caused by natural radiation. Although this is a relatively small percentage, the absolute figures amount to more than 50 000 for Germany, and about 600 000 for Europe, during a whole life-span, i.e. 71 years.

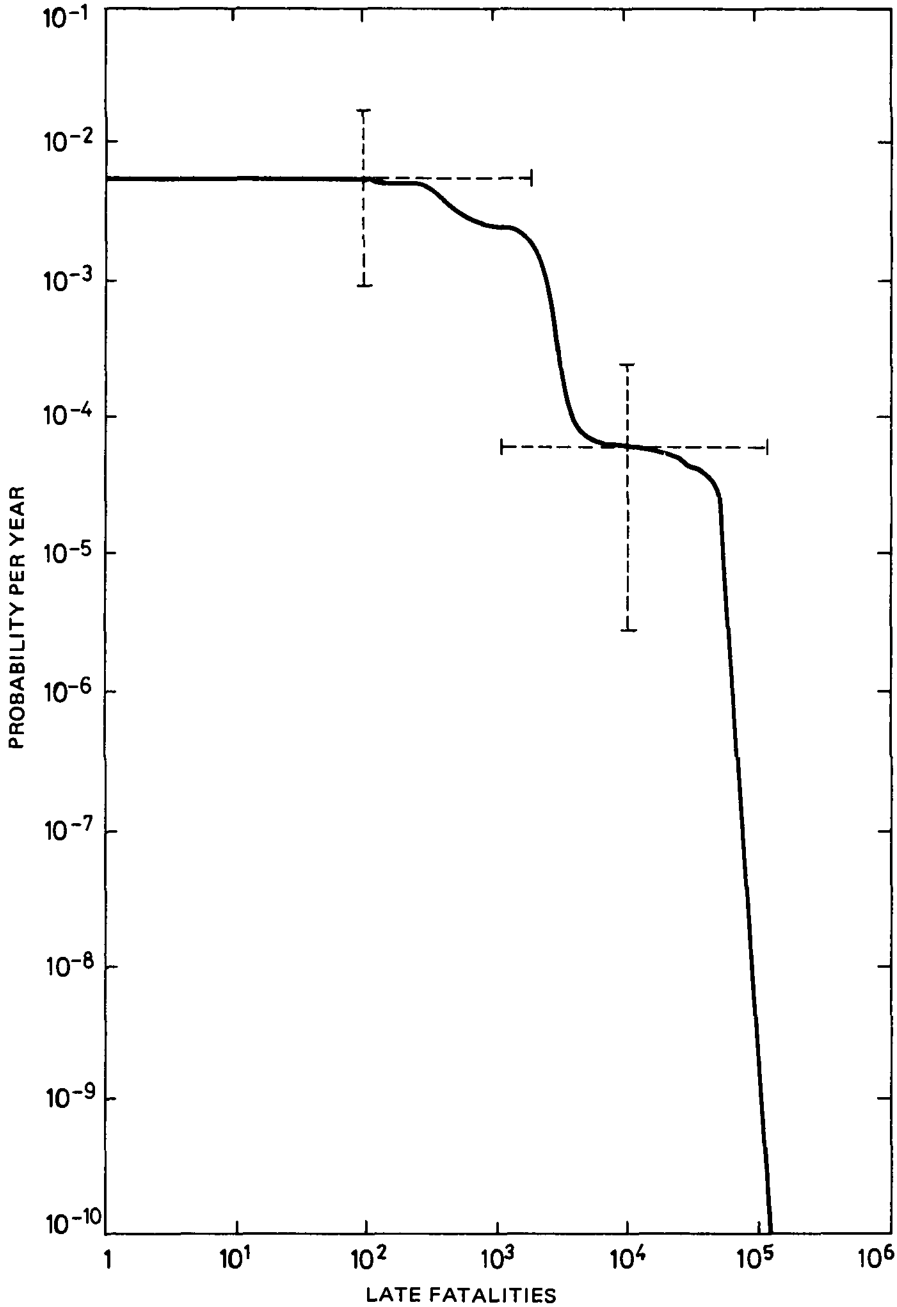

Figure 4. **Complementary cumulative distribution function for late fatalities per year for 25 plants (dashed lines indicate 90% confidence limits).**

Conclusions

Risk analyses are quite a new, yet effective, way to assess the level of safety of large technological systems. Although the uncertainty margins of results are considerable, valuable information can be gained from the analysis, provided that, for the interpretation, the limitations of the methods are taken into account. As a result of this Study, a number of possible system improvements in the reference plant have been identified. In some cases, a significant reduction of core melt probability could be realized by minor modifications of plant design.

The overall results of the Risk Study allow a rough comparative evaluation of risks from different sources. Although large-consequence nuclear accidents cannot absolutely be ruled out on a theoretical basis, the Study has calculated that the potential extent of health effects is not beyond that possible from other natural or man-made hazards However, the probabilities for a nuclear catastrophe are very low. This is in agreement with the results of WASH-1400.