German Risk Study—Main Report
A Study of the Risk Due to Accidents in Nuclear Power Plants

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Originally published by TÜV Rheinland in 1979 as *Deutsche Risikostudie Kernkraftwerke: Eine Untersuchung zu dem durch Störfälle in Kernkraftwerken verursachten Risiko.*

FOREWORD

The Federal Republic of Germany published the _Summary of the German Risk Study_ in August 1979 and subsequently made available an excellent English translation of that document. The main report of the _German Risk Study_ was released in early 1980, and this special EPRI report is its English translation. The technical appendixes will be published starting in late 1980. It is our understanding that the U.S. Nuclear Regulatory Commission (NRC) will prepare English translations.

The purpose of this special report is to make the _German Risk Study_ available to a larger audience and thereby promote technical consensus on proper methodologies for probabilistic risk assessment. A probabilistic risk assessment of a nuclear power plant is a large and complex task encompassing a broad range of technical disciplines and many subtle analyses. In its report (NUREG/CR-0400) to the Nuclear Regulatory Commission, the Risk Assessment Review Group (Lewis Committee) criticized the _Reactor Safety Study_ (WASH-1400) for inadequate peer review process. Given the fact that the _Reactor Safety Study_ was the first of its kind, such inadequacy was almost inevitable.

Although Phase A of the _German Risk Study_ is based upon _Reactor Safety Study_ methodology, in the process of doing the work the German team under Professor Birkhofer has probably performed an in-depth peer review of the _Reactor Safety Study_. The fact that the conclusions reached by the Germans on the public risk from Biblis B are comparable to those reached by the _Reactor Safety Study_ for Surry 1 and Peach-Bottom 2 does not guarantee the validity of the _Reactor Safety Study_ results. However, the similarity does lend confidence that the _Reactor Safety Study_ contains no major errors. It is evident from its report that the German team learned from the _Reactor Safety Study_ experience, and especially from the critical reviews thereon.

The German Risk Study is a noteworthy effort to communicate complex technical ideas. If other groups conducting major risk assessments can profit from reading this translation, then the primary EPRI purpose will have been achieved.

The benefit of a probabilistic risk assessment stems from identifying the accident sequences that dominate public risk rather than from stating the "bottom line" risk. The differences between the estimated risks in the German Risk Study and those estimated in the Reactor Safety Study are not statistically significant, and such comparisons do not appear particularly fruitful. The interested reader should be aware that the reference plant (Biblis) for the German Risk Study represents a state of the art advanced several years beyond the American counterpart (Surry). Another difference originates from the fact that contrary to the American approach, intermediate results of the German analyses were presented at various occasions. They were utilized to introduce modifications to the plant systems. Plant system modifications as of 1978 are incorporated into the study and its results.

In this translation, every effort has been made to render an accurate English equivalent while using the technical terminology commonly expressed within the Reactor Safety Study. Please bring any error to the attention of the undersigned.

Ian B. Wall
Nuclear Safety & Analysis Dept.

ABSTRACT

A translation of the Deutsche Risikostudie Kernkraftwerke, this report assesses the risks due to accidents caused by the operation of nuclear power plants in the Federal Republic of Germany. The study, performed under the direction of the Federal Ministry for Research and Technology, is organized into two phases: The current report presents an overview of the investigations and results of the first phase (Phase A), whereby the probabilistic risk assessment methodology used in the American Reactor Safety Study (WASH-1400) is applied to the particular reactor system technology and siting conditions of the FRG. Phase B, planned primarily to intensify study of individual problem areas, will employ further methodological developments and will consider the current status of safety research.

To assess risks, all sites in the FRG with LWR plants of at least 600 MW(e) that were in operation, under construction, or approved for construction on July 1, 1977 were included. Thus, 19 sites with a total of 25 generating plants were considered. Plant studies were performed for Biblis B, the selected PWR reference plant.

Within the uncertainties, the results of the German and American studies agree, the German study confirming that accident-caused risks from nuclear power plants are relatively small.

PREFACE

The responsible use of nuclear energy requires an unprejudiced and factual discussion of its risks. The attempt to systematically analyze and present these safety problems through new methods was first undertaken in the United States. The results of this research are published in the Reactor Safety Study (WASH-1400), better known as the "Rasmussen Study."

In the Federal Republic of Germany (FRG) the need to obtain comparable data on the risks of utilizing nuclear energy arose when it became evident that American results could not be applied "as is" to German reactors. Because of differences in reactor technology and siting, the need for our own study became apparent.

I therefore issued a contract for a German risk study analogous to the Reactor Safety Study to determine the total risks due to accidents caused by operation of nuclear power plants in the FRG. Another objective was to formulate an attitude on the accuracy of such studies and to find starting points for their further refinement through extensive supplementation of the Reactor Safety Study.

One important outcome is the agreement within error limits between the results of the German and American studies. The statement in the Reactor Safety Study that accident-caused risks from nuclear power plants are relatively small--particularly when they can be compared to other everyday risks--was confirmed.

Another significant result was the indication that latent deaths from reactor accidents can be distributed over a large area. With respect to German nuclear power plants, half of this area lies beyond our borders. Conversely, this also means that we will be affected by the consequences of accidents in neighboring countries, clearly illustrating that efforts to prevent accidents and to limit subsequent damage require an international agreement. This point of view is the stated position of the federal government, which will continue to press for an international discussion on questions of reactor safety.

This study forms a basis for scientific discussions not affected by nonspecific concepts like "hypothetical" or "unbelievable"; indicates which questions should receive preference in the framework of reactor safety research, and serves as the basis to discuss the potential contribution of refined engineered safety features and other measures to reduce risk. The error limit is still very large; therefore, careful use of the absolute figures seems appropriate.

When the contract for the German Risk Study was issued, there was already a lively discussion about various weak points in the Reactor Safety Study. Additional criticisms were later addressed in the Risk Assessment Review Group (Lewis) Report.

In light of this criticism, a decision was made to divide the German Risk Study into two sections. In the present Phase A, comparison with the Reactor Safety Study is emphasized with consideration of existing differences in system technology and location. Subsequently, with regard to the Reactor Safety Study and additional weaknesses found in Phase A, methods and models will be refined and then incorporated, along with new information on safety research, into an improved Phase B.

For Phase B, I desire the broadest collaboration of different qualified groups, even those that are skeptical about nuclear energy. I am convinced that an extensive discussion of this difficult matter is needed. All voices must be heard in order to validate the results in every regard.

Therefore, I request all interested parties to critically analyze the results of this study and to send me comments and notes that might be useful in Phase B.

This study uses the example of nuclear power plants to describe the risks--including the quite remote potential for injury--caused by application of modern technology. There is no doubt that other areas of technology pose considerable risks and that delayed injury and effects over large areas must be expected. Because of our rudimentary level of knowledge, a comparison of potential, long-term injury is hardly possible. However, it must be assumed, for instance, that pollutant emissions from the utilization of fossil fuels contribute significantly to the cancer rate and also cause long-term environmental changes.

In order to better understand the hazards of our modern industrial society, the risk analysis with its potential for detection, limitation, and minimization of harmful influences, can be a great help--thereby improving our quality of life. I therefore favor the refinement and broad application of risk studies, and I hope that from an intense discussion of the results of this study, strong impetus for corresponding analyses in other areas will arise.

Volker Hauff
Federal Minister for Research and Technology

CONTENTS

Section 1

OBJECTIVE AND ORGANIZATION OF THE STUDY

In the spring of 1976 the Federal Minister for Research and Technology issued a contract for a risk study for a pressurized water reactor (PWR) nuclear power plant.

It is the objective of this study to determine the risk caused by accidents in nuclear power plants licensed and sited under German conditions. The investigations needed for this were to be undertaken with respect to the American Reactor Safety Study WASH-1400 (1) (a).

Work began on this study in the summer of 1976. It was performed by different institutions and coordinated by the Gesellschaft fur Reaktorsicherheit (GRS, Reactor Safety Company) as the prime contractor.

The study is organized in two phases. The present report contains the results of the first phase of the study.

1.1 INTRODUCTION

Safety considerations have always played an important role in nuclear engineering. Thus, nuclear power plants have to meet comprehensive safety requirements with regard to design, construction, and operation. 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 (2) and in additional detailed guidelines and regulations [e.g., (3,4)].

Parallel with the development of the concept of technical safety, considerations at an early date had led to the estimation of effects of severe accidents in nuclear power plants. The first and best known study of this type is the Brookhaven Study WASH-740 (5) prepared under contract to the United States Atomic Energy Commission (USAEC) and published in 1957. However, the limiting influence of safety features designed into nuclear power plants at that time was not included in the study. Probability evaluations--where performed--were very inaccurate. They played a subordinate role in the study.

Studies concerned primarily with the determination of the extent of consequences are unsuitable for an evaluation of risk. They do not include any statement about the actual risk connected with the operation of the nuclear power plant. The risk depends rather on the extent of the designed safety features and their success in interfering in a nuclear power plant to overcome occurring accidents and to limit the consequences of potential accidents. In addition to information on the magnitude of consequences itself, the probability with which designed safety features may fail and severe reactor accidents occur should be of primary importance.

Nuclear power plants have been in operation in various countries for more than 25 years. During this time no one has been killed by radioactive release from a nuclear power plant nor has there been any demonstrable health injury. But it is not possible to derive nuclear risk values from available experiences of past accidents. The risk values must therefore be determined by analytic means.

Since the mid-1960s--especially in England (6)--suggestions have been under discussion for including the frequency, in addition to the extent, of injury as a criterion for evaluation of safety. Such suggestions could only be discussed as concepts at that time. The status of methodology and available experience was insufficient to implement a realistic determination of risk values. In subsequent years the application of probabilistic methods--not only to nuclear technology--made significant progress. The primary objective was to evaluate system reliability and to improve it as necessary. The usual deterministic criteria of safety feature design were to be supplemented thereby. The reliability analysis developed into an important aid in the evaluation on safety. It therefore proved to be a decisive prerequisite for the implementation of risk studies.

The American Reactor Safety Study (WASH-1400) is the first extensive study in which probabilistic methods were used to determine the risk posed by accidents in nuclear power plants. The study was published in October 1975 after about three years work (1). Thus, the first attempt was made to quantify in detail the risks of a complex technology. The procedure of theoretical determination of risk has not previously been performed in any other technical discipline on this scale.

1.2 OBJECTIVE AND ORGANIZATION OF THE STUDY

After publication of the American Reactor Safety Study (WASH-1400), the question arose of how much its results could be directly applied to German conditions.

Even though the same light water reactor model used in the USA is also found in the FRG for the commercial generation of electricity, there is a series of differences that negate a direct application of American results to German conditions. Two of the most important of these are:

1. The reference plant used in the American study has a different technical design in several important points from the German systems. This pertains primarily to the design and operation of several important safety features.

2. The siting of power plants in the FRG differs considerably from that in the USA. For instance, the average population density in the FRG is about 10 times greater than in the USA. Even in the close vicinity of nuclear power plants, the average population density in the FRG is about three times greater than around American reactor sites.

In order to directly evaluate the special German circumstances, the influences of siting, and the differences in plant design, a domestic study was needed--regardless of the results of the American study.

The Federal Minister for Research and Technology, therefore, issued a contract in the spring of 1976 for a German risk study. In accordance with the contract, the following objectives and assumptions were established:

Like the American study, the collective risk connected with potential accidents in nuclear power plants is determined.

In order to be able to directly evaluate differences in plant technology and siting locations, the German study permits comparisons with the American safety study. The study was therefore divided into two working phases (Phase A and Phase B). In accordance with the objective of the contract, the basic assumptions and methods of the American study were generally assumed for the first phase of our study (Phase A). For the second phase (Phase B), which is planned primarily to intensify study of individual problem areas, further methodological developments shall be employed and the current status of safety research considered.

The plant studies are performed for a selected, representative PWR nuclear power plant. As a reference plant, we selected the Biblis B nuclear power plant. This plant has a typical German-designed PWR (manufacturer: Kraftwerk Union AG) with a thermal power output of 3750 MW. The plant began operation in the spring of 1976.

For the determination of risk, all sites in the FRG where there were nuclear power plants with light water reactors having an electrical power output of at least 600 MW in operation or under construction on July 1, 1977, or for which a request for approval had been received on the above date, were included. Thus, 19 sites with a total of 25 generating plants were considered in the study.

The current report gives an overview of the investigations and results achieved in the first phase of the study. The report is organized into the following Sections:

Section 1: Objective and Organization of Study

Section 2: Principles of Risk Assessment. This Section contains an introduction to the basic concepts and methods of hazard investigation. The direct parameters, probability, and magnitude of consequences and their interrelationship are discussed.

Section 3: The Nuclear Power Plant. This Section provides an overview of the design and operation of the PWR nuclear power plant. It also discusses the principles of the safety design and the designed safety features of a PWR nuclear power plant by using the reference plant as an example.

Section 4: Object and Methods of the Risk Analysis. First, accident sequences are described that are treated in a risk analysis. After an overview of the most important stages of a theoretical risk analysis, the methods of event tree and fault tree analysis are explained.

Section 5: Results of the Event Tree Analysis. This Section contains the results of the system studies correlated with event tree and fault tree analysis. The probability is determined for the studied accidents that a core melt could occur after failure of safety systems.

Section 6: Release of Fission Products. The models and studies are presented that describe processes in a core melt accident down to a potential release of fission products to the environment.

Section 7: Consequence Model. The Section provides an overview of the principles of the consequence model, the atmospheric dispersion, the dose model, the protective action and countermeasures considered in the study, and the model to determine health effects.

Section 8: Results. The Section contains the results of the consequence calculations and the risk presented by the different types of injury considered in the study. In addition, the methods and calculations that were used to estimate the uncertainties in the risk results obtained are explained.

Section 9: Conclusions.

Section 10: TMI. In Section 10 of this report, the accident at Three Mile Island is discussed in connection with the investigation performed in this study.

As supplements to the current report, a series of appendixes will be provided. In these volumes, the individual investigations performed for the study will be documented in detail. Thus the interested reader will be able to procure and evaluate the results of the investigations in detail.

The following appendixes will be provided:

- F1. Event Tree Analysis
- F2. Reliability Analysis
- F3. Reliability Data and Operating Experiences
- F4. External Events (including plant-internal fires)
- F5. Study of Core Melt Accidents
- F6. Determination of Fission Product Release
- F7. Results of Plant System Studies
- F8. Consequence Calculations and Risk Results

1.3 LIMITATIONS OF THE STUDY

When evaluating the results, it is important to remember that the study is subject to a series of limitations. These are caused by the limits of technical knowledge and by available methods; they also arise from the objectives of the contract.

The objective of the study is to determine the risk, under German siting conditions, caused by accidents. Since it is not possible to implement a separate risk study for each plant, the plant system studies were performed using Biblis B as a model. This means that in the study, this plant is considered to be representative for all other nuclear power plants in the FRG. This assumption is naturally not correct in all details of technical design. But it is justified because all plants are evaluated in the nuclear licensing procedures by the same safety criteria and quality requirements.

In addition, the plant system analysis is, strictly speaking, not valid for the reference plant. The design characteristics and safety features of the Biblis B model plant were taken as a basis, but numerous data on physical models and verifications from studies on other plants were used. For example, the data needed for the reliability analyses are obtained from statistical evaluations of operating experiences. These come from experiences with other power plants and from other technical spheres. Previous considerations show the model character of the study; this is even more pronounced in the study of accident sequences leading to a release of fission products. We are dealing primarily with models that will describe core melts, fission product release, and dispersion and biological effects of radiation. The absence of detailed knowledge is made up for here by simplifying assumptions, generally of a conservative nature.

For these reasons, the results of this study are affected by considerable uncertainty that will only permit an assessment of the order of magnitude of the risk.

The plant system studies are based on determinations made, if possible, during the licensing procedure. For instance, in the accident analysis for the efficiency requirements on the safety systems, the minimum requirements established by the nuclear licensing procedures were used. The results of calculations performed in this regard were used in the study without checking them in detail.

It was not the objective of the study to examine all potential influences that could contribute to the risk of nuclear power plants. For instance, in this study the risk caused by accidents, but not the risks originating from the continuing operation of nuclear power plants, was determined. Risks due to acts of war and sabotage were not included.

1.4 IMPORTANT INDIVIDUAL ASPECTS

The studies in Phase A of this report could not be fully analogous to the procedure in WASH-1400. Significant deviations are listed below:

Differences in plant technology between the American and German reference plant (Surry 1 and Biblis B) led to different emphasis--in comparison to WASH-1400--especially with regard to reliability studies. For instance, initial intermediate results of the event tree and fault tree analyses indicated that studies more detailed than originally planned were needed to evaluate transient accidents.

The consequence model used for the German study corresponds in principle to that of the American study. But in several individual points, it was necessary to adapt the model to German siting conditions. In particular, in Phase A of the study our own model of protective and countermeasures was developed that considers existing official recommendations (7).

Methods and objectives of risk investigations have been under increasing discussion in recent years. Several important points for the German study emerged from this discussion, especially on WASH-1400. For instance, in contrast to WASH-1400, a linear dose-risk relationship was used with the risk factors given in ICRP 26 (8) to determine late health effects.

Other differences from the American study will be discussed in the appropriate Sections of this report.

In the course of the work,the current status of the studies was reported several times (9,10). The intermediate results discussed at that time indicated that significant contributions to risk could be due to individual weak points resulting from the interconnection of instrumentation and control, power supply, and process techology. Such influences can often be eliminated by slight changes, e.g., in instrumentation and control, or by expanding the scope of maintenance. Changes undertaken in the reference plant during the course of the study are generally included here. Changes up to the year 1978 are taken into account.

1.5 ORGANIZATION OF THE STUDY

Work began on this study in the summer of 1976. Scientific leadership of the study was with Prof. Dr. A. Birkhofer, Director of the Gesellschaft fur Reaktorsicherheit (GRS) mbH. The work was performed jointly by several institutions, the GRS being the prime contractor. Primary participants in the studies were:

- Gesellschaft fur Reaktorsicherheit (GRS): plant system studies: event tree analysis, fault tree analyses for safety systems, description of core melt accidents, determination of fission product release.

- Kernforschungszentrum, Karlsruhe (KfK): preparation of the consequence sequence model (dispersion and consequence determination), implementation of consequence calculations.

- Gesellschaft fur Strahlen-und Umweltforschung (GSF): biological effects of radiation and types of radiation injury.

A number of other institutions were included in the treatment of different subtasks and provided advice on individual problems:

- Institut fur Kerntechnik, Technical University of Berlin: determination of reliability data.

- Lehrstuhl fur Elektrische Meßtechnik, Technical University of Munich: determination of reliability data for electronic components.

- TUV-Arbeitsgemeinschaft Kerntechnik West: evaluation of the VdTUV failure statistics for conventional pressure vessels and boilers.

- State Material Testing Center, University of Stuttgart: assessment of the reactor pressure vessel.

- TUV Norddeutschland: evaluation of operating experiences.

- Ingenieur F. Mayinger u. Co., Barsinghausen: studies on steam explosion.

- Konig & Heunisch, Consulting Engineers, Frankfurt: assessment of building structures to determine earthquake effects.

- TUV Rheinland, Institut fur Unfallforschung: work on the model of protective and countermeasures.

- Bonnenberg & Drescher Ingenieurgesellschaft mbH (B+D) Aldenhoven: supply of population data.

At this point, we wish to thank all institutions and study groups who participated in the study and in this report for their contributions and cooperation. Our thanks is also due to others who supported the work and gave valuable advice. Our special thanks is due the United States Nuclear Regulatory Commission, which made available various computer programs used in WASH-1400. Finally, we wish to thank the authors of the American Reactor Safety Study, who provided much valuable information through consultations.

FOOTNOTES

(a) Because of the frequent references to the American Reactor Safety Study WASH-1400, this study is cited in detail only in Section 1. In all following Sections, it is abbreviated in the text as WASH-1400.

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Section 2

PRINCIPLES OF RISK ASSESSMENT

2.1 INTRODUCTION

In this study, risk is expressed in numbers. Only in this manner a comparison of different risks is possible. But what is risk? How do we obtain a numerical value for it? What does it mean?

These questions will be answered in the following four sections of the chapter. The last two sections treat the composition of numerical values for risks and the significance of very small probabilities.

2.2 WHAT IS RISK?

Anyone who in everyday conversation uses the word "risk" is thinking of "hazard" or "danger," that is, the possibilities of suffering injury without really knowing whether these chances are real.

Anyone driving a car, for instance, could be involved in a rear-end accident, a head-on collision, or a multiple-car pile-up, etc. Each of these possibilities describes a potential event. Will one of these events occur on the next trip?

Anyone whose speculates in the stockmarket is taking a "risk." He exposes himself to the potential of losing money by a drop in exchange rates. With pharmaceuticals, there is the risk of side-effects. Avoidance of pharmaceuticals results in the risk of continued illness or even a deterioration of the illness.

Everyday usage also deals with the magnitude of risk. For instance, the risk of being killed by a meteor is extremely small, but the risk of dying of influenza in the winter is considered to be high because the event is very probable.

Someone who participates in a project with 100 DM takes a lower level of risk than someone who participates at a level of 10,000 DM because the potential damage to the former is much lower.

Insurance agencies are concerned with more accurate definitions of risk. They calculate the risk, for instance, of fire damage or burglary, and from this obtain a premium to be paid for protection against such risks.

2.3 WHICH RISKS SHALL BE ASSESSED?

At the beginning of a risk assessment, it must be clearly described which of the various possible risks is to be expressed in numbers. The following questions can help here:

- What risks shall be assessed?

  For every person and social group (family, community, nation, etc.) there are quite specific risks. For an individual person, we speak of individual risks; otherwise the term collective risk is used (also, societal, public, or group risks).

- Which risk to the person or the public is to be assessed?

  This can be a quite specific risk, e.g., death due to lightning strikes or the total risks attributed to a more narrowly defined activity like "working in a household," "truck driving," or "operation of nuclear power plants."

- Over what time period is the risk to be assessed?

  The person or the group is exposed to the risks as long as the potential to suffer harm exists. Many possibilities exist day in and day out, but others last only for certain days or times of the year or for clearly defined activities (e.g., during take-off or landing of an aircraft). Frequently, the numerical value of a risk is determined for an exposure time of one calendar year, and this is called the "annual risk".

2.4 HOW IS A NUMERICAL VALUE FOR RISK COMPOSED?

Every potential to suffer harm represents a risk, provided it is uncertain whether the potential will become a reality. Harm and uncertainty, therefore, are components of risk. Thus, the numerical value of risk must be composed of a numerical value for harm (injury) and one for uncertainty. The numerical value of risk will be called the "risk coefficient" below.

2.4.1 The Element Consequence

In order to obtain a numerical value for consequence, the injury must be measurable, that is, expressed as a unit of measure. Usually, dimensionless units, (e.g., number of affected persons) are used in connection with individually different value weights. For instance, the death of associates generally weighs more heavily than the death of strangers. Since these weights as a rule are unknown and frequently cannot be expressed in numbers, the consequence must be included in the overall risk as unweighted. For a catastrophe that takes human lives, the consequence is measured by the number of deaths. If there is property damage due to fire in a factory, for instance, then the consequence is measured in monetary units. Therefore, consequences cannot be compared "as is" because of the different units. Consequently, for each type of consequence--which has to be measured in its own unit--a special risk factor must be determined.

Examples of consequence units are:

Type of injury	Unit
Human deaths	Number of deaths
Health effects	Number of affected persons (e.g., injured persons)
Regions rendered uninhabitable	Surface area
Materials damage (replaceable)	Monetary units

Once we know what units are to be used to express the extent of consequences from an event, then the number of these units remain to be determined . Description of the event primarily contains details needed to distinguish the event from other occurrences not included in the risk. On the basis of the other details, many other partial results can be distinguished, all of which satisfy the "rough" description of the event, but which as a rule cause consequences of a different type and extent. For instance, the rough description of an event "work-place accident" is satisfied by many partial events like "electrocution during assembly," "collapse of scaffolding," and "death due to dust explosion." Each of these contributes its part to the risk coefficient of the event "work-place accident" as expressed by the type of consequence: "loss of human life," "health hazard," "material damage," etc. With respect to an individual for the former type of consequence, we have only the alternatives "the person is affected" or "the person is not affected;" i.e., using the dimensionless unit of measure, we have injury to 1 (one person) or 0. In the case of a group, many different numerical values are possible for this type of injury since the event can take several lives of the group.

2.4.2 The Element of Uncertainty

Uncertainty is expressed in the question: "What is the probability of occurrence of the event?" We speak of the probability as the uncertainty expressed in numbers. In mathematics, the concept of probability has been precisely defined [see for instance (1)].

An impossible event has the probability zero, and an event with absolute certainty of occurrence has the probability unity. That which is possible but uncertain has its probability expressed as a number between 0 and 1. The more probable the event, the closer is this number to 1.

Of the 733,400 deaths in the FRG in 1976, 14,616 had pneumonia as the cause of death (2)--this is nearly as frequent as death due to vehicle accidents.

Let us assume that each individual death in 1976 was noted on a special card and all the cards were placed in a file cabinet. Now, someone is told to remove one of the cards on which the cause of death is listed as "pneumonia." If he does not know the filing system of the cabinet; that is, if he must make a random selection, it is uncertain whether he will withdraw the proper card. How likely (or probable) is it?

There are 14,616 of these special cards among a total of 733,400 cards in the file cabinet. Therefore, the chance of withdrawing a specific card is 14,616 in 733,400. The probability of withdrawing one of the desired cards upon random access to the file cabinet is therefore: (14,616)/(733,400) approximately equals 2/100 = 0.02.

This probability is calculated from the number of different, equally justified possibilities of drawing one of the desired cards, divided by the number of different equally probable possibilities of withdrawing any card from the file cabinet.

If we had a file cabinet containing one card for each person in the FRG alive at the beginning of 1976, then this file would contain 61,645,000 cards in accordance with the number of inhabitants.

The probability of drawing a card randomly from the file cabinet of someone who died of pneumonia during 1976 is: (14,616)/(61,645,000) approximately equal to 24/100,000 = 0.00024.

If we recalculate this probability each year and if it turns out that it does not change significantly from one year to the next, then it can be used as an estimated value for the coming year. In fact, the probability calculated as above for several years before 1976 has been quite constant (for 1974 approximately equals 0.00021, but for 1960, for instance, it was still approximately 0.00041). Therefore at the beginning of 1977 we could justifiably estimate that the probability of any person in the FRG dying during the course of 1977 of a pneumonia is 0.00024. This figure would then be an estimated value for the probability of each inhabitant dying of a pneumonia during the current year if each inhabitant had the same chances of contracting a fatal pneumonia, this is the same probability of drawing any card from the file cabinet. In reality, we are seldom dealing with really equal chances, so that 0.00024 is only an estimated value of the average individual probability, averaged over all 61,645,000 persons. The rough division by age groups shown in Table 2-1 illustrates how much individual probability can differ from the average. The probabilities given there are not averaged over the entire population, but only by the age of more narrowly defined cohorts. Accordingly, a fatal contraction of a pneumonia during old age is about 50 times more probable than in youth.

Table 2-1. Estimated value of average individual probability for the occurrence of "fatal illness from pneumonia in the previous year, with an age between X_1 and X_2 years"

Age between X_1 and X_2 years		Persons on Dec. 31, 1975 (population)	Cases of death in 1976 (cause of death, pneumonia	Average individual probability per year %
0	15	13,084,000	338	0.000026
15	45	26,042,400	381	0.000015
45	65	13,513,400	1,108	0.000082
65	--	9,004,700	12,789	0.001420

2.4.3 The Interrelationship of Consequence and Probability in the Risk Value

How can we calculate a quantity which can be used as a measure for risk from the probability of occurrence and the magnitude of the consequence?

2.4.3.1 Individual Risks. As an example, we again use the event "fatal illness due to pneumonia." In the case of an individual, the magnitude of injury is 1 for the injury type, "loss of human life;" the average individual probability for this in the FRG can be estimated for the coming year as 0.00024.

For each person there are various possibilities for the course of the coming year. Only one of these will actually occur and it is uncertain which one. The probability states that of 100,000 different, equally probable possibilities for the course of the coming year, 24 possibilities will result in a fatal pneumonia; that is, there are 24 chances that a consequence magnitude of 1 will result from this event. Of the many different possibilities, exactly one will occur. Therefore, the injury, related to the individual possibility for the sequence of the coming year, is called risk and we thus obtain: (1 X 24)/100,000 = 1 X 0.00024 = 0.00024 as the individual risk coefficient for the coming year. In other words, this is an estimated value of the average individual risk of a "lethal pneumonia" in the FRG in the coming year.

By using the same line of thinking, the risk coefficient R can be calculated for each event as the sum of the products from the potential consequences and their probabilities:

$$R = y_1 \times w(y_1) + y_2 \times w(y_2) + y_3 \times w(y_3) + \ldots \qquad (2.1)$$

provided the annual consequence magnitudes can assume only a finite number of different values. Here, y_1, y_2, . . . mean the different possible magnitudes of consequence per year and $w(y_1)$ is the probability that the event will cause a consequence of magnitude y_1 during the year (a).

The numerical value R is an estimated value of the risk if the consequences or probabilities are estimated values. In this case it always agrees with the estimated value of the average individual probability of the event if it can occur at most once per year and cause a consequence of magnitude 1 per event.

Table 2-2 shows how the average individual risk depends on the amount of detail used in describing the event. If we expand it as shown in the table by adding the detail "age of persons in the coming year," then definite differences in risk are seen. The individual risk is therefore not equal for every person for these event examples. Not only age, but numerous other, often inaccurate or unknown factors like health history, state of health, living conditions, etc., can have an effect. Therefore, in many events only the average individual risk (averaged over a larger number of persons) can be estimated in a meaningful manner. In the example, we are interested only in the average individual risk to inhabitants of the FRG; therefore it is sufficient to observe that of the 61,645,000 inhabitants in 1976, 14,616 died of pneumonia. From this it can be concluded that the average individual chance in 1976 must have been equal to 14,616 in 61,645,000 or, more simply stated, 24 in 100,000; therefore the average individual risk equals 0.00024. The other average individual risks shown in Table 2-2 were estimated similarly.

How is individual risk assessed if the event can occur to the individual several times per year? It is possible to contract influenza twice each year. Three and more attacks in one year would be so rare that the average individual probability for purposes of this example can be safely estimated as zero.

The probabilities in Table 2-3 indicate that the chances of getting through the present year without the flu are five in ten. The chances of contracting the flu only once are four in ten, and for contracting it twice, the chances are one in ten.

This means that on the average, of ten different, equally probable potentials for the course of the coming year, five will proceed without an influenza attack, four will proceed with one influenza attack, and one will have two attacks. Based on the individual possibility for the course of the coming year, because only one course will actually occur, we obtain the value: $(0 \times 5 + 1 \times 4 + 2 \times 1)/(10) = 0.6$.

We call this the average expected individual frequency of influenza attacks per year, that is, the expected value. This means that in the course of ten years, for instance, one can expect six attacks of influenza if the probability information in Table 2-3 is correct.

Table 2-2. Estimated values of various average individual risks per year in the Federal Republic of Germany (2)

Occurrence: death by	Cases of death in the Federal Republic 1976 (2)	Estimated value of average individual risk per year	Estimated value of average individual risk per year	Estimated value of average individual risk per year with an age between:			
				0 & 15 years	15 & 45 years	45 & 65 years	65 & -- years
Heart diseases	203,586	330 in 100,000	0.00330	0.00001	0.00012	0.00217	0.01899
Malignant growth (cancer)	152,590	247 in 100,000	0.00247	0.00005	0.00024	0.00282	0.01194
Pneumonia	14,616	24 in 100,000	0.00024	0.00003	0.00001	0.00008	0.00142
High blood pressure	13,360	22 in 100,000	0.00022	x	0.00001	0.00011	0.00129
Highway or traffic accident	14,445	23 in 100,000	0.00023	0.00011	0.00028	0.00018	0.00037
Other accident	17,214	28 in 100,000	0.00028	0.00011	0.00010	0.00016	0.00122
Total	415,811	674 in 100,000	0.00674	0.00031	0.00076	0.00552	0.03523

x = less than 0.00001.

Table 2-3. Example of an event to demonstrate the use of the term "expected frequency per year"

Occurrence "illness, flu"	
Cases of loss (illnesses) per year	Average individual probability (fictitious values)
0	0.5
1	0.4
2	0.1
3 and more	0

If we measure the annual injury due to the event "influenza attack" merely by the number of contractions of the illness, then either the event does not occur or it causes injury of magnitude 1 or 2. The probabilities in Table 2-3 indicate that in ten different equally probable cases for the course of the coming year, four have injury magnitude 1 and one has injury magnitude 2. If we relate the injury to the individual possibility for the course of the coming year, then we again obtain the sum of the products from the possible magnitude of injury per year and the attendant probability of: 1 X 0.4 + 2 X 0.1 = 0.6 as the individual risk per year corresponding to the equation (2.1). Stated differently, this is an estimated value of the average individual risk of "influenza attack" in the coming year. It agrees with average expected individual frequency per year because only injury of magnitude 1 is possible per event. The product of the potential magnitude of injury per event and the expected annual frequency of the event: 1 X 0.6 = 0.6 also gives the risk coefficient.

In general, the risk coefficient R can be calculated for events that can occur several times in a year and whose consequences can assume a finite number of different values, according to the equation:

$$R = x_1 \times h(x_1) + x_2 \times h(x_2) + x_3 \times h(x_3) + \ldots \qquad (2.2)$$

Here, x_1, x_2 . . . are the different possible magnitudes of injury per event and (x_1) is the expected annual frequency of the event of injury magnitude x_1 (b).

If we wish to determine the risk from all events in a more closely specified relationship, like, for instance "vehicle traffic" or "operation of nuclear power plants," then all we have to do is add up their risk coefficients. Strictly speaking, this is only true when the events are mutually exclusive, i.e., the occurrence of one event cannot at the same time indicate the occurrence of the other. So, for instance, events like "fatal fall" and "occupational fatal fall" are not mutually exclusive. The probability of the latter is contained in the probability of the former, i.e., the risk would be over-estimated by addition. The events, "fatal fall during leisure sports" and "occupational fatal fall," are mutually exclusive. Addition of their risk coefficients does not, however, give the risk coefficient for the event "fatal fall," because they are not a complete statement of subevents. There are other possibilities to lose one's life as the result of a fall.

2.4.3.2 Collective Risks. The event "fatal pneumonia" caused 14,616 deaths in the FRG in 1976. Therefore 14,800 is not too bad an estimation for the coming year. Therefore, it is expected that the community, "Federal Republic of Germany," will lose an estimated 14,800 people in the coming year due to this event. We call this number the estimated value of collective risk of the FRG due to the event "fatal pneumonia" in the coming year. The exact number of deaths is unknown.

In principle, the collective risk is equal to the product of the number of exposed individuals in the group and the average individual risk. Frequently, we obtain the estimated value of average individual risk simply by dividing the estimated value of collective risk by the number of exposed individuals in the group (see, for instance, the risk coefficients in Table 2-2).

According to the above discussion, which resulted in equations (2.1) and (2.2) for individual risk, a numerical value can naturally also be calculated with these formulas for the collective risk. If we knew, for instance, the probability that the total highway accidents in the FRG would claim y_1 or y_2 or y_3, etc., lives in the coming year, then we could determine the collective risk, i.e., the expected number of deaths in the coming year, according to the equation: $R = y_1 \times w(y_1) + y_2 \times w(y_2) + \ldots$ If, instead, we knew the frequency of traffic accidents in the coming year in the FRG with expected fatalities of x_1 or x_2 or x_3, etc., then we could calculate the same collective risk according to the equation: $R = x_1 \times h(x_1) + x_2 \times h(x_2) + \ldots$

Naturally, hardly anyone can determine collective risk by these formulas because, for instance, the number of traffic fatalities in 1978 is already a useful estimation for the coming year. In Section 2.5 we will discuss events that are rare or have never yet occurred, so that risk coefficients from previous years alone do not permit a satisfactory estimation of risk. In these cases, the risk coefficient can only be determined by formulas of the type (2.1 or 2.2).

2.4.4 Uncertainties in Estimated Values of Probabilities

The term "probability" was introduced in Section 2.4.2 on the basis of a model case; namely we were given a file with exactly X cards as a so-called basic group of which Y had a certain characteristic. Here, we were asked to find the probability of drawing a card with that characteristic. In reality, questions of probability are frequently difficult to answer because:

(a) The magnitude X of the total group is generally unknown.

(b) The fraction Y bearing the desired characteristic is also generally unknown.

(c) From the basic group a "random sample" is taken--and this is not always completely random--i.e., we draw average X cards--to stay with the above example--and count off how often (e.g., average Y times) the desired characteristic is found. The quotient average Y/average X is called the relative frequency of the occurrence of a card with the characteristic in the random sample. It is used as an estimated value for the probability, i.e., for the unknown quotient Y/X.

(d) The interpretation of the card entry can be inaccurate. For instance, it can not always be definitely known whether (in the case of the example) the pneumonia or another complication was the actual cause of death.

Except for the inaccuracy (d) and the possibility that selection is not entirely random, the uncertainties in (a) and (b) can be expressed by a value range (so-called confidence interval) within which there is a certain probability (also called probability range) that the particular probability value will be found.

(e) The file (basic group) comes from the year 1976 but the probability information is to be valid for the coming year (different basic group). There can be significant differences between the two, e.g., significant improvements in medical treatment or deterioration in general living conditions.

Thus, it is apparent that the probability values in practical risk calculations can in general be estimations only. So-called objective estimations are based on random samples from those basic groups for which the probability information will be obtained. However, if random samples from different basic groups and other information contents--about whose suitability there can be a difference of opinion--form the basis of the estimation, then we speak of subjective probability values. They are based on the personal judgement of the estimator that, for instance, the potential differences between the basic group from which the random sample is taken and the basic group for which the probability is to be estimated do not significantly affect the estimated value. For this reason, probability estimations for future time frames are often subjective. In accordance with analogous criteria, we speak of subjective confidence intervals and subjective confidence in conclusions. Subjective estimations can provide meaningful values if the resulting personal judgement is based on the factual experience of the estimator (expert judgement) (3).

2.5 RISK COEFFICIENTS OF EVENTS THAT HAVE SELDOM OR NEVER OCCURRED

The preceeding section described the connection of joining of damage magnitude and probability to the risk coefficient. This is formally expressed in the simple equations (2.1) and (2.2). By using these equations in the case of a finite number of damage magnitudes, the numerical risk value can be calculated for every possible potential of experiencing injury. A prerequisite, however, is that all possible damage magnitudes together with their probability or frequency are known.

When observations of the event are so numerous that they permit an estimation of frequency or probability within equation (2.1) or (2.2), then use of these equations is generally no longer necessary because the annual risk can be estimated directly from the annual observed damage. However, if the event is rare or has never been observed, then the risk coefficient can only be determined by means of equations (2.1) or (2.2). At the same time, in this case there is often no possible meaningful estimation of probability or frequency based on observations of this event. By using simple examples we will show how to handle this situation.

2.5.1 Probability Estimations Based on Observations of Complementary Events

The situation of probability estimation is naturally better when the so-called complementary event has been observed often.

Commercial aircraft annually make several million take-offs and landings. The event "no crash during take-off or landing" is complementary to "crash during take-off or landing." If we assume that at airport X there have been 10,000 commercial take-offs or landings without a single crash; and if we take the first

1,000 flights as a "random sample," then we will find no event "crash during take-off or landing." By using standard statistical methods, the conclusion can be drawn (restriction [e] in Section 2.4.4 should be noted) that the probability per flight for the occurrence of the event, "crash of a commercial flight at airport X," is about 1/1000 = 0.001. With a confidence interval of 95%, it is not above 0.003. The confidence interval therefore contains values between 0 and 0.003. The upper bound is obtained on the basis of the following simple consideration:

> "What must be the minimum value for the crash probability per flight in order that more than zero crashes are observed among 1,000 flights having a chance of at least '95 in 100'? But since no crash was observed in the 1,000 flights, the crash probability is 95% certain to be under this value" (c).

Were we to look at not only the first 1,000, but all 10,000 flights as a "random sample," then as the upper 95% limit of probability of the same event, we would obtain the smaller value of 0.0003.

In the same manner we could estimate upper limits of probability of certain events, like a "core melt," from operating experience with light water reactors. From 500 reactor-years (worldwide experience with light water reactors of the power class "400 MW (e) and more") without the occurrence of the observed event, we conclude (restriction [e] in Section 2.4.4 should be noted) that the probability for its occurrence is less than 0.006 per reactor year with a confidence interval of 95%. The upper limit is determined as if we knew absolutely nothing about reactors except for the fact that no event had occurred in 500 reactor years.

Unfortunately, no additional observations are available--as in the case of the aircraft take-offs and landings--to show that the upper limit chosen was too high because of the relatively few number of observed reactor years. But even if a very small upper limit could be estimated on the basis of additional observations of the complementary event, no statement could be made about the second component of the risk coefficient, namely about the damage caused by the event under certain circumstances.

For these reasons the study describes the considered results in more detail by using detailed knowledge about reactors (see Chapters 3-6). The degree of detail in the description is increased until:

- Partial results can be distinguished by their influence on the magnitude of damage; and

- Sub-events of these partial events can be seen; these sub-events signify the occurrence of the partial event only in connection with other sub-events and their probability can be usefully estimated by observations and additional detailed knowledge.

This procedure, known by the name "event breakdown," is illustrated in the following section.

The concept of "partial event" was introduced in Section 2.4.1. For instance, the event "fall from a scaffold" is a partial event of the event "occupational accident," because other events like "death due to dust explosion," satisfy the description "occupational accident." The partial event "death due to dust explosion" can in turn be broken down into sub-events like:

- "The dust concentration in the air has reached a critical value in one section of the building."
- "Spark formation is occurring in the same section of the building."

If either of the two sub-events occurs independently, then there will be no dust explosion. Only the joint occurrence of both sub-events results in an explosion.

2.5.2 Estimation of Probabilities and Expected Frequencies Based on Detailed Information

When a new structure is erected, this represents in many regards a unique process. Therefore, the probability that the structure will collapse within the planned service life cannot be estimated directly. Only when the event "collapse" can be described in such detail that the different sub-events contributing to its occurrence become visible, will the probability of collapse be calculable from the probabilities of the sub-events.

In Figure 2.1 a partial event of the event "collapse under load" (d) is described as a function of only two sub-events:

(a) "The load acting on the structure has a certain value that is denoted here as s."

(b) "The strength of the structure called here r is (in a suitable unit) less than or equal to s" (Figure 2-1).

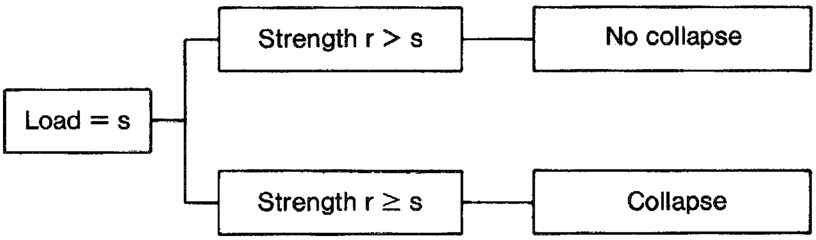

Figure 2-1. Example of the description of a partial event for event, "collapse from the influence of load," of two subevents (load = 2, stability r ≤ s)

The probability of the partial event in Figure 2-1 is equal to the probability that the load will assume the value "s" and, at the same time, that the strength has a value less than or equal to "s". According to the rules of probability calculations, it is the product of the probability of the sub-event, "load = s," and the probability of "strength r less than or equal to s" (e), provided load and strength values are independent.

From the relative frequencies of measured snow loading and wind velocities, etc., that is, from random samples and their conversion into loads, we obtain a probability diagram of the load that can look something like Figure 2-2.

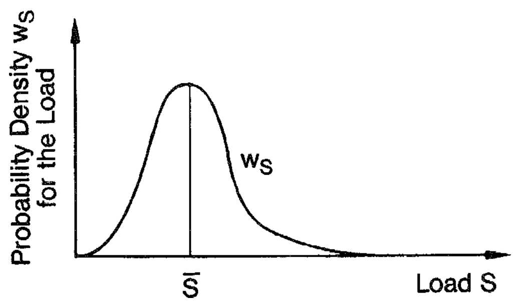

Figure 2-2. Probability density of the load

As a check of the quality of the construction materials, etc., that is, from random samples with subsequent conversion into strength values, we obtain a similar probability diagram for strength. If both load and strength are plotted in a coordinate system, then we obtain Figure 2-3 for the example.

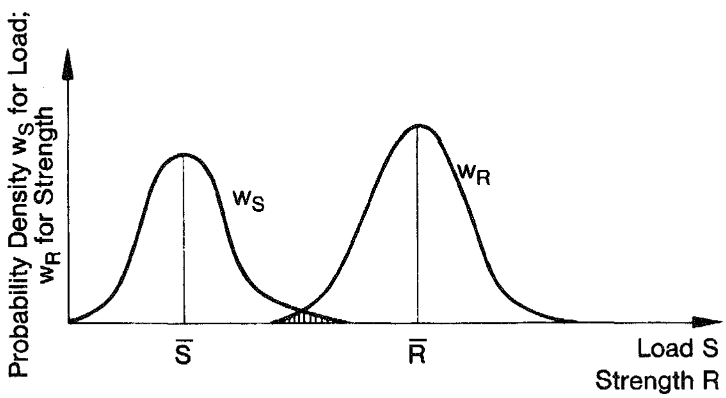

Figure 2-3. Probability density functions for load and strength

The hatchmarked region, where the two density curves overlap, is critical. Here, the strength can be smaller or equal to the load which, according to the assumption, will lead to collapse. Therefore, it is desirable to keep this critical load as small as possible. In civil engineering this is achieved by a "safety interval" of the two average values S and R, the greater this interval, the smaller is the critical region of overlap (4).

The quality of the probability estimation for the event "collapse under load" naturally depends decisively on the accuracy with which the density functions and, in particular, the overlapping end sections of the functions were determined. Even the assumption that load and strength assume independent values affects the calculated probability.

Using the same principle--breakdown into partial events and sub-events and summation over the partial events--the study determines the expected frequency of the event "core melt." The required reactor-specific terminology will not be introduced until the next chapters. Nevertheless, in order to be able to illustrate probability or frequency estimation on the basis of a detailed breakdown of event, we will not use the risks from the "operation of a more carefully specified pressure water reactor"; instead, we will use the example of the "occupation of a more closely specified building." In this regard the first question to be answered is, "What potentials for injury to the individual person or a collective of persons result from occupation of a more carefully specified building?"

The answer would have to contain the following events: (a) collapse of the building, and (b) burning of the building.

The former was broken down into a great number of partial events; each of these was, in turn, broken down into two sub-events. By using the event, "burning of the building," a more intensified breakdown of the event--as performed in principle in the study--will be illustrated. In order to break down the event into partial and sub-events, that is, to increase the resolution of the description, we must further ask, "What events can initiate a fire in the building?"

Among these are: (a) handling of open fire, and (b) overloading of electric circuits; and furthermore, how can a "fire in the building" result from "overloading of electrical circuits?"

As an aid in answering these additional questions, we use event trees (see Figure 2-4) from which each event sequence can be determined, proceeding from the so-called initiating event and moving along other secondary events down to the event being considered.

In Figure 2-4, for instance, the event sequence $T = D_1 \rightarrow A_1 \rightarrow A_2 \rightarrow A_3 \ldots$ leads to "fire in the building." It is a partial event of the event "fire in the building" since other sequences can also result in a fire. For the expected annual frequency of event sequence T, we obtain the following equation according to the rules of probability calculation:

$$h(T) = h(D_1) \times \tilde{w}(A_1) \times \tilde{w}(A_2) \times \tilde{w}(A_3) \times \ldots \quad (2.3)$$

provided the expected annual frequency $h(D_1)$ and the probabilities $\tilde{w}(A_1)$, average $\tilde{w}(A_2)$,. . . can be usefully estimated. In general we are dealing with conditional probabilities, i.e., they must be valid under the conditions of the preceding sub-events. For instance, the fact that we are dealing with an

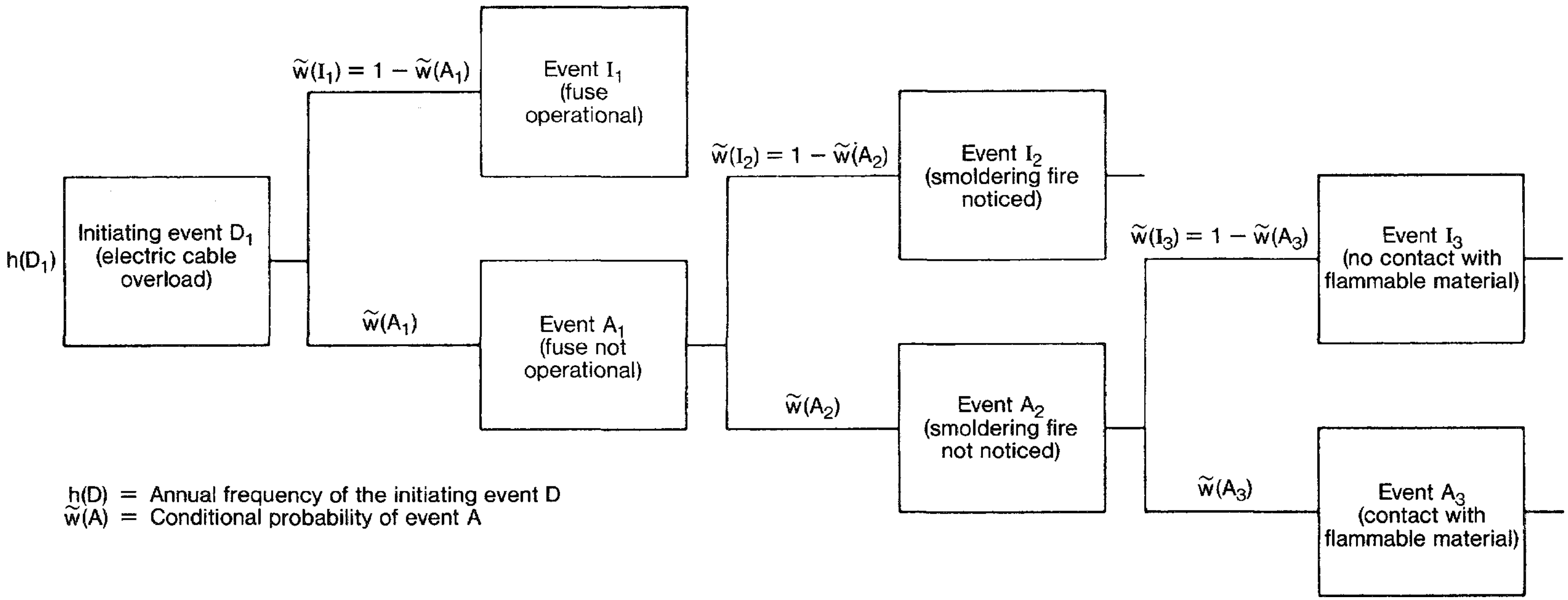

Figure 2-4. Schematic example of an event tree

electrical fire can affect the probability that the fire will not be noticed. This means that an interrelationship exists between the events D_1 and A_2 so that in 2.3 we cannot in general use the probability $w(A_2)$ for the failure to notice the fire, but we can use the conditional probability average $\tilde{w}(A_2) = w(A_2/D_1A_1)$. Probabilities are always positive and never greater than one, and therefore very small frequencies h(T) may result as the product in equation 2.3.

The probabilities of the sub-events are generally determined by means of so-called fault trees. Figure 2.5 shows a schematic example where only two fault combinations lead to the event A_1, namely the joint occurrence of failure F_1 and F_2 or failure F_3. For the probability $w(A_1)$, provided the two failure combinations are mutually exclusive, we therefore obtain:

$$w(A_1) = w(F_1) \times w(F_2/F_1) + w(F_3)$$

$$\tilde{w}(A_1) = w(A_1/D_1) = (w[A_1/D_1/w(A_1]) \times (w[F_1] \times w[F_2/F_1] + w[F_3]) \qquad (2.4)$$

and thus, for the expected annual frequency of the event sequence $T = D_1 \rightarrow A_1 \rightarrow A_2$ ...from (2.3):

$$h(T) = h(D_1) \times c_1 \times (w[F_1] \times w[F_2/F_3] + w[F_1]) \times c_2 \qquad (2.5)$$

where $c_1 = w(A_1/D_1)/w(A_1)$.

We can determine h(T) by means of (2.5) if the frequency $h(D_1)$, the "coefficents of dependence" c_1, c_2, . . . and the probability $w(F_1)$, $w(F_2/F_1)$, . . . can be meaningfully estimated.

If 'm' mutually exclusively relevant partial events contribute to the event "fire in the building," then for its frequency h we obtain:

$$h = h(T_1) + h(T_2) +. . .+ h(T_m) \qquad (2.6)$$

where $h(T_1)$, $h(T_2)$, . . . are determined by equations of the type (2.5).

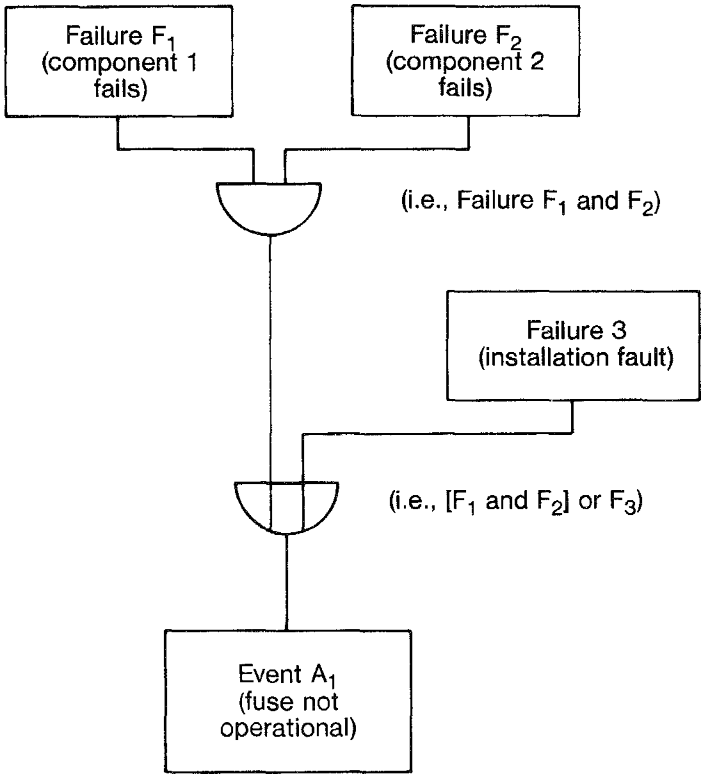

Figure 2-5. Schematic example of a fault tree

2.5.3 Risk Assessment Based on Detailed Information

Figure 2-6 sketches the outline of a risk assessment based on detailed knowledge proceeding from a description of an event down to the determination of damage and risk coefficient.

The outline can be divided into four regions (5):

(a) the region "event sequences" must describe in detail all events contributing to the risk (e.g., "fire in the building"). The description takes place on the basis of event trees and so-called fault trees. The different event sequences (partial events) are characterized by:

--their expected frequency h(T) (e.g., estimated according to [2.5]), and

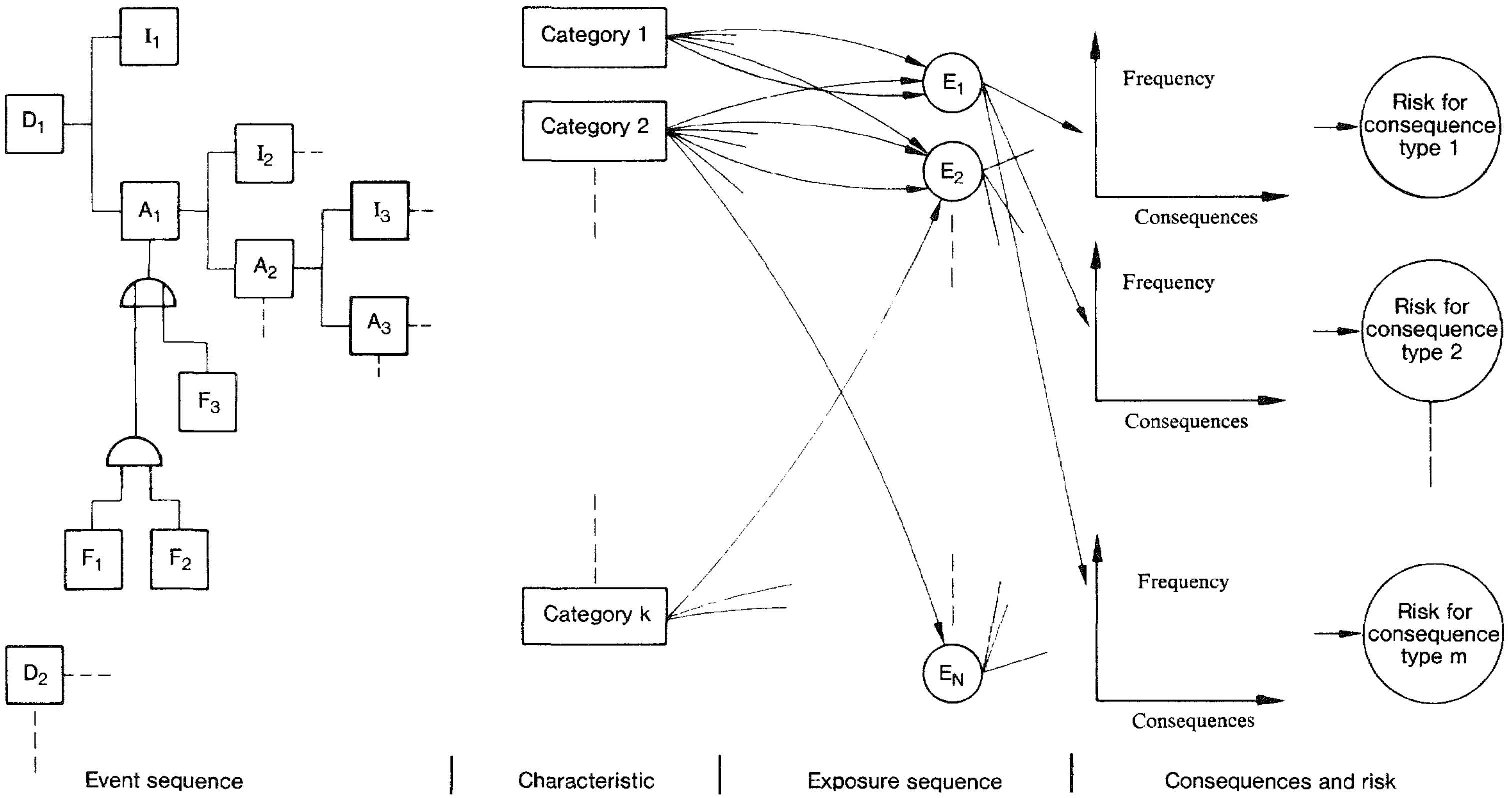

Figure 2-6. Schematic diagram of risk determination for N exposed persons (or groups) based on detailed knowledge of the event and exposure sequences. (Regarding subevent groups D, I, A and failures F, see for instance Figures 2-4 and 2-5)

--information on event characteristics (details to characterize the fire, e.g., origin, location of source of fire, burning substances, etc.).

(b) The region "characteristics" must contain a description of the subsequent results of the different event sequences. This description uses the components of the event characteristics that are important to an assessment of damage (e.g., intensity of a potential explosion, level of heat and smoke generation, etc.).

Depending on the value range of these components and their significance to an assessment of damage, the results are divided for reasons of simplification into classes or categories K_1, K_2, . . .

The categories are designated by:

--representative values of the components of the event characteristic as needed to assess the damage, and

--the sums of the expected frequencies of the mutually exclusive event sequences from (a), which were assigned to the particular category because of its characteristic. If, for instance, category K_1 contains only event sequences T_1, T_2 and T_5, then for its frequency $h(K_1)$ we have: $h(K_1) = h(T_1)+h(T_2)+h(T_5)$ where $h(T_1)$, $H(T_2)$, $h(T_5)$ are determined according to equations of the type (2.5).

(c) The region "exposure sequences" must describe all processes (exposure sequences) according to time, location, intensity, and probability, through which the event characteristic could have a detrimental effect on the particular person (or groups of persons).

The description must contain, for instance:

--the propagation of harmful components of the event characteristic (e.g., smoke in the case of "fire in the building") in accordance with prevailing local conditions M;

--the local distribution of exposed persons (the risk to exposed persons) B;

--protective actions and countermeasures (evacuation, fire extinguishing, etc.) G.

In addition, probability estimations are needed for the various possible values of the components of M, B, and G. In practical calculations of risk, the considered values and the estimations of attendant probabilities often come from "random samples." The quantity of exposure sequences is thus approximately represented by a finite number of specific values sets (m, b, g). The probability for an exposure sequence similar to the specific local conditions m, the specific exposure distribution b, and the specific protective action and countermeasures g thus becomes: $w(m) \times w(b/m) \times w(g/mb)$.

(d) The region "damage and risk" must describe the relations between the intensity of the damaging effect and all damages resulting from it. Therefore, for each value set v = (category k, local conditions m, distribution of exposed persons b, emergency countermeasures g), it must provide an estimated value x(v, a) of the consequence magnitude for each type of damage. In practice, we are often dealing with an estimated value of the magnitude of damage that is expected under the conditions of the value set v. The averaging extends over still unresolved details of the accident sequence, despite a detailed consideration.

With the estimated value of the annually expected frequency: h(v) = h(k) X w(m/k) X w(b/km) X w(g/kmb), the amount R(v,a) of the value set v of the desired risk coefficient estimated as the product of the expected magnitude of damage and the expected frequency R(v,a), is about equal to x(v,a) X h(v). The estimated value of the risk coefficient for a particular type of damage 'a' is then the sum of the risk contributions R(v,a) of all considered value sets v, or, according to (2.2):

$$R(a) \cong \sum_{v} R(v,a) = x_1 \times \tilde{h}(x_1) + x_2 \times \tilde{h}(x_2) + \ldots$$

where for instance, $\tilde{h}(x_1)$ is the sum of the frequencies of that value sets v whose expected damage was estimated at x_1.

2.5.4 Uncertainties in Risk Assessment Based on Detailed Information

The discussion in Section 2.4.4 clearly illustrates that risk estimations based solely on observations from previous years do not have anything to do with objective certainty. If, for instance, someone were to estimate the collective risk for the type of injury "loss of human life" due to the event "traffic accidents" on the basis of numbers from a previous year, then this could be done either by directly taking the value from the year 1978 or by the detour via (2.2) with frequency estimations based on accidents from the year 1978. This statistic estimation would not consider the uncertainty under point (e) in Section 2.4.4. If, for instance, traffic safety regulations are tightened for the forthcoming year, this could affect the expected frequency of damage cases or the number of deaths per event, so that the statistic estimation using numbers from the year 1978 would not be credible. For a better estimation we should use not merely observations from previous years, but technical knowledge, as well, including this amendment in a detailed description (model) of the accident situation. The following section shall show that even risk estimations based on models (i.e., detailed descriptions of event and exposure sequences) can still contain considerable uncertainties in spite of the use of detailed information.

2.5.4.1 Does the Model Describe the Risk? Several questions arise regarding the four regions outlined in Figure 2-6:

- Event sequences

  --Does the model contain all event sequences contributing to risk? It is clear that in an estimation of risk, not all event sequences can or need be included. In principle, we can imagine any number of event sequences. The number depends only on the degree of resolution of the description. Several event sequences will provide dominant risk contributions; others, however, will contribute amounts that are irrelevant to the magnitude of damage and their total frequencies. The problem consists in considering all relevant event sequences. Whether this has been done cannot generally be demonstrated mathematically.

  --Are all important relationships between the failures contained in the fault trees and those existing between the subevents contained in the relevant accident sequences? Missing relationships can result in underestimation of the frequency sequence and thus in the risk.

  --What is the level of the uncertainty propagated via fault trees and event trees in the frequency and probability estimations? These are expressed by probability distributions, i.e., by citing value ranges within which we can find, with a certain subjective probability, the expected frequency of the event sequence.

- Characteristics

  Were important components of the event characteristics (e.g., possible occurrence of poison gas in the case "fire in the building") overlooked or their extent overestimated or underestimated? This could have an effect on the assignment to categories, on classification within categories, and possibly on the spectrum of types of damage to be considered.

- Exposure sequences

  Are the considered exposure sequences overestimated or underestimated with regard to their time and location-dependent intensity or according to their probability? Are exposure sequences (relevant to consequence and probability) sufficiently represented in the considerations?

- Damage and risk

  --Are all important types of damage included?

  --Are the relationships correct for a conversion of the harmful effect into type and magnitude of injury?

  --What is the level of uncertainty in the estimated expected consequence x(v,a) and in the estimated expected frequency h(v) in the particular value sets v from Section 2.5.3?

2.5.4.2 Quantification of Uncertainties in the Risk Assessment. Uncertainties in estimation result from:

- Inaccurate knowledge of fixed quantities--or quantities presumed to be fixed for time period under consideration--like probabilities, expected frequencies, average values in general, etc.

- Approximated description of regularities in the event and exposure sequences. Among these regularities are the laws of chance, expressed by distribution functions or reduced to expected values.

Uncertainties in estimation are expressed by probability distributions, i.e., by citing value ranges within which the correct value of the uncertain quantity will be found with a certain subjective probability. If we run the quantified uncertainties through the outline of risk assessment (see Figure 2.6), then as a result we obtain probability distribution for the risk coefficient. Thus, we have value ranges that contain the correct numerical value of the risk with a certain subjective probability called the confidence interval.

The value sets v from Section 2.5.3 thus yield a whole region as a risk contribution, not merely a single point in the (magnitude of damage [x], frequency [h]) diagram, together with numbers g(x,h), as the density of subjective probability, for the level of the particular risk contribution (see Figure 2-7).

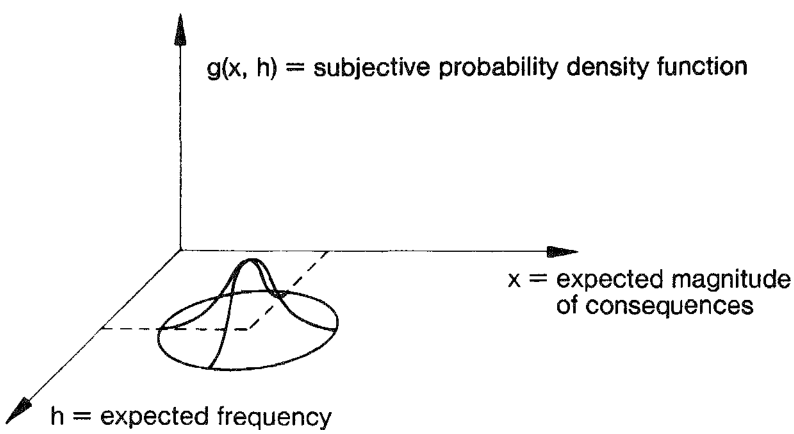

Figure 2-7. Risk contribution of a subevent with subjective confidence interval

2.6 PRESENTATION OF ESTIMATED RISK COEFFICIENTS

The immediate factors affecting the risk coefficient are the magnitude of damage and the probability with which damages of the particular scope and nature will be caused in the particular time frame (usually one year). In principle it is sufficient to give the risk coefficient together with a subjective confidence region when:

- only damage magnitude 0 or 1 is possible (e.g., for individual risk in the damage type "loss of human life"), or
- the potential damage magnitudes per event are not too different and the event occurs frequently (e.g., the collective risk from the event "pneumonia" or "traffic accident").

The requirements for the type of citation of risk coefficient look somewhat different if contributions from rare occurrences are included that lead to a large magnitude of damage. For instance, the risk coefficient "0.01 per year" means that on the average, of the different, equally possible sequences for a year, the magnitude of damage per year is 0.01 (see Section 2.4.3). However, this ratio can come into being by comparing, for instance:

- one possibility with level of injury 1:99 possibilities with level of injury 0, or
- one possibility with level of injury 10,000:990,000 possibilities with level of injury 0.

If the possibility to suffer damage exists only during the next 100 years, then the risk coefficient "0.01 per year" states in the first case that during this period, damage of level 1 is expected. But in the second case, it is meaningless to speak of an expected damage 1 because in the course of 100 years, there will either be no damage (probability about equal to 0.9999), or damage of magnitude 10,000 (probability about equal to 0.0001) will occur. For these reasons, in the case of rare events with a large magnitude of damage, we need not only the risk coefficient, but also the two components of damage magnitude per year and probability (or magnitude of damage per event and expected annual frequency).

As a rule, in this regard we are interested in the probability that the level of damage in a year will be greater than or equal to a given value Y* or, in the expected annual frequency with which damage of a magnitude X greater than or equal to X* will occur. In order to answer this question in the case of frequency, we would have to add all expected frequencies of risk contributions with injury scope X greater than or equal to X*. This addition is already anticipated in the risk illustration using the so-called complementary cumulative distribution function. It is called "complementary" because it denotes the expected frequency of X greater than or equal to X*, whereas the cumulative frequency distribution does this for X less than or equal to X*. The complementary cumulative distribution function therefore answers the question for each value X*: "What is the level of expected annual frequency at which damage magnitude greater than or equal to X* will be caused?"

Subjective confidence intervals for the individual risk amounts (see Figure 2-7) give a picture in Figure 2-8 of the determined complementary cumulative distribution function. This picture is the subjective confidence interval of the curve, and it states: "Propagation of quantified uncertainty from Section 2.5.4 by means of the outline in Figure 2-6 permits the conclusion that the particular curve with P% subjective confidence interval lies somewhere in the region between the two boundary curves, provided the influence of nonquantified uncertainties is negligible."

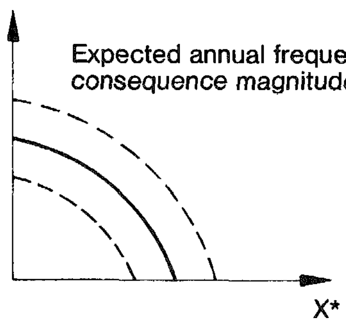

Figure 2-8. CCDF with subjective confidence interval

A region of analogous significance can also be given for the risk coefficient. As a result of the separate risk presentation of the different types of damage, we no longer know which consequences in the various types are connected with a value set v from Section 2.5.3; that is, which ones are caused simultaneously. This state of affairs could be illustrated for individual value sets by means of a table.

2.7 SIGNIFICANCE OF SMALL PROBABILITIES AND FREQUENCIES

The expected frequency of "0.001 per year," for instance, means the same thing as "on the average, once every 1,000 years." It is not possible to imagine periods of 1,000, 10,000 years, or even 100,000 years and more. Table 2-4 attempts to give an impression of this by using a historical scale.

Table 2-4. Periods corresponding to the frequency "10^{-3}, 10^{-4}...10^{-9} per year"

Years before our time	
1,000	Charlemagne Era
10,000	End of the fourth Ice Age (middle of Stone Age)
100,000	Beginning of first Ice Age (Neanderthal Man)
1,000,000	Australopithecine Period
10,000,000	Evolution of man?
100,000,000	Formation of modern mountain ranges (first blooming plants)
1,000,000,000	Beginning of central Precambrian (age of so-called prehistoric times).

The annual probability of an event is never greater than the expected annual frequency. Therefore, we can use the latter value, provided it is not greater than one, as an estimated value of the annual probability without underestimating the probability.

Whether, for instance, an event with annual probability 10^{-6}--that is, an event that occurs, on the average, once every million years--will occur and, if so, in which year, are questions that cannot be answered. Therefore, it is enough for many persons to know that the event can occur at any time, provided its probability is not exactly zero. Therefore, with regard to the question "In this year?", one should illustrate the probability on the basis of a number of potential, equally probable annual frequencies without the event. With this probability of 10^{-6} per year, we have, on the average, one possibility for the sequence of a year with occurrence of the event among 999,999 equally probable different potentials for the sequence of a year without occurrence of the event. Only one of the many different, equally probable sequences can occur in the particular year. The chances that exactly one annual sequence will occur that contains the event are therefore one to 999,999. The understanding of probabilities on this order of magnitude depends primarily on whether one can imagine, for instance, a quantity of 1,000,000 equal objects, none of which has any distinguishing characteristics. For instance, 1,000,000 railroad ties correspond to a distance of Frankfurt-Paris, Munich-Cologne, Hannover-Munich, or about 640 km of tracks, or about seven hours of fast train travel.

Were someone to draw the correct identity card belonging to an inhabitant of a city of 1,000,000 population on the basis of available fingerprints (the probability is 0.000 001 = 10^{-6} for a random access), then we would call this an unbelievable accident. 1,000,000 file cards measuring 200 X 150 X 1 mm will fill about 100 files measuring 100 X 100 X 50 cm. Were the experiment repeated with another set of fingerprints and again the correct card were drawn after only one access to the million cards (the probability is again 10^{-6} for random access, but 0.000000000001 = 10^{-12} for the event "successful draw in both cases"), then we would probably talk about a parapsychological case.

The assertion that a certain event of probability "10^{-6} per year" could happen in this year is therefore no more or less correct than, for instance, the assertion of any person that he could in a single attempt--that is, spontaneously--withdraw the card belonging to the set of fingerprints or correctly identify the railroad tie along the Frankfurt-Paris route where, for instance, a certain key dropped from a train window had come to rest.

If an event has the probability "0.000001 per year" for each of the next 50 years, then the probability that it will occur during these 50 years is somewhat less than 0.00005, i.e., on the average, of the 20,000 different, equal possibilities for the sequence of the next 50 years, there is exactly one in which the event occurs.

In all these attempts to illustrate the significance of small probabilities, we have always assumed that the probability value is correct. Is it possible to obtain a meaningful estimation of probabilities of this order of magnitude (0.001 and less per year), i.e., with a usefully small confidence interval?

In the above example of the population file it was easy to determine the probability. The number of different, equally justified possibilities of drawing the correct card is 10^{-6} = 1/1,000,000; this is also the number of different, equally justified possibilities of drawing any of the 1,000,000 cards. Or the other probability, the number of different, equally justified possibilities of drawing the correct pair of cards is 10^{-12} = 1/1,000,000,000,000 which is the number of different, equally justified possibilities of twice drawing any of the 1,000,000 cards. The latter is equal to the product of the probability of making a correct first draw (10^{-6}) times the probability of making a second correct draw. The event sequence is easy to overview; there can be no dependence of the second draw on the first if we ignore personal preferences in the draw, as was done here. In addition, only one event sequence (partial event) leads to the occurrence of the event, namely drawing the correct card (or the correct pair of cards) on the first try (or on the first two trys). Therefore, completeness of the sum over all partial events of the considered event has been demonstrated.

The situation is similar, for instance, in the game "6 of 49" (f). The probability of drawing the six proper numbers from 1, 2, 3, . . ., 49 in the next game is equal to: w_1 times w_2 times. . .times w_6 = 6/49 times 5/48 times...times 1/44 is about equal to 0.00000007 = 7 times 10^{-8} where, for instance, w_6 = 1/44 is the conditional probability of drawing the 6th number after the five previous numbers have already been correctly drawn (g).

Even the frequencies and probabilities as determined by the procedure in Section 2.5 come into being by multiplication and summation (see equations 2.5 and 2.6). The reason for small frequencies or probabilities is due to formation of the product. However, the probabilities affecting the product are not quite so simple to estimate as in the above examples. Even the event sequences are not so simple to oversee. For this reason, the validity of forming the product (inclusion of possible relationships between secondary events and failures) and the completeness of summation (consideration of all relevant event sequences) are not so easy to estimate. The subjective confidence intervals are a means of numerically expressing the influences of these uncertainties in our best judgement.

FOOTNOTES

(a) If the annual consequence magnitude y is not limited to a finite number of different values but is distributed uniformly with the probability density function w(y), then we obtain the risk coefficient in an analogous manner according to R = integral from zero to infinity of y X w(y)dy.

(b) If the consequences are independent per each event, uniformly distributed and independent of the number of events, then the risk coefficient can also be determined according to $R = \overline{x} X \overline{h}$ ("expected injury per event" times "expected number of events per year"). Here, $\overline{x}$ is the expected magnitude of damage per event and $\overline{h}$ is the expected annual frequency of the event.

(c) Under an assumption of a binomial distribution, for the number of crashes among "n" flights, from n = 1,000 observed flights with no crash, we can give a confidence interval of zero less than or equal to w less than or equal to w* for the crash probability w, which will contain w with a probability of P%. w* can be obtained in this case from a relation between binomial and Poission distribution with a parameter a = n X w. From the Poission distribution we obtain a probability (average w) for more than zero crashes of:

$$\tilde{w} = \sum_{i=1}^{\infty} (e^{-a} a^{i}/i!) = 1 - e^{-a} = 1 - e^{-nw}$$

Therefore, w* can be obtained from the following expression according to the text:

$$\sum_{i=1}^{\infty} (e^{-w^{*}n} (w^{*}n)^{i}/i!) \geq P/100 \text{ oder } e^{-w^{*}n} \leq 1 - P/100$$

If P = 95 there results w* is about equal to 0.003. Even by means of the known relationships between binomial and F distribution or Poisson and χ^2-distribution, we obtain the upper 95% bound 0.003 for w.

(d) The probability of the event "collapse under load" generally does not correspond to the probability of the event "collapse within the planned service life." For its calculation we either need information about the frequency of load application, or we must consider the partial event in the form "strength = r, load s greater than or equal to r" and the probability density function of the maximum annual load can be applied.

(e) As a simplification we assume here that the load and strength can assume independent whole number values s and r from the value range $I_S = [0,s^*]$ or $I_R = [0,r^*]$ in accordance with the probabilities w_S (s) or w_R (r). The probability w(s, E) of the partial event in Figure 2.1 is thus the product of the probabilities of the secondary events "load = s" (= $w_S(s)$) and strength r less than or equal to s":

$$(= \sum_{r=o}^{s} w_R[r]),$$

therefore:

$$w(s,E) = w_S(s) X \sum_{r=o}^{s} w_R(r).$$

With every load value s from I_S, there is one connected partial event of the event "collapse under load." The partial events are mutually exclusive because of the different load values so that:

$$w(E) = \sum_{s=o}^{s^*} (w_S [s] X \sum_{r=o}^{s} w_R [r])$$

is the probability of the event "collapse under load." For the case where the load and strength can assume any independent values from (0, infinity) in accordance with the so-called probability density functions $w_S(s)$ or $w_R(r)$, then we have:

$$\dot{w}(E) = \int_0^\infty w_S(s) \times \int_0^s w_R(r)drds$$

(f) Translator's note: a game similar to Bingo.

(g) The fact that we often hear of "six correct draws" in spite of this low probability is simply due to the fact that so many games are terminated and each terminated game represents an access to the "file," which contains 13,983,816 different equal probabilities of drawing "6 from 49." Of 10,000,000 terminated games in a week, a probability of about 0.5 results that at least one of the games will have the six proper numbers. If we were dealing with an event of probability "7×10^{-8} per year," then each of 10,000,000 terminated games would represent one completed year.

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Section 3

THE NUCLEAR POWER PLANT

3.1 INTRODUCTION

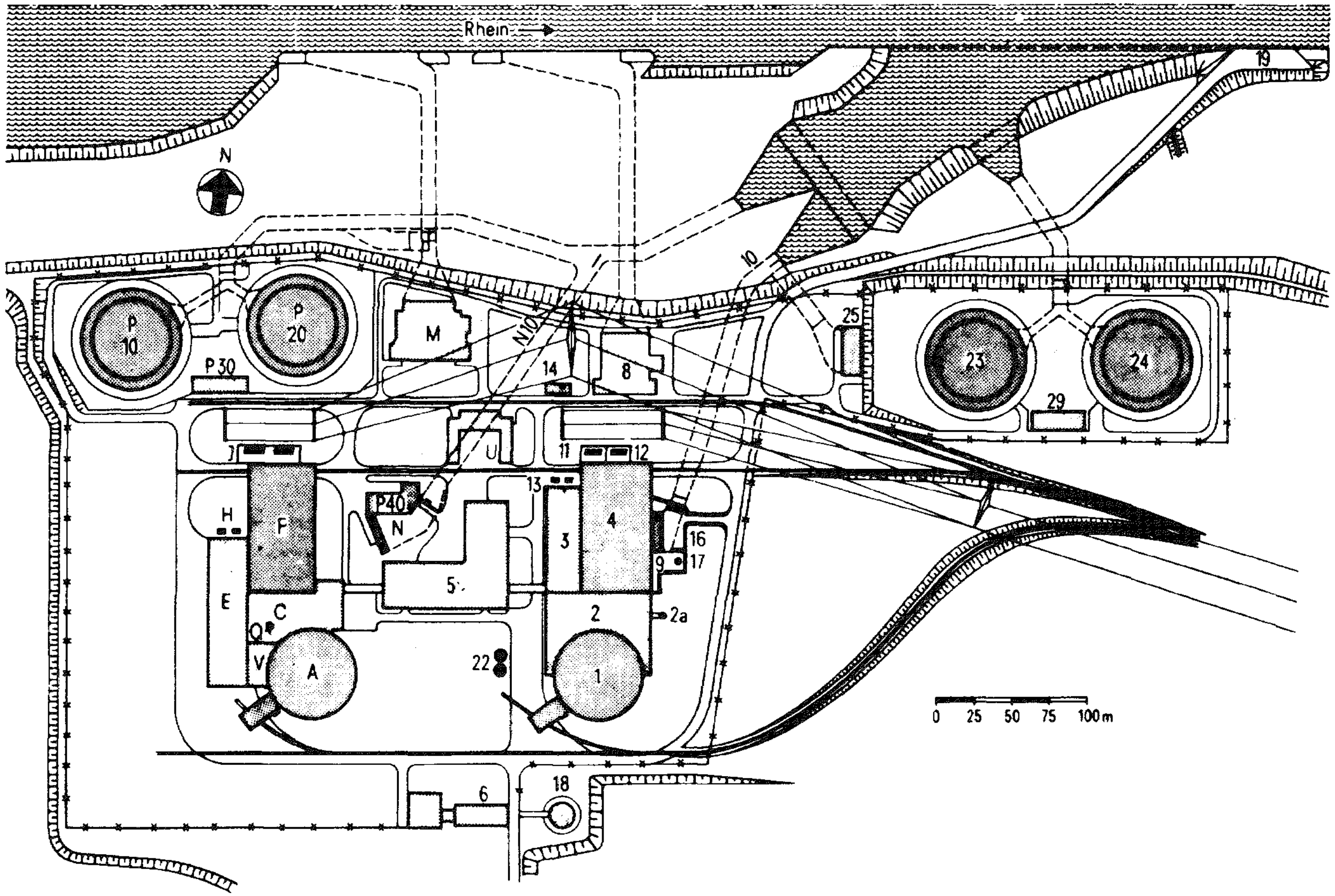
This Chapter provides an overview of the design, operation, and safety equipment of a nuclear power plant with pressurized water reactor (a). For plant systems analysis studies, it was necessary to select a certain nuclear power plant design as a reference. Various considerations had to be taken into account for this.

On the one hand, the power output, design status and engineered safety features would have to be "state of the art" for plants presently in operation or in planning.

On the other hand, a risk assessment requires very detailed information for plant system studies. This information is sometimes not available until after completion and operation of a plant.

After weighing both considerations, the nuclear power plant Biblis, unit B, was selected as a reference plant for the study. Nuclear test operation of this plant began in the spring of 1976. The plant was transferred to the owners in the beginning of 1977. The Biblis B nuclear power plant has a pressurized water reactor (PWR) with a thermal power output of 3,750 MW. The electrical output of the plant is 1,300 MW. The power plant was constructed by Kraftwerk Union (KWU) AG and Hochtief AG in the community Biblis under contract to the Rheinisch-Westfaelischen Elektrizitaetswerke (RWE) AG. Figure 3-1 shows a site plan and Figure 3-2 an aerial photograph of the power plant grounds with units A and B. The most important buildings of unit B are the reactor building, the primary auxiliary building, the operations and switchgear buildings with diesel emergency power generating wing, the turbine hall, the coolant water purification and pump structure, and the buildings of the cooling tower area.

Section 3.2 gives a brief overview of the basic design and mode of operation of the PWR nuclear power plant. In Section 3.3, general viewpoints and principles of reactor safety engineering are discussed. Section 3.4 describes the design and



Unit A
1 Reactor building
2 Reactor auxiliary building
2a Exhaust stack
3 Control room and operations building
4 Turbine hall
5 Service building
6 Administration building
8 Coolant pumping house
9 Collection pool
10 Coolant return channel
11 380 kV switching station
12 Unit transformers
13 Auxiliary transformers
14 Foundation for reserve unit transformer
16 Rainwater pump station
17 Settling pond
18 Information center
19 Ship landing
22 Demineralizer tank
23 & 24 Cooling towers
25 Cooling tower pump station
29 Cooling tower switching station

Unit B
A Reactor building
C Reactor auxiliary building
E Operations and control room and emergency diesel generator
F Turbine hall
H Auxiliary transformers J-220 kV/380 kV-outdoor switchyard and unit transformers
M Coolant purification and pump assembly
N Collecting pond
N10 Coolant return channel
Q Vent stack
P10 & P20 Cooling towers
P30 Cooling tower switching room
P40 Cooling tower pump station with waste water elevator
U Parking structure
V Covered hallway

Figure 3-1. Site plan for Biblis Nuclear Power Plant, Units A and B

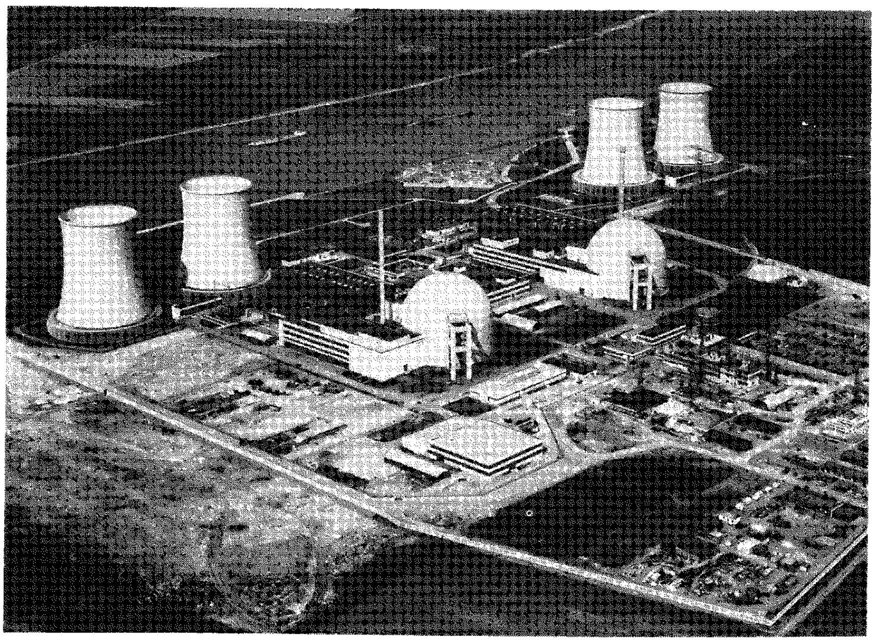


Figure 3-2. Air photograph of Biblis Nuclear Power Plant, Units A and B

function of the most important systems and components of the Biblis B nuclear power plant.

3.2 DESIGN AND OPERATION OF THE PWR NUCLEAR POWER PLANT

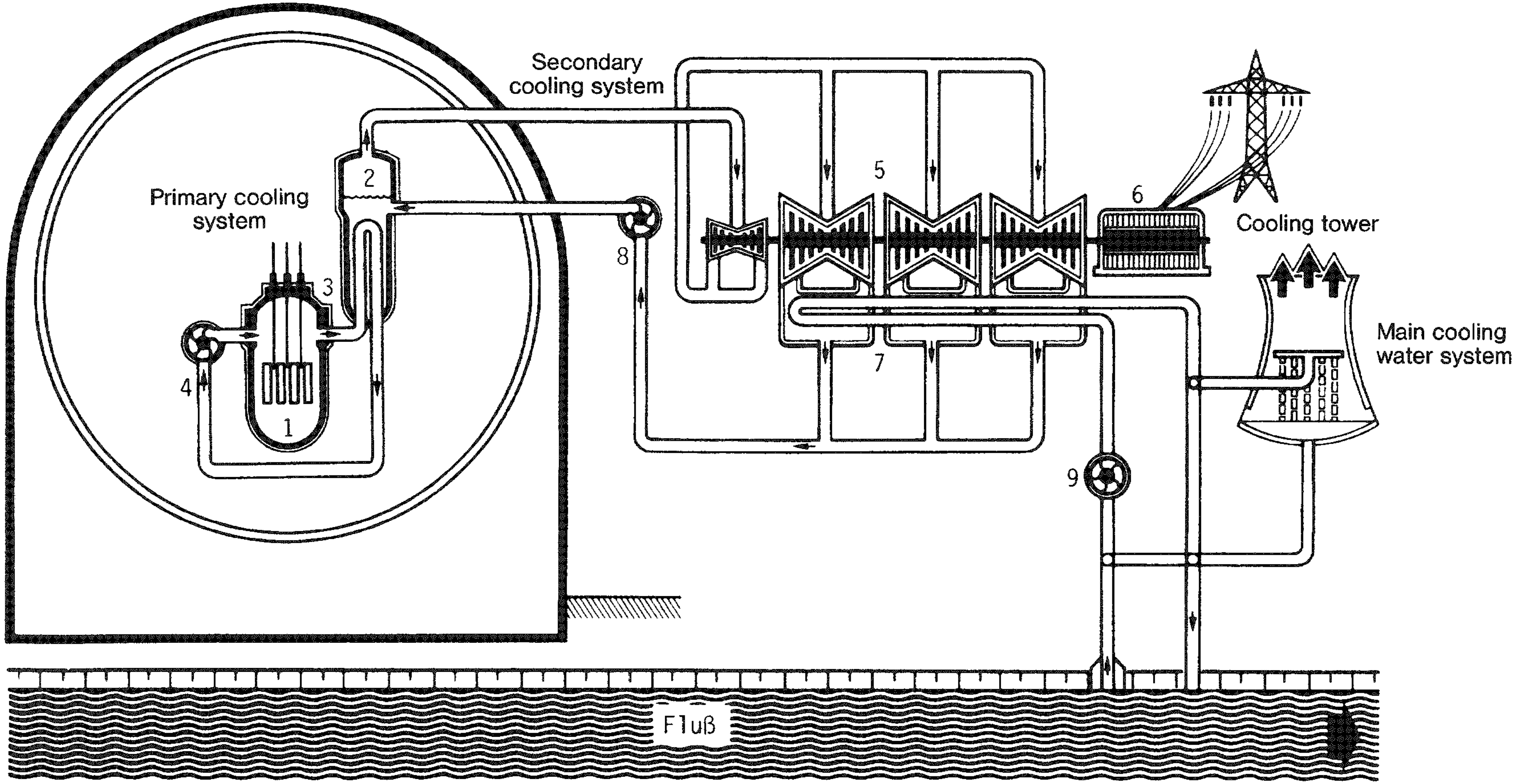
Figure 3-3 illustrates the basic design and operation of the PWR nuclear power plant. The heat generated in the reactor core (1) due to nuclear fission is transferred to the feedwater-steam circuit (secondary loop) via the steam generator (2) by means of the closed reactor cooling loop (primary loop). A sufficiently high coolant water pressure prevents steam formation in the reactor cooling loop (therefore the designation "pressurized water reactor").

The secondary feedwater flowing into the steam generator is vaporized by absorption of heat from the reactor cooling loop. The resulting steam drives the turbine (5), and this in turn drives the generator (6). The steam flowing from the turbine can no longer be used to generate electric energy, and it precipitates in the condenser (7). The water thus obtained is pumped back to the steam generator.

The heat removal from the condenser takes place by means of the main cooling water system. Here, the condenser picks up about 2/3 of the heat generated by the reactor from the flowing coolant water. This heat is released to the environment either directly to the river or through cooling towers, depending on environmental conditions.

The conversion of heat into electric energy in nuclear power plants takes place in the same manner as for other thermal power plants. Energy generation by nuclear fission often causes special problems, however, since radioactive substances are generated on a considerable level. The radiation emitted by these substances can lead to health hazards. The central task of reactor safety engineering is therefore to prevent release of radioactive substances to the environment.

As a result of the disintegration of radioactive substances formed during reactor operation, heat is generated even after reactor shutdown; this is called the residual heat. In comparison with the heat generated during power operation, this amount is small and continues to decrease with time. Unless the reactor core is cooled, residual heat will warm it up to the point where radioactive substances would be liberated. Therefore, it is necessary to cool the reactor core even after shutdown.



1 Reactor core
2 Steam generator
3 RPV
4 Primary coolant pump
5 Turbine
6 Generator
7 Condenser
8 Main feedwater pump
9 Main cooling water pump

Figure 3-3. Functional schematic for a PWR nuclear power plant

From the above discussion, there results a number of safety engineering requirements for the design of a nuclear power plant. The following sections will discuss them in detail.

3.3 THE CONCEPT OF SAFETY

The objective of all safety considerations and safety measures derived from them is to ensure at all times the retention of radioactive substances present within a nuclear power plant. In order to do this, an extensive safety concept has been developed in nuclear engineering. It consists of a multiple enclosure of the radioactive substances generated in a reactor and of engineered safeguards and other measures that maintain at all times the integrity of this enclosure of radioactive substances. The basic design of the safety concept will be outlined briefly below.

3.3.1 Barriers for Radioactive Material

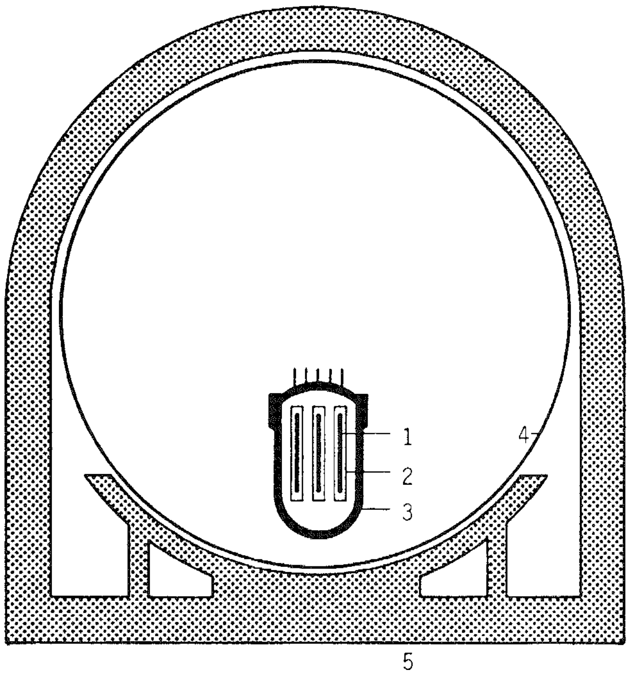
By far the greatest part of radioactive material originates from the nuclear fuel as a result of nuclear fission (b). These fission products are enclosed by several layered structures, called the fission product barriers. Figure 3-4 illustrates the principal arrangement of these structures. In particular we have:

- the crystal lattice of the fuel itself, which retains the vast majority of fission products (under normal operating conditions, more than 95%),
- the fuel rod cladding, which is welded gas-tight,
- the reactor pressure vessel together with the completely enclosed reactor coolant loop,
- the gas-tight and pressure-resistant containment building, which encloses the reactor cooling loop.

The external steel-concrete structure has only a limited sealing function. It permits evacuation of gas leakage from the containment and protects the system against external effects.

3.3.2 Engineered Safety Features

As in all technical systems, nuclear power plants must expect accidents due to various causes. For instance, accidents due to the failure of system parts, due to human error, or even due to external events are all possible. In order to ensure enclosure of fission products, damage to the fission product barriers--even



1 Crystal lattice of the fuel
2 Fuel cladding
3 Reactor coolant system
4 Containment
5 Steel-concrete structure

Figure 3-4. Fission product barriers

during accidents--must be prevented by appropriate design. In order to achieve this, a defense-in-depth safety concept is used in nuclear engineering.

3.3.2.1 Classification of Accidents. It is customary to divide operational states and accidents in nuclear power plants into three safety groups:

Specified Normal Operation

The plant is functioning normally or occurring malfunctions do not affect the operation and safety of the plant. The radiation exposure due to operational discharges of radioactive substances to air and water must be kept as low as reasonably achievable. The prescribed limits for radiation exposure of the environment are established in the German Radiological Protection Ordinance.

Accidents (Design Basis)

Accidents are defined as event sequences during which operation of the plant cannot be continued for safety reasons, but for which the plant has been designed so that the consequences to the environment will not exceed the defined limits. The corresponding acceptable limits for radiation exposure in case of accidents are also established by the radiological protection ordinance.

Emergencies (Class 9 Events)

Beyond the safety design of nuclear power plants, there remains a range of possible event sequences called Class 9 events or Class 9 accidents (c). By these we mean event sequences that either are so improbable, as far as one can judge, that specific measures to prevent or limit their consequences are normally not taken, or whose occurrence and sequence are not forseeable. During Class 9 events, the acceptable limits established for radiation exposure in the Radiological Protection Ordinance can be exceeded.

3.3.2.2 Defense-in-Depth Principle. The task of reactor safety engineering is to prevent accidents, if possible, or when not possible, to limit the consequences of accidents. A defense-in-depth concept has been developed for nuclear power plant safety for this reason. We generally distinguish three levels of safety measures:

Quality Assurance

This safety level includes all requirements of standard design and quality, especially of the nuclear components. In addition to component and system designs with a high level of safety margin, measures are also provided for an extensive quality assurance in the manufacture of components and in the construction of the plant. For instance, a multiple, independent inspection of important safety components like the reactor pressure vessel, coolant pipe lines, and containment is performed. The high level of quality is assured through continuous recurrent testing during the entire life of the plant. These measures for quality assurance are aimed at keeping the accident frequency as small as possible, i.e., to prevent accidents as much as possible.

Prevention of Accidents

To prevent accidents that could develop from other malfunctions, nuclear power plants are equipped with redundant control and protective equipment. These systems are used for timely recognition of potential accidents and to trigger actions for the immediate limitation of occurring malfunctions.

The most important safety feature is the reactor protection system. It continuously monitors all important measured values in the system, for instance, reactor power output, pressure in the reactor coolant system, main coolant pump speed, etc. The reactor protection system automatically initiates protective measures such as reactor shutdown, when the monitored process parameters exceed defined setpoint values.

Limitation of Accident Consequences

The third stage of the safety concept for nuclear power plants is the provision of extensive engineered safeguards called safety systems. Triggered by the reactor protection system, these safety systems generally act automatically during accidents to maintain the integrity of the enclosure of fission products and to limit the harmful consequences connected with the accident. These safety systems are designed to effectively control a broad spectrum of potential accidents.

The design of the safety systems is oriented, however, to a few important design basis accidents that generally lead to the highest exposures and thus to the greatest demand requirements of the safety systems. For instance, a double-end break of a primary coolant pipe is the determining accident for the design of the containment.

3.3.2.3 Principles of Engineered Safety Features. The primary requirements of the engineered safeguards of nuclear power plants in the FRG are primarily established by the safety criteria by the Federal Ministry of the Interior (1) (analogous to General Design Criteria in the U.S.). Regulatory positions and standards (similar to Regulatory Guidelines and Standard Review Plans in the U.S.) are given by the guidelines of the Reactor Safety Commission (RSK) (2) and in technical safety regulations of the Nuclear Safety Standards Commission (KTA) (3). In addition to requirements established in detail for protective and safety equipment, these criteria and regulatory standards also contain general design principles. Besides the already mentioned measures for quality assurance, a high level of operational reliability of the protective and safety equipment is to be achieved by these design principles.

Since failures of components are possible, the design principles require that functioning of the protective and safety systems is assured even for failure of individual components. This considers both independent as well as mutually dependent failures.

The primary design principles are explained below.

Redundancy

The most important principle against individual failures is called redundancy. Redundancy means that for each safety function there are more components or subsystems available than actually needed to perform the function. For instance, of the four mutually independent individual subsystems of the emergency and residual heat removal (RHR) system, generally two loops are sufficient for satisfactory reactor cooling. If one subsystem fails due to an individual fault (e.g., the pump of subsystem one does not start up), then operation of the emergency cooling system is not endangered since there are still three other subsystems available.

In addition, redundant safety systems are usually physically separated from each other and provided with special functional protection. These measures primarily offer protection against subsequent failures and against interfacing internal influences (e.g., fire, flood), as well as against external events.

Diversity

The redundancy principle--multiple design of a system using the same type components or subsystems--does not always afford sufficient protection against interdependent failures. Such failures can occur simultaneously due to a common cause in redundant subsystems and can neutralize the redundancy. Failures that go beyond the redundant systems are generally called common mode (common cause) failures. In order to minimize common cause failures, extensive measures have been taken both in design and manufacture, as well as in operation.

Diversity is an important protection against common cause failures. Multiple equipment designed for the same safety purpose has been built-in according to different design principles, and their actions are initiated by physically different functional and triggering principles. The diversity principle is used primarily in the reactor protection system. In accordance with the Nuclear Safety Standards Commission (KTA) regulation 3501 (4), each accident considered within the framework of a design basis accident analysis should be monitored by the measurement of at least two diverse process quantities. Accordingly, the reactor power output can be measured either by the neutron flux or by the heat-up rate of the coolant, for example.

Fail-Safe

Another important principle to prevent the consequences of independent or common cause failures is the principle of failure in a safe direction, briefly called the fail-safe principle. Accordingly, safety systems are designed so that, if possible, the plant or plant components go to a safe mode upon failure. The most important examples are the control rods. They are held by electromagnets so that, upon failure of their electric power supply, they drop into the reactor core and shut down the reactor.

Self-Regulation

If a system is designed so that failures of components or subsystems are recognized and indicated by the system itself, then we call this self-regulation. This principle is generally applied in the plant protection system.

3.4 DESCRIPTION OF SYSTEMS AND COMPONENTS

Before we discuss in detail the most important systems and safety features of the nuclear power plant, a brief overview is provided (Figure 3-5).

- The reactor core produces the thermal power output of the nuclear power plant, it contains the most important fraction of radioactive substances in the plant (Section 3.4.1).

- The reactor scram system is used to quickly terminate the nuclear chain reaction. Thereafter, energy production in the reactor core is merely the residual decay heat (Section 3.4.1).

- The reactor coolant system consists of the reactor pressure vessel, the primary side of the steam generator, the main coolant pipelines, the main coolant pumps and the pressurizer. The heat generated in the reactor core is transferred from the reactor coolant system to the steam generators. (Section 3.4.2).

- The feedwater-steam system consists of the secondary side of the steam generator, the main steam pipelines, the turbine with turbine condenser, the main condensate system with main condensate pumps, feedwater tank, and the main feedwater system with main feedwater pumps. At full power operation the heat from the boilers is transported to the turbine in the feedwater-steam system (Section 3.4.3).

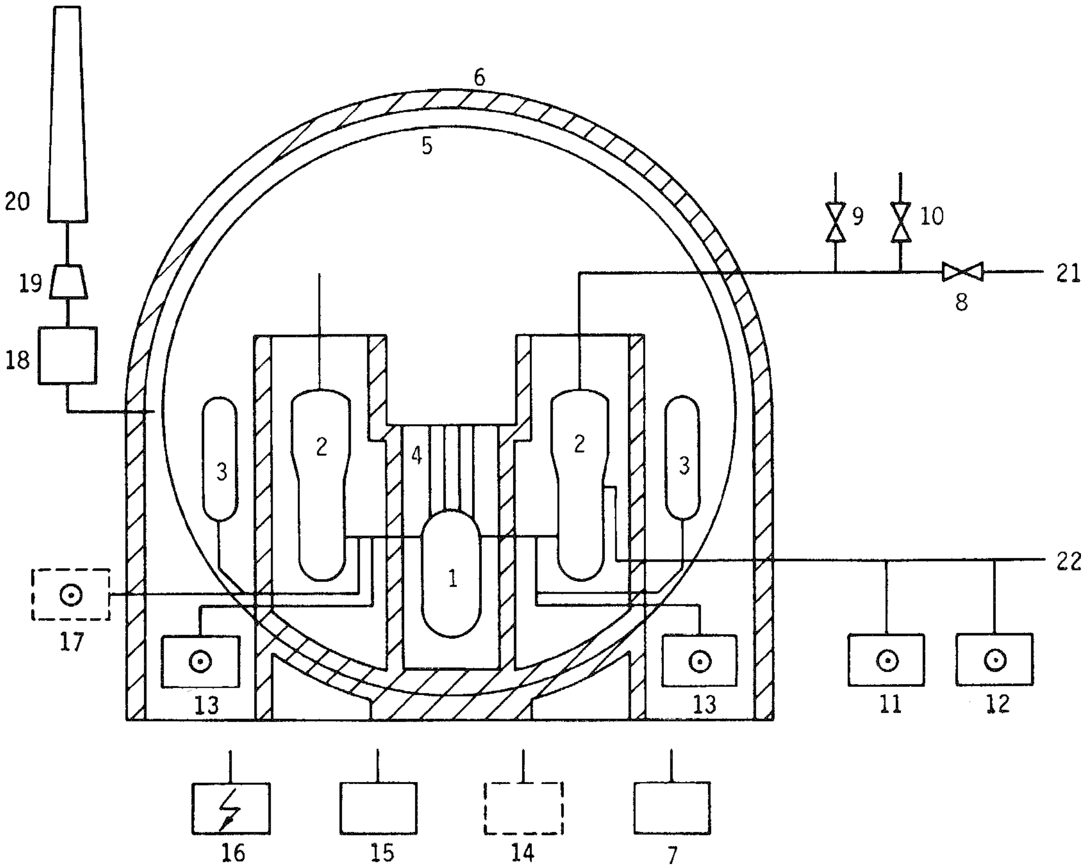
- The volume control and chemical injection systems keep the volume of coolant water in the reactor coolant system constant. By injection of boric acid or deionate, the reactor power output can be controlled over the long term (Section 3.4.4).

- It is the task of the control features to keep the important operating parameters within given operating ranges (Section 3.4.5).

- The reactor protection system controls all parameters relevant to safety and initiates automatic protective actions when limiting setpoint values are reached (Section 3.4.6).

- The electrical energy supply consists of the internal supply system and the emergency power system. The emergency power system supplies the important safety components on failure of the normal power supply (Section 3.4.7).

- The emergency feedwater system supplies the steam generator whenever the primary feedwater system is not available. The



1	RPV	13	Residual decay heat removal system
2	Steam generator	14	Control equipment (not safety-related)
3	Pressurizer (emergency and residual decay heat removal system)	15	Reactor protection system
4	Reactor scram system	16	Auxiliary power system
5	Containment	17	Volume control system (only partially safety-related)
6	Steel-concrete structure	18	Ventilation system
7	Exhaust and waste-water system	19	Exhaust air filter
8	Main steam isolation valve	20	Vent stack
9	Main steam relief valve	21	To the turbine
10	Relief/control block valve	22	From the main feedwater pumps
11	Auxiliary feedwater system		
12	Emergency system		

Figure 3-5. Safety-related systems in a nuclear power plant

emergency feedwater system can be used to remove decay heat and to shut down the plant (i.e., to reduce the coolant temperature, Section 3.4.8).

- The emergency and RHR system is used to remove decay heat over the long-term after shut-down and cool-down of the reactor. During a loss-of-coolant accident (LOCA), it must also inject cool water into the reactor coolant system (Section 3.4.9).
- The emergency system transfers the plant to a safe mode in case of severe external events (Section 3.4.10).
- The containment encloses the important, radioactive parts of the plant. The surrounding reinforced concrete shell protects the containment against external events (Section 3.4.11).

3.4.1 Reactor Core

Thermal energy is generated in the reactor core by nuclear fission. The fuel, primarily uranium dioxide (UO_2), is located in the fuel rod. A bundle of 236 fuel rods forms a fuel element (Figure 3-6). The reactor core comprises 193 fuel elements. The fuel elements are arranged within the reactor core so that an approximately circular cross-section results (Figure 3-7).

In each fuel element, 20 of 256 possible positions are not occupied by fuel rods. In 61 fuel elements, these positions are assumed by the rod cluster control (RCC) assembly. The 20 control rods contain neutron-absorbing material and form a control element by means of a spider-like structure (Figure 3-6). The control drive shaft is connected to this spider. A control element with drive shaft is called a control rod below.

By stepwise movement of the control rods up and down and the changing neutron absorption in the reactor core connected with this, the nuclear chain reaction can be controlled and terminated by complete insertion of the control rods.

Movement of the control rods takes place by electromagnetically operated jack systems located outside the reactor pressure vessel on the control rod nozzles. During reactor scrams the control rods are disengaged so that they drop into the core.

Open fuel rod positions in fuel elements not equipped with control elements are partly used for measurement purposes.

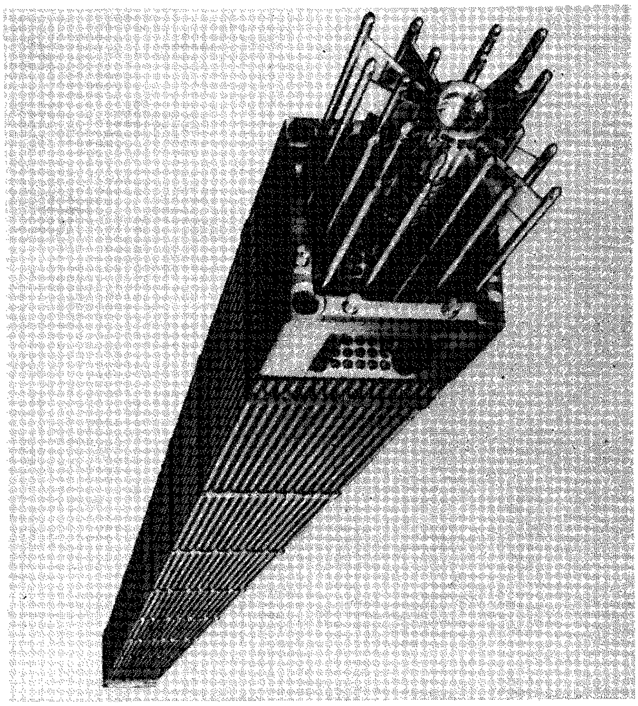
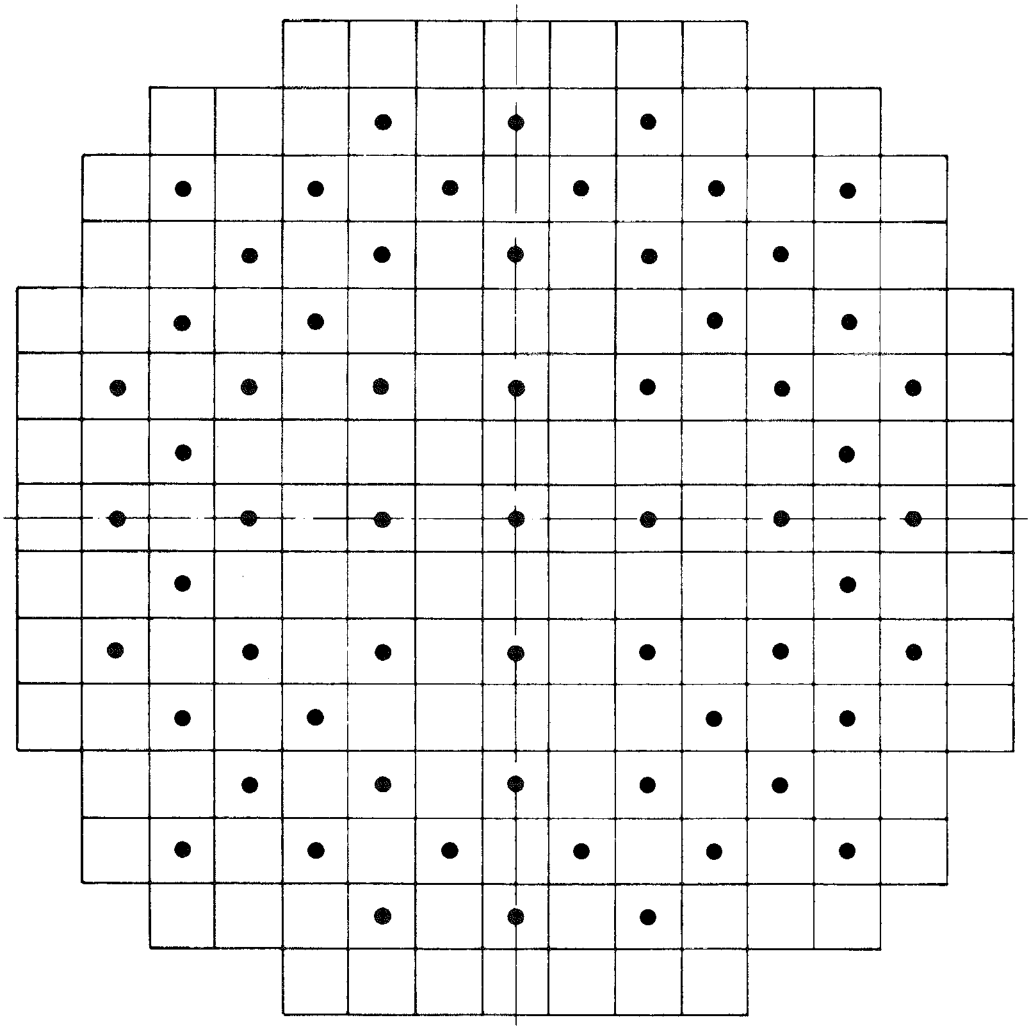
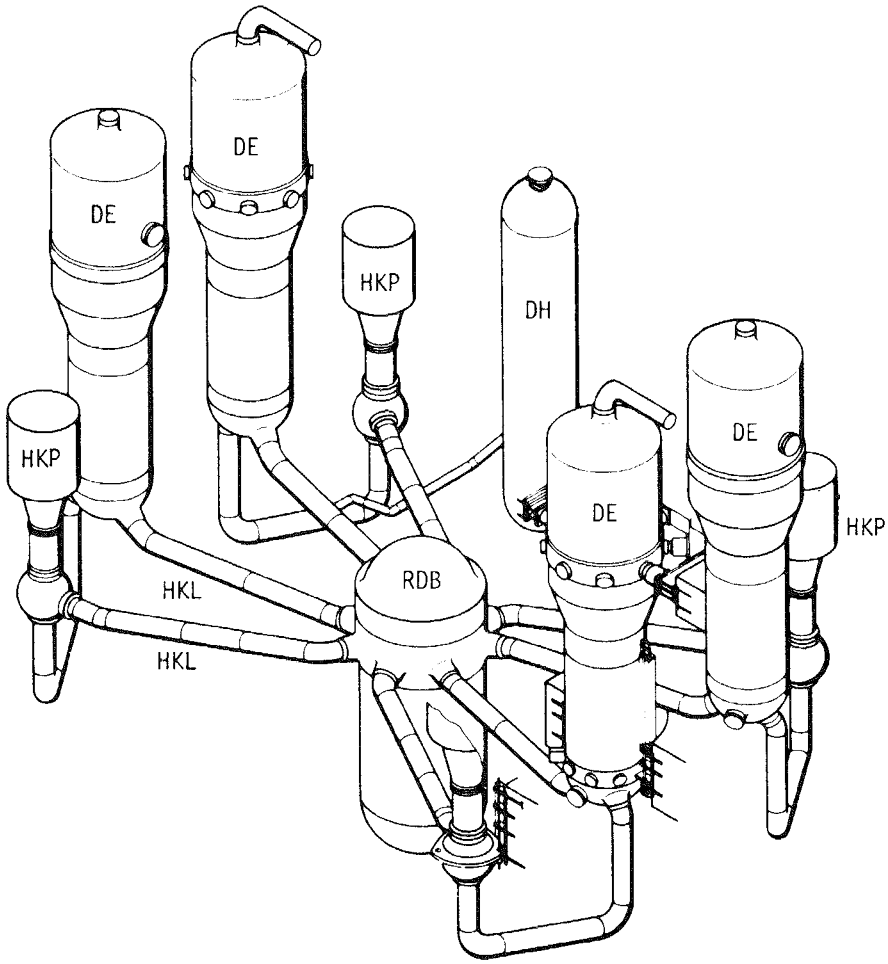


Figure 3-6. Fuel rods with control element



● 61 control rod positions

Figure 3-7. Cross-section of the reactor core



DE = Steam-generator
DH = Pressurizer
HKL = Main coolant pipeline
HKP = Main coolant pumps
RD = Reactor pressure vessel (RPV)

Figure 3-8. Components of the primary cooling system

3.4.2 Reactor Coolant System

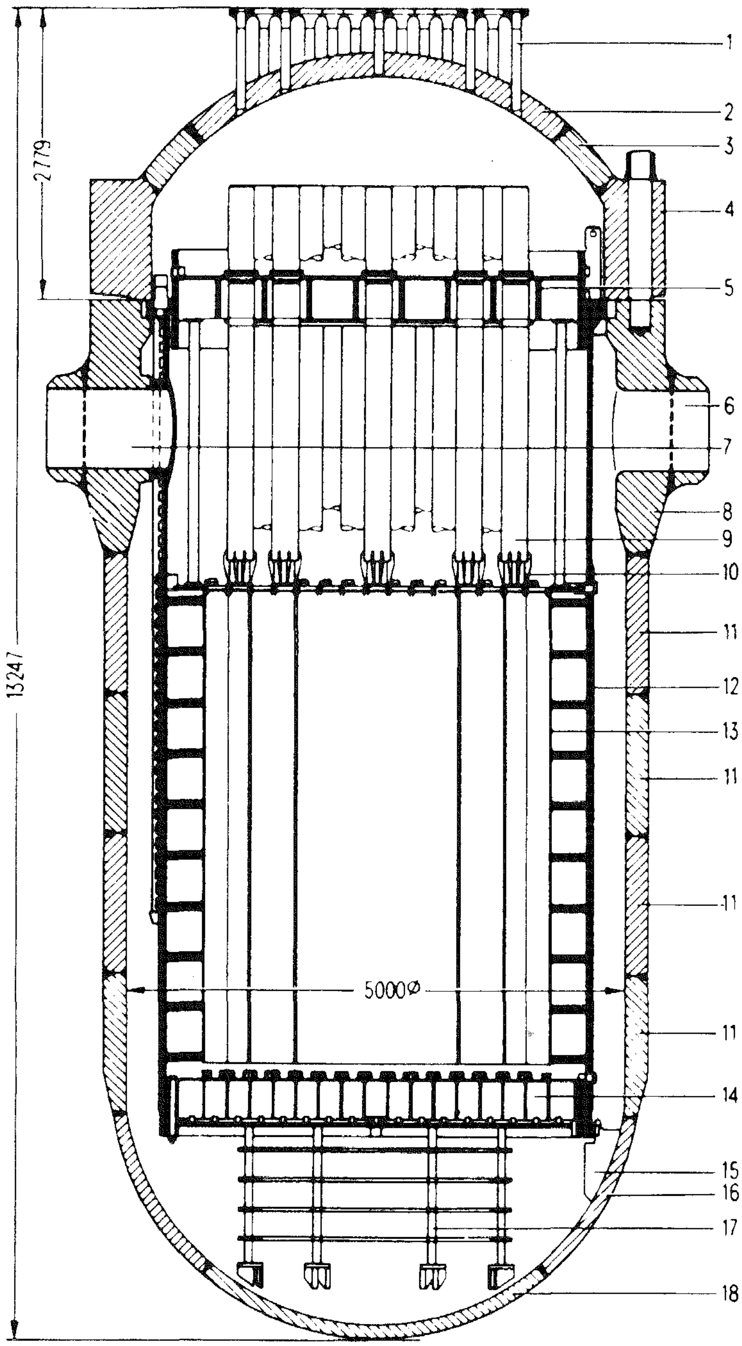
The reactor coolant system (pressure retaining enclosure) consists of the reactor pressure vessel, the four primary coolant loops (each with primary coolant pipelines, steam generator, and primary coolant pump) and the pressurizer system and pressurizer (Figure 3-8).

3.4.2.1 Reactor Pressure Vessels. Figure 3-9 shows a cross-section of the reactor pressure vessel. The hemispherical shaped base of the reactor pressure vessel (RPV) consists of the base calotte and the base ring comprised of several segments welded together. The cylindrical vessel shell is welded together from several, seamless forged rings and attaches to the hemispherical base. The seamless forged shell flange ring with its eight coolant supports forms the transition of the vessel shell to the top head. The reactor pressure vessel lid consists of three forged pieces with screwed and welded control rod nozzles. The lower part and lid of the reactor pressure vessel are screwed together. The RPV is made entirely of the material 22 NiMoCr 37. The interior surfaces wetted by coolant are clad with a corrosion-resistant, austenite-welded plating.

The internals of the RPV illustrated in Figure 3-9 assume mechanical support and exact positioning of the reactor core, guidance of the control rods and of the coolant flow. The entering coolant flows in the annulus between the pressure vessel internal wall and core shell downward into the lower plenum. After a radial deflection, it flows through the reactor core from bottom to top and, after a second diversion, moves to the outlet nozzles.

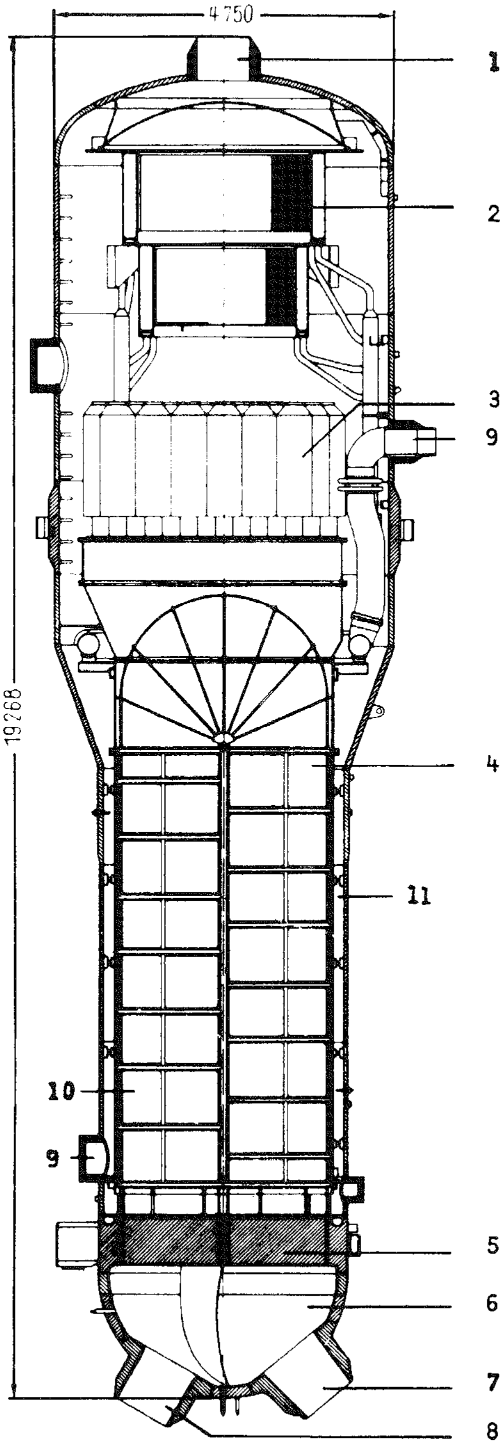
3.4.2.2 Steam Generators, Pumps, Pipelines and Pressurizers Figure 3-10 shows a cross-section through one of the four steam generators. The steam generator is designed as a standing U-tube with steam generator with natural circulation. The important components are a horizontal tube plate with vertical U-tube bundle, a hemispherical channel head under the tube base (divided by a baffle), and a cylindrical vessel on the tube base that surrounds the tube bundle and extends as a dome above the tube bundle. The primary coolant used as heating agent flows through the inlet of the channel head beneath the tube base into the U-tube and from there back into the outlet of the channel head.

On the secondary side, the steam generator operates according to the natural circulation principle. The majority of the feedwater first enters a preheating section of the steam generator situated on the coolant outlet side, where it is



1 Control rod support
2 Upper lid
3 Cover zone ring
4 Cover flange ring
5 Control rod grid plate
6 Coolant inlet nozzle
7 Coolant outlet nozzle
8 Mantle flange ring
9 Control rod shroud tube
10 Upper grid plate
11 Forged ring
12 Core barrel
13 Core baffle
14 Lower grid plate
15 Core vessel support
16 Base zone ring
17 Core support structure
18 Lower lid

Figure 3-9. RPV with internals



1 Main steam outlet
2 Fine separator
3 Coarse separator
4 Tube bundles
5 Tubesheet
6 Collection pool
7 Entry of main coolant
8 Outlet of main coolant
9 Entry of feedwater
10 Preheating chamber
11 Downflow chamber

Figure 3-10. Steam generator

heated to about 10°C below the boiling point. Vaporization of the feedwater occurs in the heating tube bundle, which is surrounded by a tube bundle wrapper. The steam moves through the course separator into the steam dome where residual water is removed by a fine separator. From there, it flows through the outlet nozzle into the main steam pipeline.

The removed water runs downward in the drop space between vessel shell and pipe bundle and enters the heating tube bundle again via the tube plate.

The four main coolant pumps are single-stage rotary pumps driven by electric motors.

The primary coolant pipelines connect rector pressure vessel, steam generator, and primary coolant pumps. The pressurizer is connected by means of a surge line to one of the "hot" primary coolant pipelines leading from the RPV to the steam generator. The pressurizer is used to control the coolant pressure and is partially filled with boiling water. The pressure of the steam cushion above it can be increased by heating the pressurizer or decreased by injecting a spray of water. To do this, spray water is taken from the "cold" primary coolant lines (between main coolant pumps and RPV).

The coolant pressure is limited during accidents by two relief and safety valves each attached to the pressurizer. The steam vented through these valves is condensed in the pressurizer relief tank.

All parts of the reactor coolant system that come into contact with primary coolant are either manufactured of corrosion-resistant material or are plated with welded austenite.

3.4.3 Feedwater-Steam System

The most important components of the feedwater-steam system (secondary system) are seen in Figure 3-11.

The main steam generated in the steam generators moves through the turbine control valves and the quick-closing valves into the high-pressure section and, after interim heating, into the low-pressure section of the turbine.

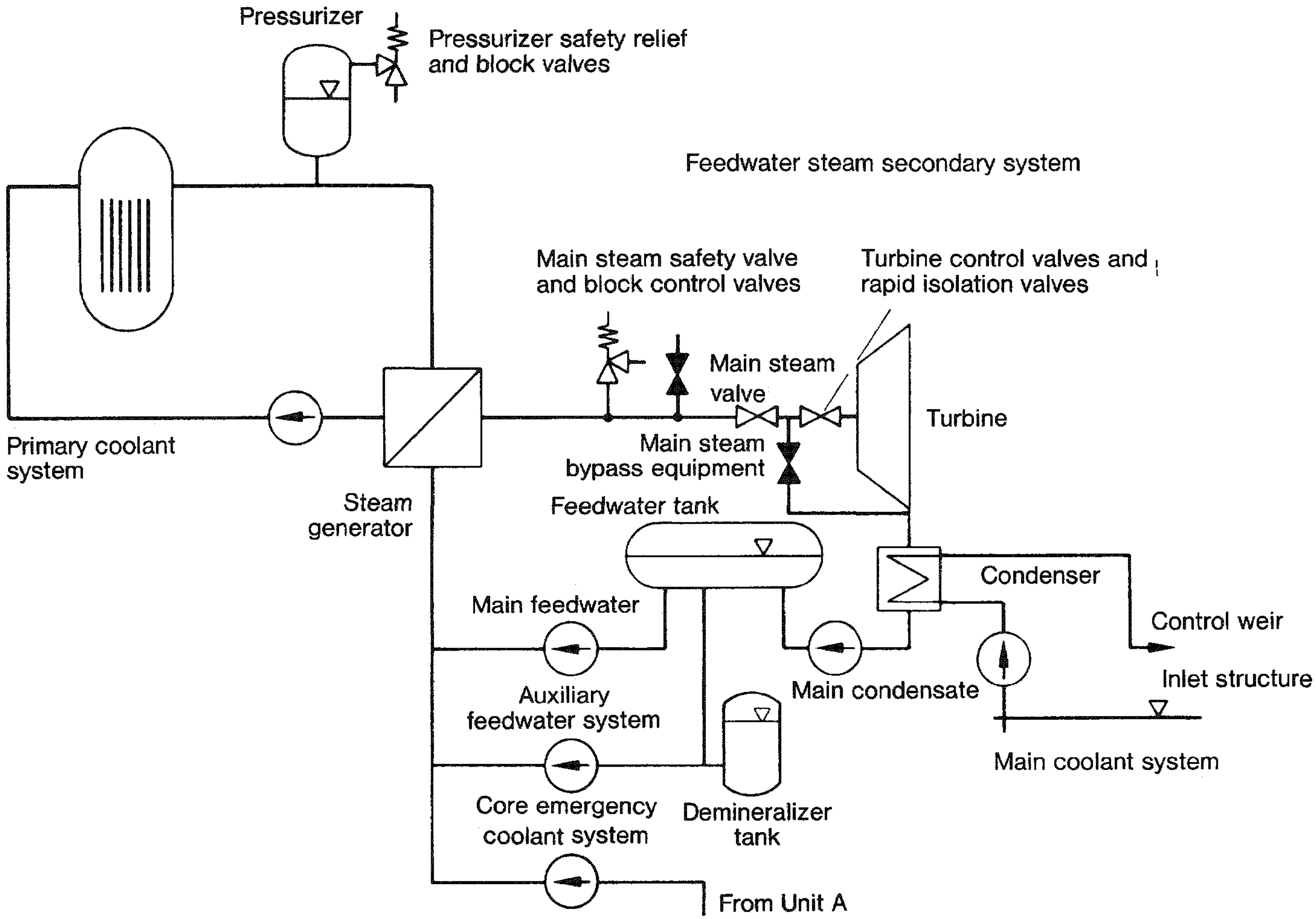


Figure 3-11. Schematic diagram of the main and auxiliary coolant systems

After passing through the turbine, the cooled steam is precipitated in the condensers and then moved as water into the feedwater tank by the main condensate pump. The water is moved by two of the three existing main feedwater pumps from the feedwater tank through the four main feedwater pipelines in which the main feedwater control valves are installed, back to the boilers. Heat removal from the condenser takes place by means of the main cooling water system. In addition, to operate the main feedwater pumps, the turbine, the condenser, and the main condensate system, different auxiliary systems are needed that are cooled by means of a conventional secondary cooling water system (these are not illustrated in Figure 3-11).

If more steam than is needed by the turbine is generated, or if the turbine has to be shut down because of an accident, then the main steam can be diverted directly to the condenser through a main steam bypass mechanism. If the condenser is not available, the main steam is vented to the outside through the relief control valves and main steam safety valves. In this manner, cooling of the steam generator and thus heat removal from the reactor coolant system are made possible. A long-term availability of the steam generator is achieved here by makeup water feed from the deionate tank into the feedwater tank.

If provision to the steam generator through the main feedwater pumps is not possible, then the emergency feedwater system and emergency system can be used for this feed. These systems are discussed in Sections 3.4.8 and 3.4.10.

3.4.4 Volume Control and Chemical Injection System

The volume control system, fed by emergency power, is used to compensate for volume fluctuations in the reactor coolant system that may occur from changes in coolant density due to operating transients or small leaks from the reactor coolant system. Another task of the volume control system is to continually take a partial flow of coolant from the reactor coolant system, run it through a purification system, and then reinject it. By means of the chemical feed system, and via the volume control system, the boron concentration to the reactor coolant system can be changed (chemical reactivity control). This is achieved by injecting boric acid or deionate (chemically pure water). The volume control and chemical injection systems together thus represent an independent, though slow-acting, shutdown system for the reactor.

In addition, the volume control system can be used as an auxiliary spray of the pressurizer during shutdown or failure of the normal pressurizer spray.

3.4.5 Control Features

It is important to the basic design of the control system during full-power operation of nuclear power plants that the generator and thus the turbine be adapted to the power requirements from the power main. Changes in power requirements are transferred via the turbine to the steam generators, and from this via the change in primary coolant temperature to the reactor. For an increase in power requirements by the mains, more heat must be withdrawn from the reactor coolant system via the heat genertors. This leads to a drop in primary coolant temperature. Since the power output of the reactor increases with decreasing primary coolant temperature, the reactor adapts itself to the changed power requirements. Pressurized water reactors thus have an inherently stable control behavior.

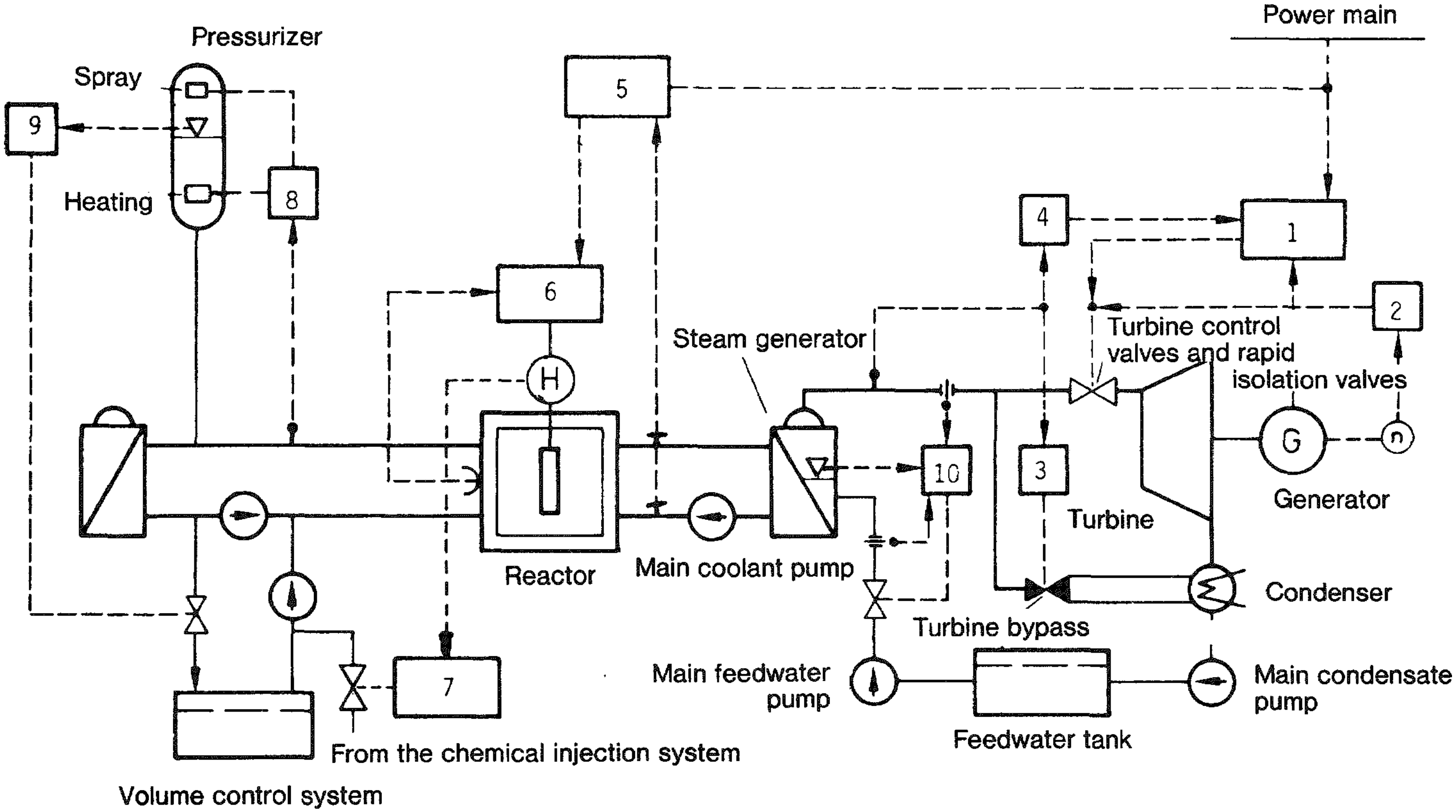
However, to keep the significant operating parameters within predetermined operating ranges under different power requirements and accidents, control mechanisms are necessary. The most important such mechanisms are (Figure 3-12):

- Turbine control
- Coolant temperature control with control rod bank position control system
- Coolant pressure control with pressurizer level control
- Feedwater control

Turbine Control

In normal operation, the nuclear power plant feeds its electric power to the interconnecting grid. During startup and shutdown of the turbine and isolated operation of the power plant, the turbine RPM is kept constant by the governor.

In order to keep the steam pressure at the turbine inlet from increasing in an unacceptable manner during turbine tripout or for load rejection (failure of power mains feed), a steam maximum-pressure controller diverts the excess steam through the main steam bypass directly into the condenser.



1 Turbine power control
2 Turbine RPM control
3 Main steam maximum pressure control
4 Main steam minimum pressure control
5 Coolant temperature control
6 Control rod control
7 Rod bank position control system
8 Coolant pressure control
9 Pressurizer water level control
10 Feedwater control

Flow monitor
Level measurement
Pressure measurement
Temperature measurement
Control rod positioning
RPM measurement
Neutron flux measurement

Figure 3-12. Overview of the control equipment of a PWR nuclear power plant

If the reactor power output cannot follow the demands of the power main, a main steam, minimum-pressure controller reduces the turbine load so much that a minimum main steam pressure is always maintained.

Coolant Temperature Control with Control Rod Bank Position Control System

This system is used to keep the average coolant temperature at a constant 306°C in a power range of between 66% and 100% of the rated power output. The control rods and the volume control and chemical injection system are used as control means. Rapid changes in reactivity are compensated by the control rods and by slowly changing the boron concentration.
The coolant temperature control drives the control rods via the control rod control. In order to achieve a favorable power distribution in the reactor core, the control rods are intermingled by means of the control rod bank position control system. To do this, the boron concentration of the coolant can also be changed.

In order to intercept accident-caused power surges before control measures trigger the reactor protection system, additional limiting features are provided. These features act particularly on the movement of control rods, they can also initiate control rod fast insertion.

Coolant Pressure Control with Pressurizer Level Control

The coolant pressure control is used during full power operation to keep coolant pressure to about 155 bar regardless of occurring accidents. Available control means are the pressurizer heating rods, spray valves, and two relief valves.

The pressurizer level control regulates the inlet and outlet quantity of coolant from the volume control system so that the pressurizer level remains constant.

Feedwater Control

To keep the steam water level within certain limits, the feedwater control adjusts the feedwater inflow to the steam quantity exiting from the turbine. The following control units are available:

- Main feedwater control "main load"
- Main feedwater control "low load"
- Emergency feedwater control

The control quantity is always the steam generator water level. Main load control is used at a reactor power output of more than 25%; at lower reactor output the secondary load control takes over. The control means are control valves in the four primary feedwater lines leading to the steam generators.

The emergency feedwater control maintains a minimum water level in the steam generator during failure or shutdown of the primary feedwater pumps and thus prevents vaporization of all water from the steam generator. Control means are control valves in the four emergency feedwater lines leading to the steam generators.

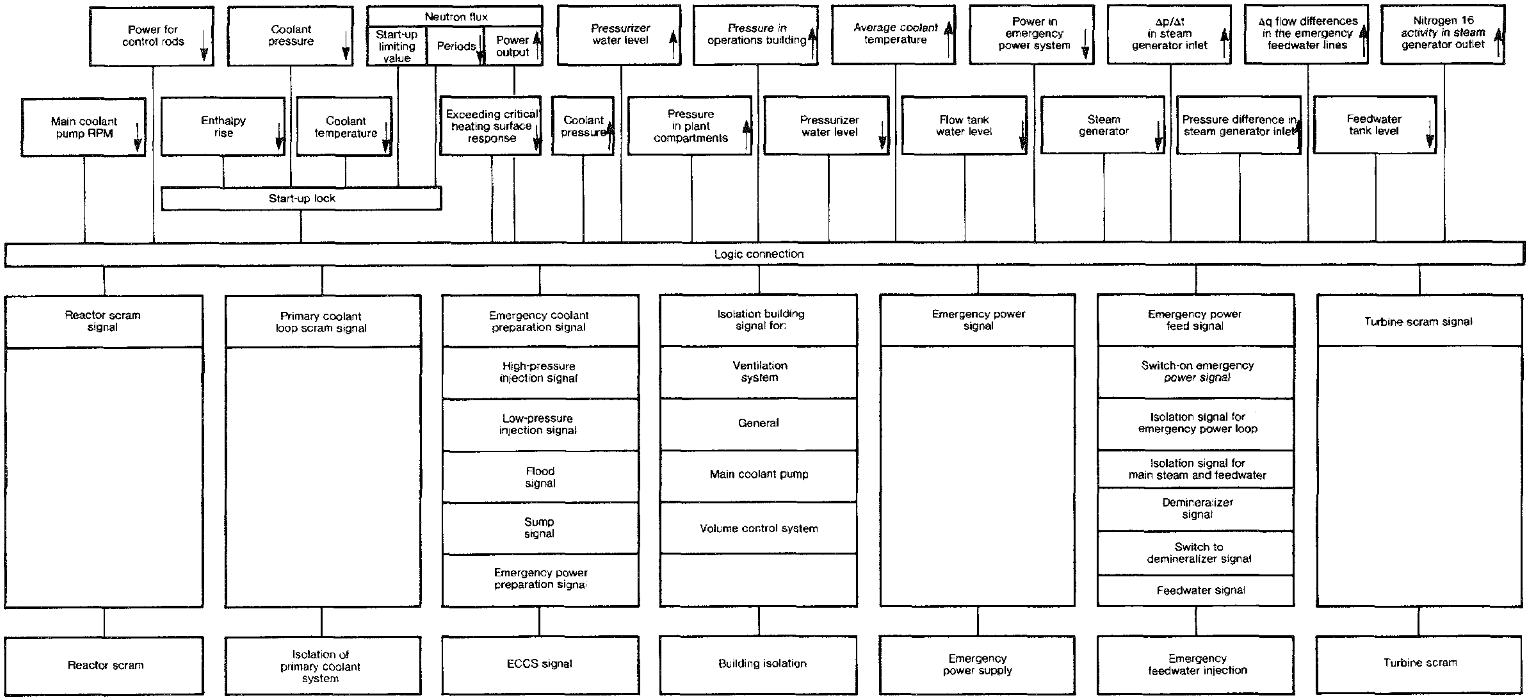
3.4.6 Reactor Safety System

The reactor protection system initiates necessary protective actions to ensure the safety of the reactor plant and environment by monitoring and processing important parameters during normal operation and during accidents. To initiate protective actions, reactor protection signals are formed that automatically trigger the appropriate safety systems. The following protective actions are triggered by the reactor protection system (Figure 3-13):

- Reactor scram
- Integrity of reactor coolant system
- Residual heat removal (emergency cooling and emergency feedwater supply)
- Isolation of the building
- Emergency power supply

The reactor protection system consists of partial systems for analog-measured value recording and setpoint signal indication (trigger levels), for logic evaluation and linkage (logic levels), and for initiation of triggering signals (control levels) (see Figures 3-14 and 3-15).

The trigger level includes the measurement channel groups for the different process variables (coolant pressure, coolant temperature, primary coolant pump,



Activation above (↑)
or below (↓) threshold

Δp/Δt in front of steam generator	=	Chronological pressure change in steam generator inlet
Δp in front of steam generator	=	Pressure difference in steam generator inlet
Δq emergency feedline	=	Flow differences in the emergency feedwater line
Steam generator level	=	Steam generator water level
Pressurizer level	=	Steam generator water level
DNB ratio	=	Exceeding critical heating surface response
FD	=	Main steam
HD	=	High pressures
HKP	=	Primary coolant pump
ND	=	Low pressures
N16 activity	=	Nitrogen 16 activity in steam generator outlet
EPBV	=	Voltage in emergency power busbars

Figure 3-13. Action in initiation criteria and response signals of the plant protection system

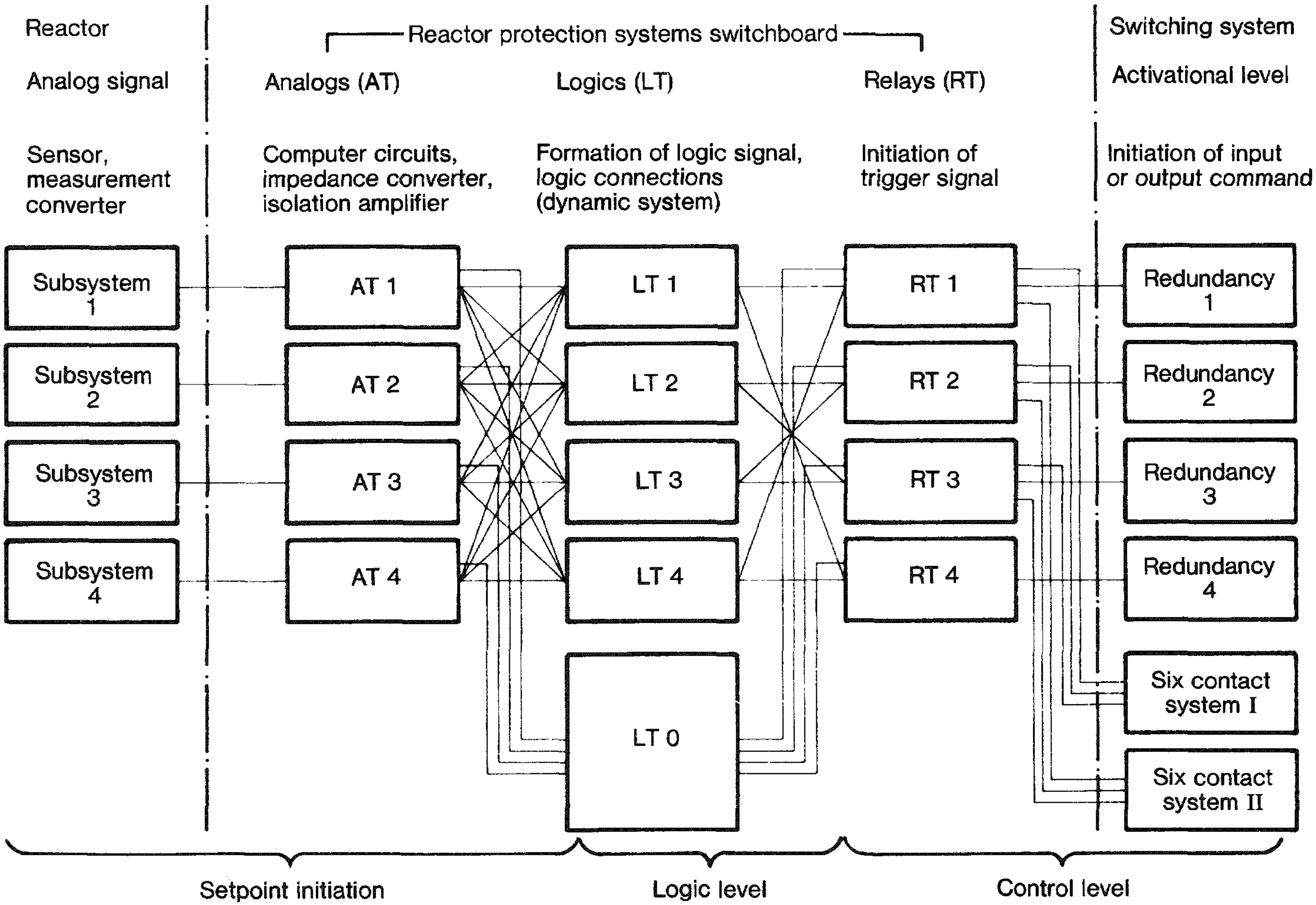


Figure 3-14. Schematic diagram of the reactor protection system

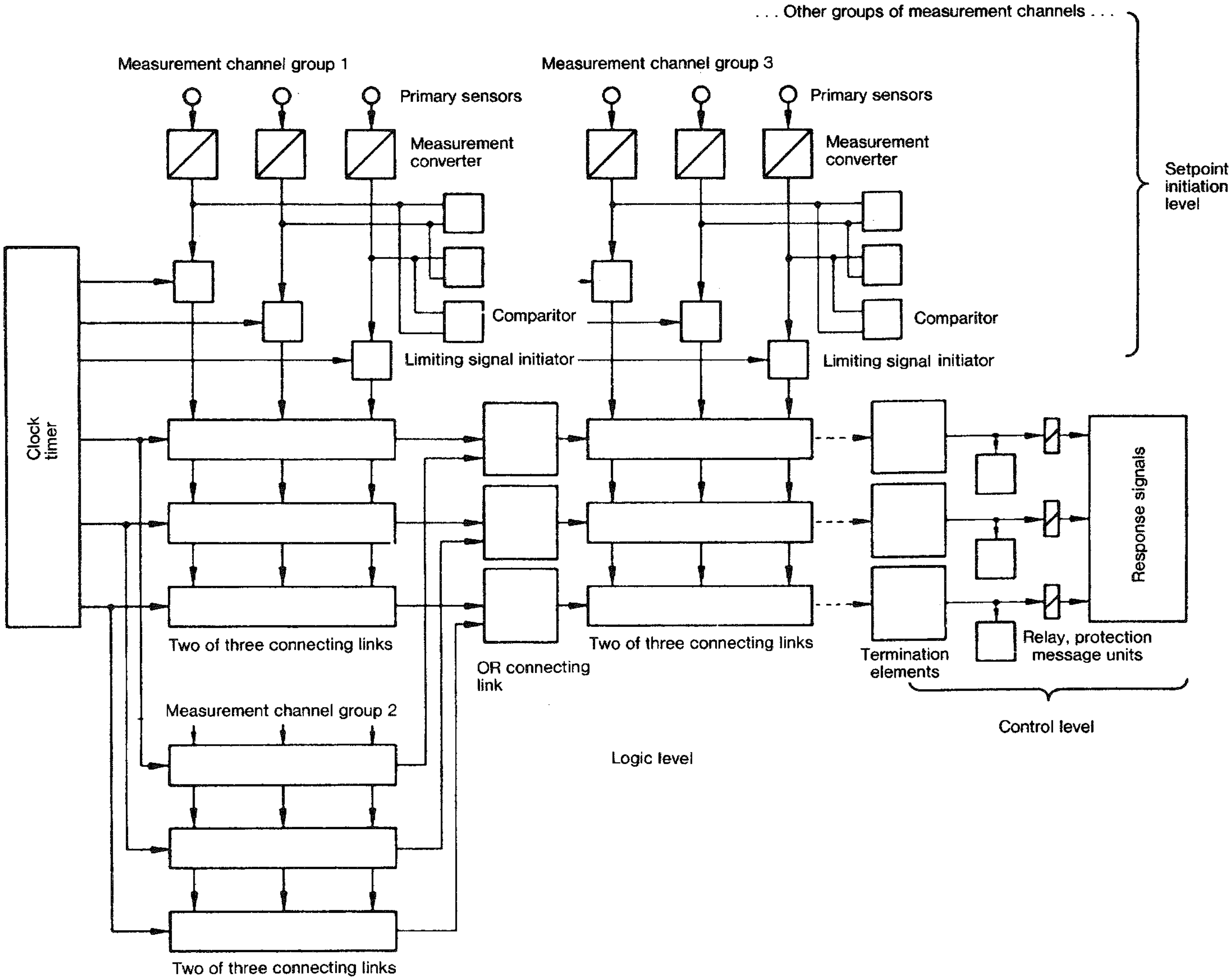


Figure 3-15. Block diagram of the plant protection system

RPM, etc.), which are measured at least three times. The individual measurements within a group are monitored for deviations by comparators. The setpoint signal transducers compare the process variables with limiting values. Any deviation beyond the values is signalled.

The logic level is that part of the reactor protection system in which the limit signals are logically linked together. If at least two of three limit signals of one measurement channel group are in a line, this is judged as exceeding the limit value.

The logic level operates according to a dynamic principle, i.e., with continuously transiting timing pulses. A defective failure of pulses is signal-triggering (fail-safe principle) and is self-indicating.

The control level of the reactor protection system is that part of the system where signals from the logic section are adapted to the switching conditions of the active safety mechanisms. The dynamic signals (timing pulses) arriving from the logic level are converted into static signals within electronic circuits (termination elements). These static signals activate relays that allow the plant protection signals to trigger either the six-contact system of the reactor scram or the activator (Figure 3-16).

In the activator, finally, "on" and "off" commands are given to the switching equipment of the individual components (e.g., pumps, valves). Predominance of reactor protection signals over other signals is assured.

The reactor protection system permits manual intervention by the operating crew in the station control room in only a few cases. In rare instances, such intervention is necessary for the function of the safety systems.

3.4.7 Electric Energy Supply

Figure 3-17 shows the electric diagram of the reference plant. During full-power operation the generator feeds electric energy into the interconnecting grid through the two mains feeds, which also include the machine transformers. By means of the 27-kV bus bar and the two auxiliary transformers, the 10-kV bus bars of the auxiliary switchgear are powered. Upon failure of the turbine or generator, the generator circuit is open. Supply to the auxiliary switchgear can be assumed without any interruption by the grid. If both main power supplies fail, then the turbine is shut down to station auxiliary power.

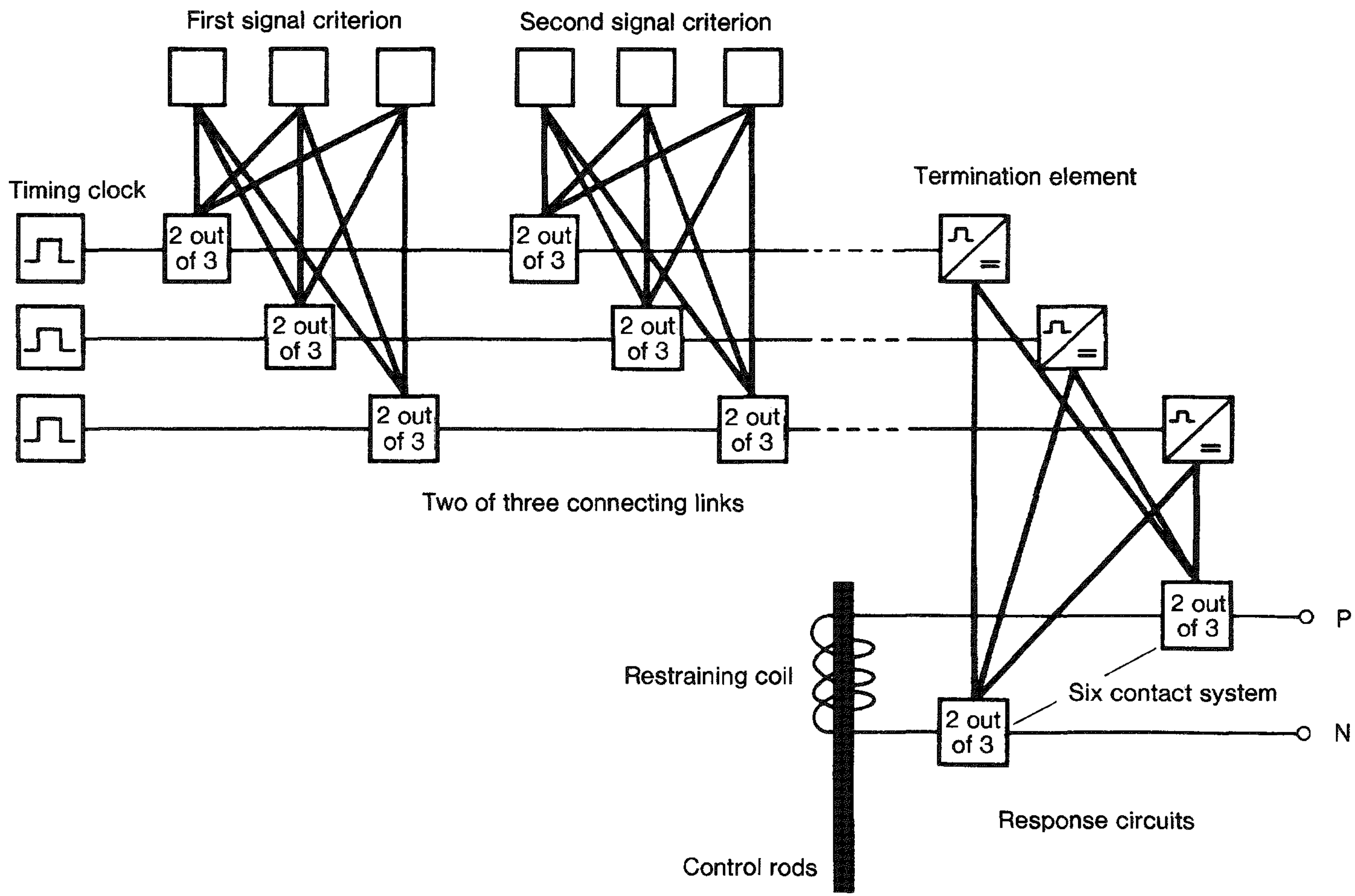


Figure 3-16. Design layout of the reactor protection system

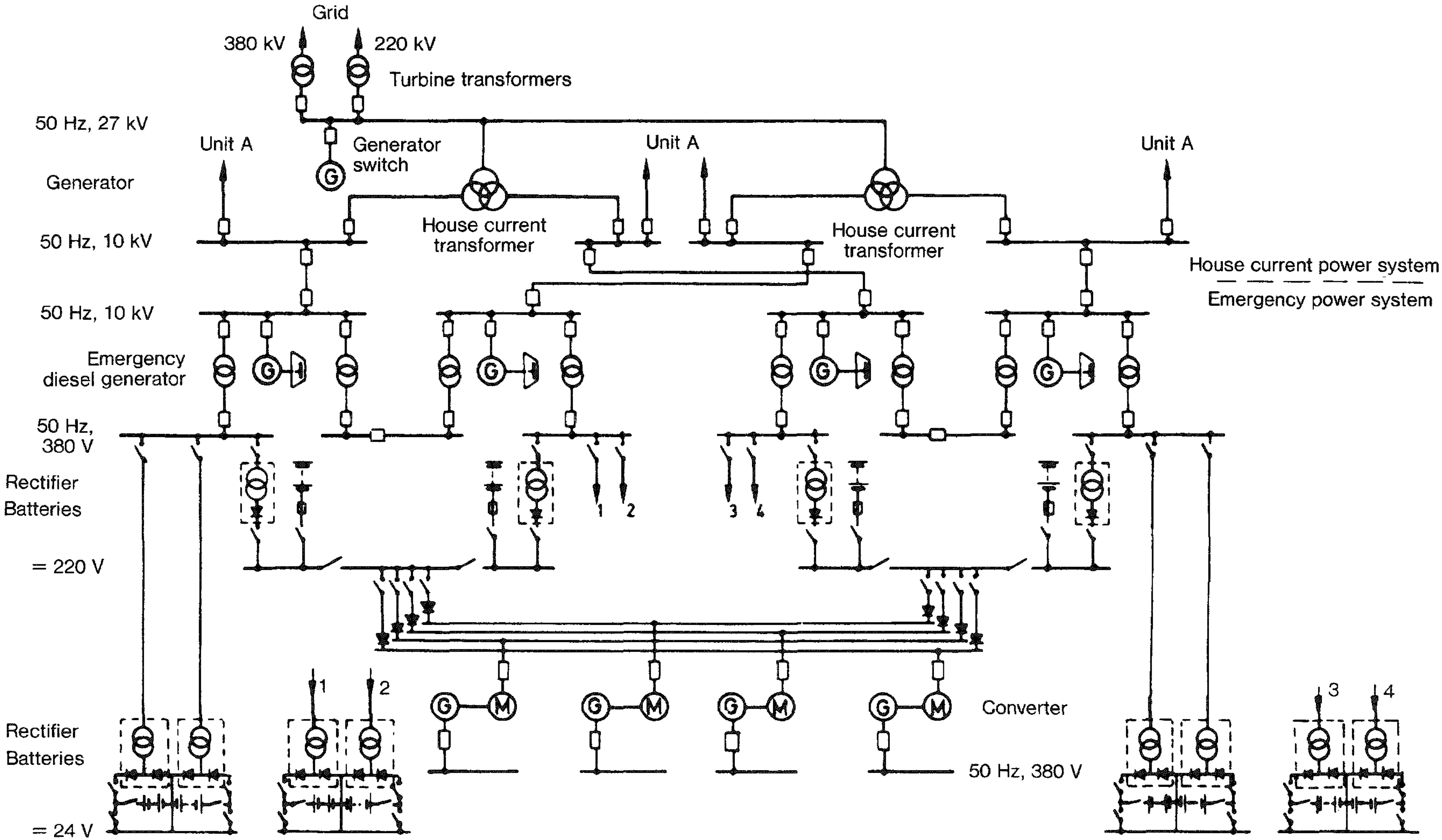


Figure 3-17. Electrical wiring diagram

Components relevant to safety are connected to the emergency power system. This is normally supplied with electricity from the auxiliary switchgear. Upon failure of the voltage to one of the four 10-kV emergency power bus bars, its connection to the auxiliary switchgear is terminated and the appropriate emergency diesel generator is started. After cutting out the 10-kV bus bars of the auxiliary switchgear, the 10-kV emergency bus bars can also draw energy through the connections to Unit A of the nuclear power plant.

In addition to the 10-kV bus bars, 380-V bus bars are present in the emergency power system. 220-V and 24-V d-c bus bars are powered via rectifiers from the 380-V emergency bus bars. By means of parallel circuited batteries a continuous power supply to the d-c bus bars is assured during a temporary voltage loss to the 380-V bus bars. For a continuous three-phase power supply, bus bars powered via transformers from the 220-V d-c bus bars are available. To operate the machine transformers, they must be cooled by the conventional secondary cooling water system. The emergency diesel generators must be cooled by the nuclear secondary cooling water system.

3.4.8 Emergency Feedwater System

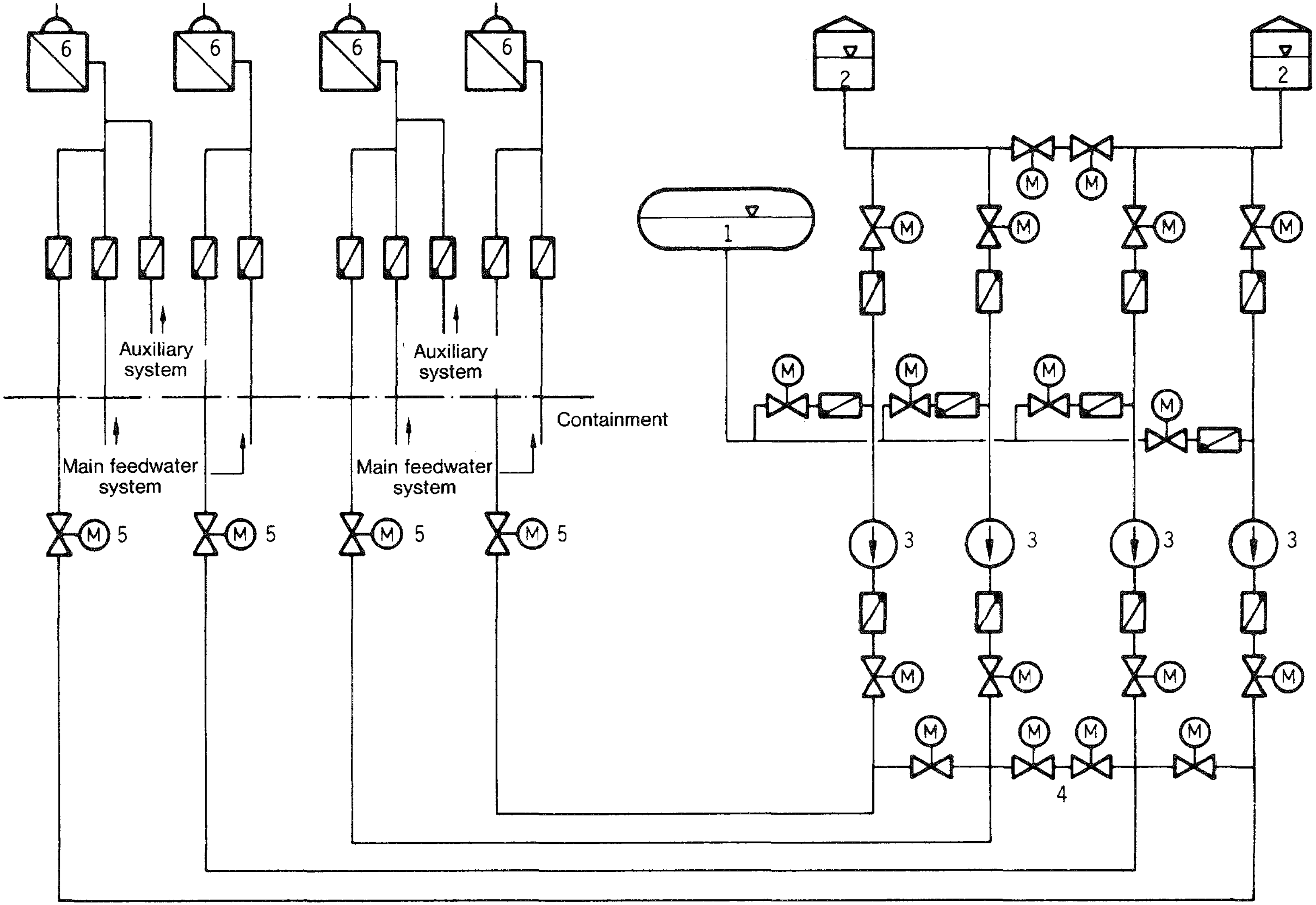
The four-loop emergency feedwater system (Figure 3-18) must supply the steam generators with water when the main feedwater pumps cannot do so. The emergency feedwater system and cooling systems needed for its operation are supplied with emergency power.

If the condenser is not available as a heat sink (e.g., in case of emergency power use), then the main steam is released to the atmosphere by means of the main steam safety valves or the relief control valves. In order to be able to supply the steam generators with water for a sufficient period of time, the emergency feedwater system has available water reserves that can maintain heat removal from the reactor coolant system for ten to fifteen hours.

3.4.9 Emergency Cooling and Residual Heat Removal System

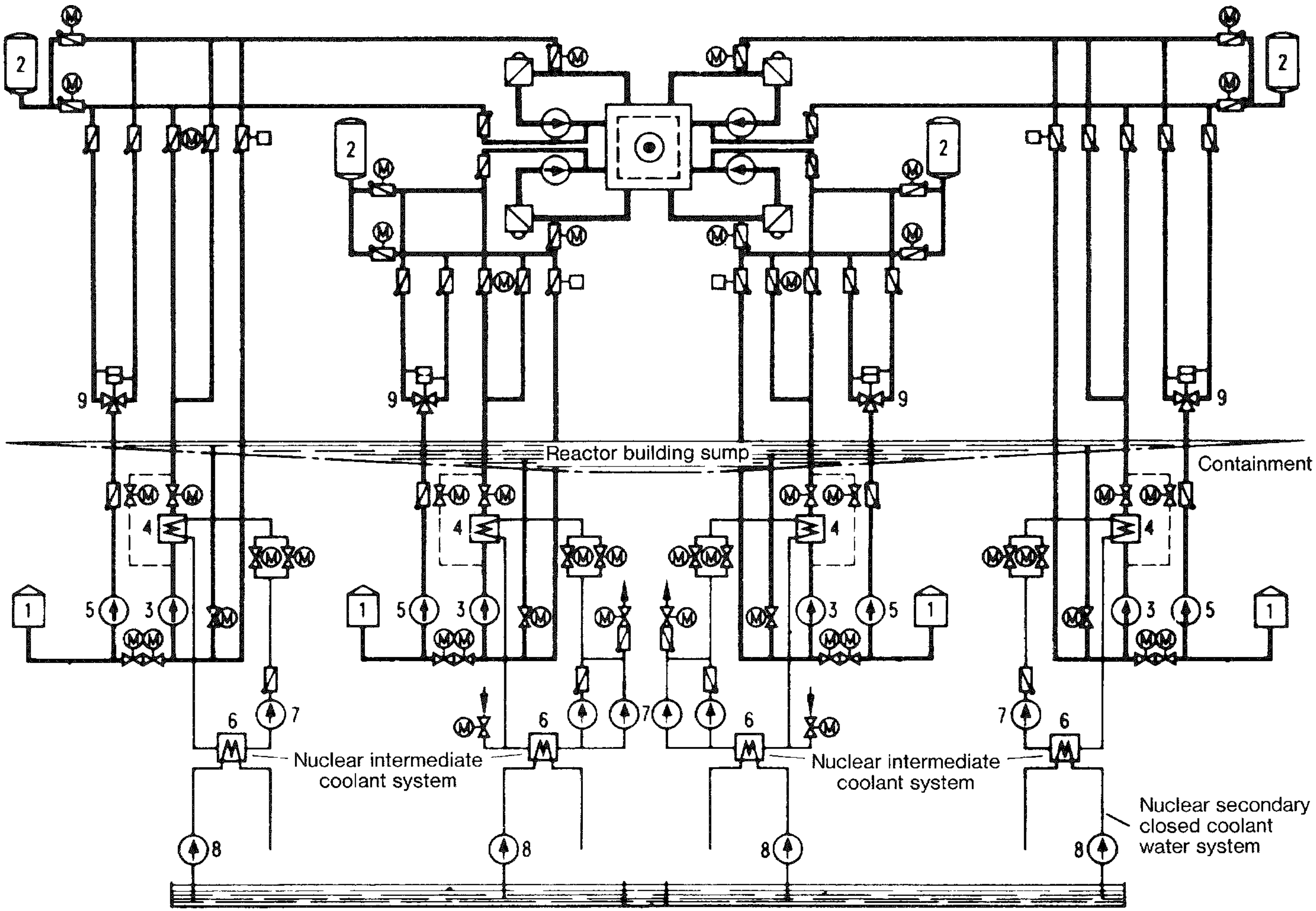
The emergency cooling and residual heat removal system (Figure 3-19) has both operational and safety functions:

- When shutting down the power plant, the emergency cooling and residual heat removal system begins operation when the pressure and temperature in the reactor coolant system have been reduced sufficiently. It then assumes the function of removing



1 Feedwater tank
2 Demineralizer tank
3 Auxiliary feedwater pump
4 Auxiliary feedwater collection line
5 Auxiliary feedwater control valve
6 Steam generator

Figure 3-18. Auxiliary feedwater system



1 Flood tank
2 Accumulator
3 RHR pump
4 RHR cooler
5 Safety injection pump
6 Intermediate heat exchanger
7 Intermediate coolant water pump
8 Secondary coolant water pump
9 Three-way valve

Figure 3-19. Residual heat removal system residual heat occurring in the reactor core and of further cooling the reactor coolant system (residual heat removal).

- During accidents that lead to loss of coolant from the reactor coolant system, the job of the emergency cooling and residual heat removal system is to refill the reactor pressure vessel and to assure sufficient cooling of the reactor core (emergency cooling).

In order to perform these tasks, the emergency cooling and residual heat removal system has the following subsystems:

- High-pressure injection systems with safety feed pumps
- Low-pressure injection system with RHR pumps
- Accumulators

The emergency cooling and residual heat removal system is composed of four loops and is supplied with emergency power. The injection loops of the system are connected to the primary coolant hot and cold legs. The RHR pumps and accumulators simultaneously feed the hot and cold primary coolant pipelines. The safety injection pumps are circuited over three-way valves so that they feed only into the cold, primary coolant pipelines. During a leak in a cold primary coolant pipeline, the three-way valve switches the appropriate safety feed pump to the hot primary coolant pipeline.

During LOCAs, the operation of the emergency cooling and residual heat removal system differs according to the size of the leak.

For large-break sizes, the pressure in the reactor cooling loop drops quickly. The high-pressure injection systems are not needed in this case. The accumulators and the low-pressure injection systems feed borated water into the reactor coolant system. The RHR pumps initially remove water from the storage tanks (flood operation). If the storage tanks are emptied, then the RHR pumps pull the water collected at the lower part of the containment (reactor building sump) upward and deliver it back to the reactor coolant system through the residual heat exchanger (RHR, sump circulation operation).

For smaller leak sizes, the pressure in the reactor cooling loop drops slowly. Therefore, the high-pressure injection systems can be used first. Below a certain leak size, heat must simultaneously be removed via the steam generators. Coolant pressure and temperature in the reactor cooling loop must be reduced by the feedwater-steam system so that the low-pressure injection system can be taken into operation.

Operation of the hooked-up cooling chain assures emergency cooling. Comprised of four loops and powered by emergency current, this cooling chain consists of a nuclear intermediate cooling circuit and the nuclear secondary cooling water system. The nuclear intermediate cooling circuit picks up the heat from the residual heat coolers of the emergency cooling and residual heat removal system and transfers it through another heat exchanger to the secondary cooling water system, which is cooled by river water. Intermediate switching of the nuclear intermediate cooling circuit between the emergency cooling and RHR system and the secondary cooling water system assures that no radioactive substances can get into the river during leaks in the fuel element cladding and in the heat exchanger of the emergency and RHR system.

3.4.10 Emergency System

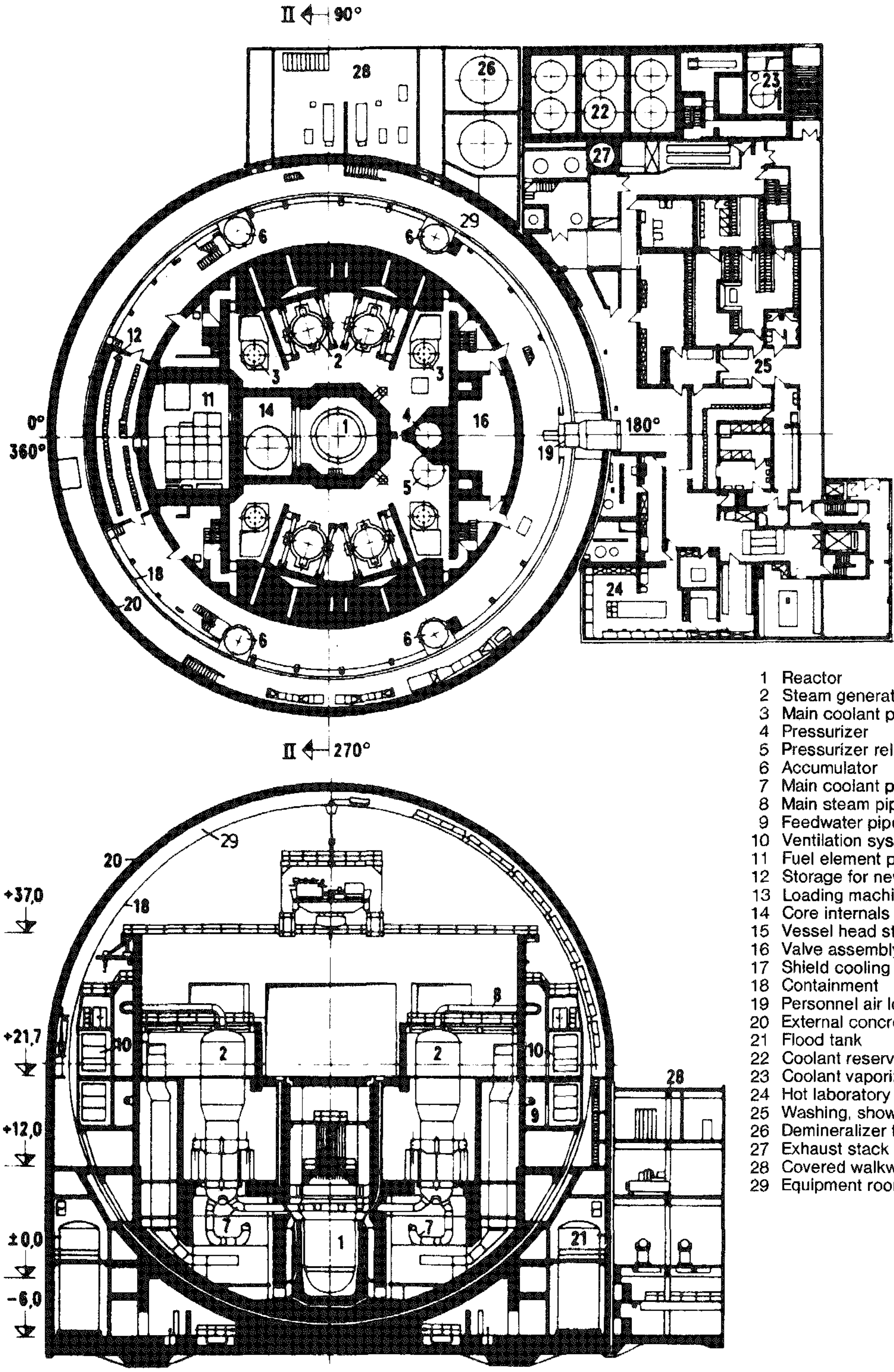
The emergency system is used primarily to transfer the plant to a safe status after damage has occurred as anticipated due to external events. To be effective, the emergency system must assure heat removal of the shut down reactor.

From the intact region of unit A, it is possible to control important components in the damaged unit B. By open-circuiting appropriate pipelines, two steam generators in unit B can be supplied with emergency feedwater from unit A (see the connections to the emergency system in Figure 3-18). Borated water from unit A can be injected into the reactor coolant system. In addition, unit A can take over the power supply to unit B.

3.4.11 Containment and Annulus Exhaust Air Handling System

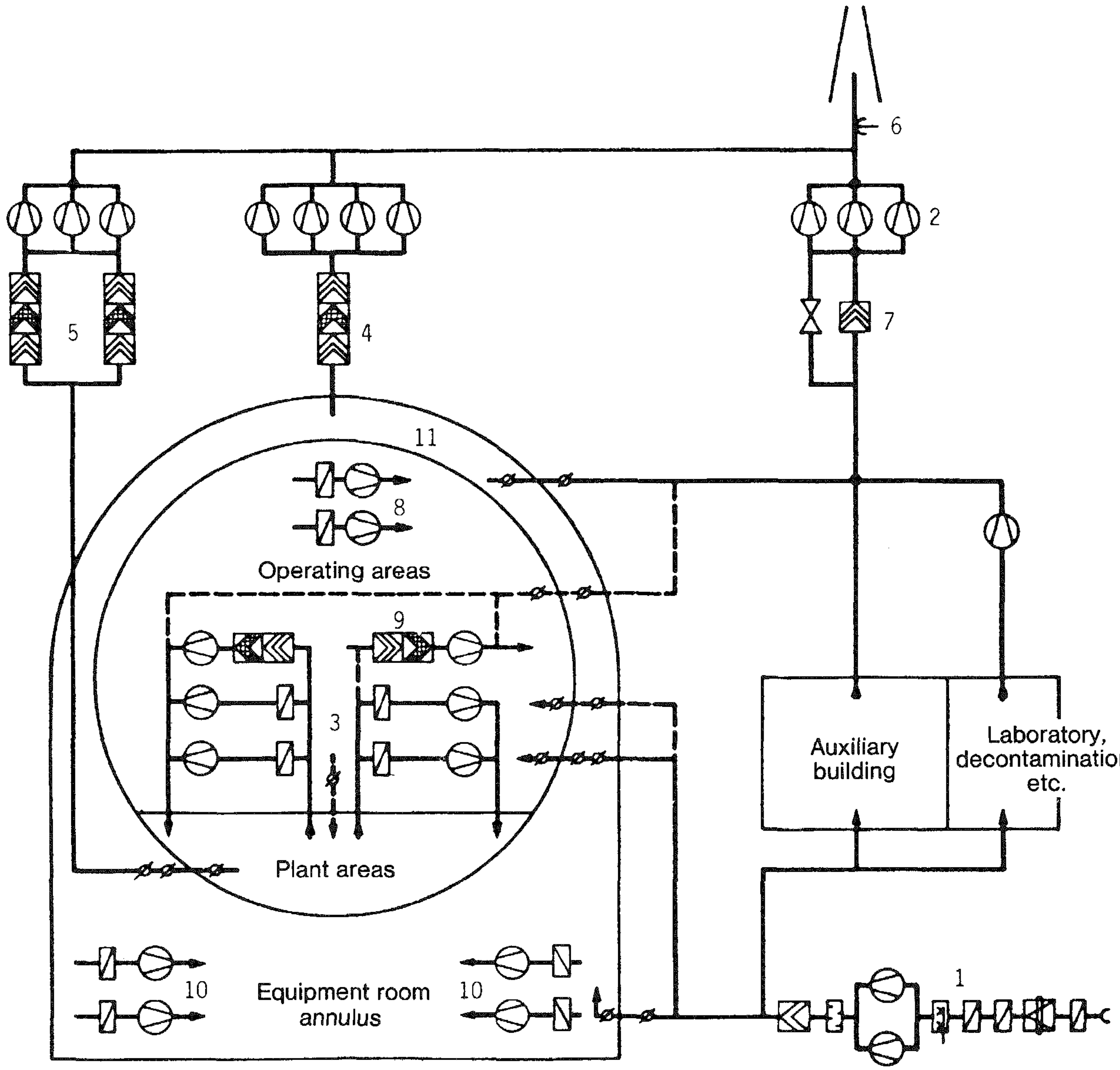
The nuclear power plant containment, which retains radioactivity during accidents, consists of a steel, spherical vessel and an external reinforced concrete structural shell (the secondary containment), with an exhaust air handling system for the annulus lying in between (Figures 3-20 and 3-21).

The steel vessel contains a number of pipeline and cable penetrations. Needed primarily to operate the systems located within the containment, these penetrations are gas-tight and pressure resistant. Highly stressed penetrations are additionally sectioned off and connected to a leak-off system. Each pipeline



1 Reactor
2 Steam generator
3 Main coolant pump
4 Pressurizer
5 Pressurizer relief tank
6 Accumulator
7 Main coolant pipeline
8 Main steam pipeline
9 Feedwater pipeline
10 Ventilation system
11 Fuel element pool
12 Storage for new fuel elements
13 Loading machine
14 Core internals letdown pool
15 Vessel head storage ring
16 Valve assembly area
17 Shield cooling air fan compartment
18 Containment
19 Personnel air lock
20 External concrete shielding
21 Flood tank
22 Coolant reservoir
23 Coolant vaporization system
24 Hot laboratory
25 Washing, shower and clothes changing area
26 Demineralizer tank
27 Exhaust stack
28 Covered walkway
29 Equipment room annulus

Figure 3-20. Containment with internals and auxiliary reactor building



1 Air intake
2 Exhaust air
3 Air circulating system to plant compartment
4 Annulus exhaust air handling system
5 Depressurizer for plant compartments
6 Radioactivity monitoring
7 Vent air filter
8 Ventilation system to operating compartment
9 Filter system to operating compartment
10 Annulua air recirculation system
11 Annulus

Figure 3-21. Ventilation system

leading through the containment can be isolated by at least two valves, and the subatmospheric pressure system lines can be isolated by three valves, one behind the other.

The reinforced concrete structure encloses the containment and forms the outer wall of the reactor building (Figure 3-20). It protects the reactor system against external events and shields the environment against direct radiation from the containment during accidents.

During an accident the reactor protection system closes the building isolation valve. Thus, all pipeline penetrations not needed to control the accident are automatically closed off. The annulus between the containment and the reinforced concrete structure is kept at subatmospheric pressure by means of the annulus exhaust air handling system. Thus, radioactivity release due to smaller leaks from the containment can be detected, monitored, and released through filters and stacks.

3.4.12 Protection Against Fire

Fires within the plant cannot be precluded in advance. Combustible materials like lubricating oils as well as other potential ignition sources are always present. Overall, the quantities of existing combustible materials are small.

In addition to precautions against the occurrence of a fire, measures are taken to prevent the spread of fire and thus the potential failure of several important safety systems. For instance, redundant loops of safety systems are either spatially separated so that they cannot be simultaneously affected by heat and smoke, or they are sectioned off by structural or fireproofing materials; this is also true for the cable connections belonging to the particular redundant loops. Cable and pipeline penetrations through structure sections have fire resistant seals.

In addition to these passive measures, active fire prevention measures include a fire reporting and extinguishing system that encompasses the entire plant. The controlled-access area and plant areas in which the safety features are located are also equipped with instruments for early fire detection. Sections of the building containing a concentration of flammable material, e.g., masses of cables or larger oil containers, have additional stationary, fast-acting extinguishing features. The technical-structural fire prevention measures are supplemented by administrative procedures.

3.4.13 Protection Against External Events

In accordance with the existing German safety requirements, nuclear power plants in the FRG are to be protected against external events. Existing safety features must also be able, in this case, to shut down the plant, to remove arising heat, and to prevent an unacceptable release of radioactive substances. However, after an accident due to external events, the plant need not be able to continue operation.

The reference plant was designed to withstand external events in accordance with regulations and requirements in effect at the time of licensing. These requirements have been modified in the meantime.

The design of the reference plant was based both on natural and man-made effects. Natural effects included:

- earthquake
- flood
- storm
- lightning strike

Man-made events were:

- aircraft crash
- shock waves from chemical explosions
- intervention of explosive and poisonous gases
- intervention by third parties

The concept of protection against external events combines design and technical measures--primarily, appropriate design of the most important system parts to the corresponding loading. To protect against external events that can lead to locally limited damage, the most important system components are also spatially separated. Additional administrative and organizational procedures are provided.

The protective concept used in the reference plant provides for coping with flood, earthquake, severe weather, explosion shockwaves, and ingress of harmful gases by means of the safety features installed in unit B. For aircraft crash and in part also for interventions by third parties, the emergency system provides an added safety feature.

FOOTNOTES

(a) The following discussion must necessarily be very compressed. For the sake of brevity, the description is based on the book Reaktorsicherheitstechnik [Reactor Safety Technology] by D. Smidt, Springer Publications.

(b) In addition, initially nonradioactive materials are made radioactive by irradiation. The total activity of these activation products is low in comparison with the fission products. The activation products are therefore not discussed separately.

(c) Translation Note: In the remainder of this report the English translation will normally not distinguish between design basis and Class 9 accidents.

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Section 4

THE OBJECT AND METHODS OF THE RISK ANALYSIS

4.1 OBJECT OF THE RISK ANALYSIS

Nuclear power plants contain considerable quantitites of radioactive material. For instance, the radioactive inventory of the Biblis B nuclear power plant is about 10^{10} Curies after a long operating time (a). If even a small fraction of the fission products escaped into the environment, danger would threaten both health and life. A nuclear power plant therefore poses a significant hazard potential.

In order to master this potential hazard, nuclear power plants are designed so that release of radioactivity on a level dangerous to humans is precluded within limits of the state of the art. In addition to retention of fission products by several redundant structures, extensive protective and safety precautions prevent damage to these structures (see Chapter 3).

All previous experience has shown that this safety concept has thoroughly proven itself. Over a period of about 25 years--at least in the Western world--no one has been killed by radioactive releases from a nuclear power plant or has had his health damaged in any demonstrable form. On the other hand, it is obvious that in spite of extensive precautions, accidents with the potential to considerably damage the environment can never be precluded with absolute certainty.

A useful, quantitative statement about the "remaining uncertainty" or, in other words, "the risk," has not been empirically found for nuclear power plants. Experiences that would permit a necessary statistical determination and evaluation of accidents are not available. Therefore, the risk can only be estimated by analytical means.

During accidents that are handled by the design of safety systems, no damage occurs outside the plant. Therefore, a contribution to risk is only expected if, during such an accident, the engineered safety features fail in such a manner that a considerable release of fission products from the plant occurs.

The risk analysis therefore concentrates primarily on events for which a failure of the safety systems is postulated. For a numerical determination of the risk, both the frequency as well as the consequences of such events must be determined in a theoretical manner.

In order to give an overview of what events could be decisive to the risk, Section 4.2 describes the sequence of reactor accidents as considered in the risk study. Section 4.3 provides an overview of procedures and methods of risk analysis. In Sections 4.4 and 4.5 the event tree analysis and fault tree analysis used within the framework of the technical studies are discussed. The reliability of data used for a quantitative evaluation of these analyses is discussed in Section 4.6. Section 4.7 discusses uncertainties connected with the risk analysis.

4.2 DESCRIPTION OF ACCIDENT SEQUENCES TO BE STUDIED FOR THE ANALYSIS

For an analytical determination of risk, model conceptions should be developed for processes occurring during a reactor accident both within and without the plant. The considerations must concentrate on those accident sequences that could lead to a relatively large release of radioactive substances from the plant and to potential damage to the environment.

The basic conceptions on the sequence of a reactor accident are described below; these are used as a basis for the methodical procedure of the study outlined in Section 4.3.

Activity Inventory in the Plant

In the first step, the location and quantity of radioactive material in the plant must be determined. Table 4-1 shows the absolute and relative fractions of the activity inventory at various points in the nuclear power plant. These values depend on different parameters (e.g., burn-up, decay time, number of fuel elements in the storage pool, operating mode of auxiliary systems) and vary with time. The table contains typical values, which are sufficiently accurate for the discussions at this point.

We find on the average 95% of the total radioactive inventory in the reactor core (including reactor coolant system). Shortly after refueling, this fraction can drop to about 80%.

Table 4-1. Typical radioactive inventory of a PWR nuclear power plant (1300 MWe)

Location	Total activity (in Curies)			Percentage of core inventory		
	Fuel	Fission gas Plenum	Total	Fuel	Fission gas Plenum	Total
Reactor core (a)	6.3×10^9	1.0×10^8	6.4×10^9	98.4	1.6	100
Fuel element storage pool (maximum) (b)	1.3×10^9	2.1×10^7	1.3×10^9	20.6	0.3	21
Fuel element storage pool (average) (c)	3.3×10^8	5.3×10^6	3.3×10^8	5.1	8×10^{-2}	5.2
Transfer cask (d)	1.7×10^7	2.7×10^5	1.7×10^7	0.3	4×10^{-3}	0.3
Off-gas system	--	--	1.5×10^4	--	--	2×10^{-4}
Waste water system (e)	--	--	1.2×10^3	--	--	2×10^{-5}
Ion exchanger (f)	--	--	1.5×10^4	--	--	2×10^{-4}
Other components in the auxiliary building (g)	--	--	1.2×10^3	--	--	2×10^{-5}

(a) Data for a time point about one-half hour after shutdown after an average burnup of 10,000/19,600/33,500 MWd/t (3 regions in the core).

(b) Inventory of 2/3 core loading, of which half has decayed for three days, and another half for 180 days.

(c) Inventory of 1/2 core loading of which 1/3 has a 180-day decay period and 2/3 a 50-day decay period.

(d) Corresponds to 10 fuel elements after 180-day decay period.

(e) Contains: concentrate tank (30-day decay period), vaporizor for waste water, waste water collection tank.

(f) For a purification rate of the main coolant of 10% per hour and a dwell time of about half a year.

(g) Contains: filter (resin trap), resin waste tank, boric acid tank, volume compensation tank, coolant reservoir, vaporizor for coolant, blowdown salt removal.

The remaining 5% of the total radioactive inventory is located almost exclusively in the storage pool for spent fuel elements. In the (loaded) fuel element cask and in auxiliary systems (e.g., off-gas system, waste water system), the radioactive inventories are very small compared with the core inventory. Under consideration of safety precautions taken in these system parts, it is therefore assumed that no significant contribution to risk is expected by their failure. Consequently, the study concentrates on possible releases from the reactor core.

Retention of Fission Products

With intact systems, the fission products arising in the reactor core are retained by several structures (see Section 3.3.1).

In addition to the "internal structures" (crystal lattice of the fuel, fuel rod cladding), which practically retain the fission products at the point of their generation, other structures, the "external structures" (reactor coolant system, containment) are available. Upon failure of the reactor coolant system or containment, the radioactive releases remain small as long as the fuel cladding and crystal lattice of the fuel remain generally intact.

Therefore, we must track primarily those events that can lead to failure of the internal structures. Subsequently, we shall examine what the consequences could be with regard to the external structures.

Failure of Fission Product Retention

According to Table 4-1, about 98% of the total radioactive inventory of the reactor core is retained in the crystal lattice of the fuel. The remaining 2% (with the exception of minor fractions that may escape through leaks in clad tubes into the reactor coolant system) is retained by the fuel cladding.

The majority of fission products thus can be released only if the fuel is overheated and, especially, if the crystal lattice of the fuel is dissolved, i.e., when the fuel melts. Even for a complete fuel melt, however, depending on the physical-chemical properties of the various fission products, various fractions would remain in the fuel melt (see also Sections 6.5).

To determine risk, therefore, we must track those events that can lead to a core melt. With respect to the above assessment of risk, it is assumed in this study that the fuel always melts completely when the core is insufficiently cooled.

Section 3.4 describes in detail which precautions are taken and which protective systems and safety features are present to assure satisfactory cooling of the reactor core, even for all considered operational malfunctions and accidents. For an assessment of risk, we must determine the probability and the circumstances that will allow accidents to result in a core melt in spite of these safety precautions. The sequence of such events need not be analyzed in all details. We are interested primarily in two questions:

1. Which safety systems (or how many loops of redundant systems) are needed to prevent a core melt?

   To answer this question the study adopts the appropriate prescriptions from the licensing procedure for the reference plant or other comparable plants, where available.

   Thus, minimum requirements of the safety systems can be established, and these can be included in the reliability analysis of these systems (see Table 5-1).

2. What is the status of the plant upon occurrence of a core melt?

   From this we obtain initial and boundary conditions for simulation of processes during a core melt.

   Rough data are sufficient here, and these can be obtained from estimations. Previously, only relatively simple models were available to simulate core melt processes and the subsequent accident sequences. Therefore, in this study, we do not distinguish between a partial and complete core melt. For each accident sequence for which a minimum number of necessary safety systems is unavailable, a pessimistic, complete core melt is assumed.

Below, we provide an overview of processes to be analyzed in a core melt accident in order to determine consequences within the plant and on the environment.

Processes During a Core Melt

If reactor core cooling fails--for example, due to a large leak in the reactor coolant system and simultaneous failure of the emergency cooling system--then the fuel heats up the reactor core as a result of the residual heat and causes the water in the reactor pressure vessel to vaporize. The steam flows through the leak into the containment.

It is assumed that the fuel cladding fails. The volatile fission products largely move from the clad tubes into the reactor pressure vessel and from there into the containment.

Once the fuel is heated to melting temperature, considerable amounts of fission products normally retained in the crystal lattice are released and can enter the containment through the leak in the reactor coolant system.

It is then of decisive importance for the extent of fission product release to the environment to know whether the containment remains sealed. Therefore, the potential effects of a core melt on the containment must be studied.

Behavior of the Containment

When the fuel melts, the core support structures also fail. The molten fuel rods collapse, together with the molten structure materials in the lower hemisphere region of the reactor pressure vessel (RPV).

It is assumed that the residual heat in the core melt is sufficient to melt through the bottom of the RPV and possibly also the concrete structures underneath.

The energy from the reactor core and from the core melt moves through different, sometimes simultaneous processes into the atmosphere of the containment and there causes an increase in temperature and pressure. Of primary importance are:

- vaporization of residual water in the RPV
- an exothermic, i.e., energy liberating, chemical reaction between metal structures and steam (metal-water reaction)
- steam generation upon contact of core melt and sump water
- vaporization of water liberated during the melt of concrete.

For the extent of accident consequences it is important to know whether and at what time the pressure or temperature in the containment will cause the steel enclosure to fail.

In the metal-water reaction and--to a lesser extent, by radiolytic decomposition of water--hydrogen is generated. This hydrogen contributes to the pressure increase in the containment, especially if it burns continuously. In addition, explosive hydrogen-oxygen mixtures can form if a hydrogen-rich mixture can accumulate.

If the hot core melt suddenly comes into contact with water, very rapid steam generation can occur under certain circumstances. Such reactions, if spontaneous, are called steam explosions. The extent to which this type of process must be included in the considerations is discussed in Section 6.4.

For accidents that lead to an increase in pressure in the containment, the containment isolation (see Section 3.4.11) is triggered by the reactor protection system. Thus, all penetrations through the steel enclosure are closed, provided they are not needed to help cope with the accident. If the containment isolation fails, fission products can move through leaks in the containment to the environment.

Release of Fission Products

The atmosphere in the containment consists of a mixture of water vapor with various gases (primarily oxygen, nitrogen, and hydrogen) during a core melt. The quantity of radioactive gases and aerosols is small, but it is also decisive for the potential consequences of a core melt accident, provided such radioactive products get into the containment upon failure of the cladding and melting of the fuel.

The radioactivity content in the containment atmosphere is reduced over time by condensation and natural deposition processes and by radioactive decay, especially of the short-lived nuclides. Overpressure in the containment simultaneously causes the steam-gas mixture and radioactive substances connected with it in the exhaust air handling annulus to flow from there to the environment if the steel liner has been damaged.

Propagation and Effects of Radioactive Substances in the Environment

The radioactive cloud formed by the released mixture of vapors, gases, and aerosols is carried away from the plant by the wind. The energy of the cloud can also cause a thermal lift. The initial relatively compact plume spreads out at an angle to the wind direction as a result of turbulent diffusion. As distance from the power plant increases, a broader region is covered by the cloud. The dilution of the cloud which takes place at the same time, as well as fallout and--if rainfall occurs--the scavenging of radioactive substances--decrease the radioactive concentration in the cloud. As a result of deposition and scavenging, the region covered by the plume is radioactively contaminated.

People residing in this region can be exposed by direct radiation and inhalation of radioactive substances from the plume and from fallout, and also by the consumption of radioactive substances in food. The level of this radiation exposure, the number of affected persons, and thus the potential consequences of various types depend not only on the expected radioactive concentration, but also on the implementation and effectiveness of emergency protection measures.

4.3 METHODS OF RISK ANALYSIS

This chapter illustrates methods for examination of accident sequences and determination of their frequency and potential consequences.

Figure 4-1 is an overview of the most important steps of the study:

- determination of initiating events
- event sequence and reliability analyses
- determination of radioactive release
- calculation of accident consequences
- assessment of risk.

The first three stages include the examination of processes within the plant (technical system analysis). Proceeding from the results of these investigations, the potential consequences of radioactive release outside the plant are studied in the fourth step.

Determination of Initiating Events

For the technical system analysis in the first step, all important "initiating events" which could, under certain circumstances, result in radioactive release to the environment are determined by type and frequency.

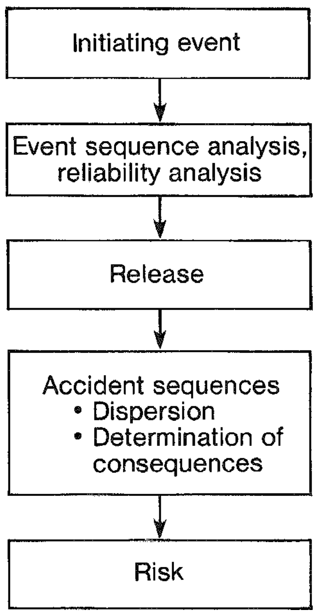


Figure 4-1. Steps in the risk analysis

It is neither possible nor necessary to present and analyze in detail all possible initiating events. It is sufficient to discuss a limited number of classes of them that will cover all other possible initiating events.

An argument for the initiating events discussed in the study is given in Chapter 5.

Event Tree and Reliability Analysis

Proceeding from an initiating event, different event sequences result, depending on the success or failure of the particular safety systems. In order to obtain an overview on the large number of potential sequences, event trees are prepared.

Next, the frequencies of the event trees are to be determined. In addition to the frequency of initiating events, the failure probabilities of systems needed to cope with the accident must also be determined. The necessary reliability studies to make this determination are implemented primarily by means of fault tree analysis.

The methods of event tree and fault tree analysis are discussed in detail in the following chapters.

The first two steps of plant system analysis are used primarily to determine the frequency of a core melt. In addition, from the event tree diagrams and from a simulation of accidents, information can be derived about the physical state of the plant before the beginning of the core melt.

Determination of Radioactive Release

In this step, the sequence of core melt accidents is followed, initially within the plant itself; the objective is to determine the radioactivity release to the environment. The following items are treated:

- processes during melt of the reactor core and the behavior of the molten core
- behavior of the containment and its possible failure modes
- fission product transport into the and release from the containment.

The core melt calculations were performed with the BOIL computer program used in WASH-1400. The results of these calculations provide the starting values for studies on fission product transport and containment loading.

To calculate the load to which the containment is exposed in a core melt accident, the CONDRU program is used. This computer program, applied in the licensing procedure, was expanded to the study. The transport and deposition processes--which are decisive for the behavior of fission products--were studied using the CORRAL computer program from WASH-1400.

Since several basically different processes can lead to containment failure, not only dynamic processes must be simulated, but the probabilities of the different failure modes must also be determined.

As a final result of the plant system analysis, we obtain the type (amount, location, time history, energy carry-over) and frequency of radioactive release from the plant. The release of radioactivity for the different accident sequences can be compiled into a series of representative releases--the release categories.

The models used to determine the radioactive release and the results of studies performed in this regard are presented in Chapter 6.

<u>Calculation of Accident Sequences</u>

Calculation of accident sequences also takes place in several stages. These are described in detail in Chapter 7 and are therefore only summarized here.

First, the weather-dependent dispersion of radioactive clouds is simulated. This yields the location and time-dependent radioactivity concentrations in the environment of the plant.

Next, the radiation doses resulting from this and the number of affected persons is determined. The influence of emergency protective measures is taken into account; these measures are provided for by official planning in the event of an accident.

Finally, it is determined to what extent health injury of various types can occur due to the calculated radiation exposures.

Several chance parameters are important to the extent of the accident consequences. Primarily these are: weather situation, precipitation, and wind direction and velocity prevailing during and after an accident which are decisive for dispersion of the radioactive substance. Information on the extent of damage is therefore dependent on frequencies that result from the probability of these chance parameters and the frequency of the particular release category.

Assessment of Risk

With the extent of injury and the attendant frequency, we now have the results needed to make statements about the risk. By summarizing these results in diagrams and tables, estimated risks can be given for the study.

4.4 METHODS OF EVENT TREE ANALYSIS

In the event tree analysis, the various potential effects of a defined, initiating event (e.g., rupture of a pipeline) is determined by the success or failure of needed countermeasures (system functions). Depending on the scope of required countermeasures, a different number of potential event sequences result; these are compiled in the so-called event trees.

As explained in previous sections, particular events will be described for the determination of risks from nuclear power plants that can lead to core melting. Systematic investigation permits grouping into types of accidents and thus to the definitions of different classes of initiating events. On this basis, more or less detailed event trees are prepared according to the complexity of events.

On the basis of a simple example, preparation of an event tree will be explained. As an initiating event, we assume a leak in the primary coolant line. This leads to a reactor scram triggered by the reactor protection system. Depending on the success or failure of this safety measure, two different event sequences result. In the further course of the accident, the systems for emergency cooling and residual heat removal come on automatically (b). Finally, the leak tightness of the containment is important. This can be affected not only by leakage from the containment itself, but also by the failure of ventilation valves, drainage lines, etc., to close. The event tree for the particular example is illustrated in Figure 4-2. The initiating event and countermeasures are designated by letters. The success of a countermeasure is denoted by an upward branch;

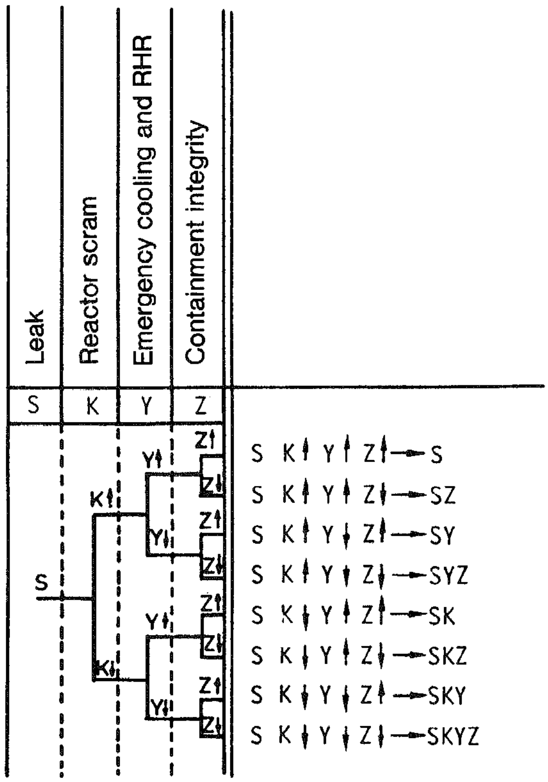


Figure 4-2. Simplified event tree for a LOCA

that of failure is denoted by a downward branch. The corresponding letters are $K\uparrow$ or $K\downarrow$ $Y\uparrow$ or $Y\downarrow$ $Z\uparrow$ or $Z\downarrow$. The event sequences are distinguished by means of attendant letter combinations: $S\ K\uparrow\ Y\downarrow\ Z\uparrow$. Later, a uniform, abbreviated notation is selected in which the initiating event and unsuccessful measures can be described, i.e., $S\ K\uparrow\ Y\downarrow\ Z\downarrow$ is written S Y Z, etc. As the example shows, even with only three different safety measures, a total of eight different event sequences result.

For each of these eight sequences, the involved physical processes must be studied, e.g., cooling of the core and radioactivity release from the core.

In practical implementation of the event tree analyses, it turns out that there are significant reductions in the scope of the particular branchings due to:

- interdependence of the systems,
- system-induced subsequent failures, and
- the suitable organization of the event trees.

On the other hand, the first two points generally lead to interdependencies of the events S, K, Y, Z, which must be carefully considered in the analysis. Therefore, we will discuss below the important viewpoints toward reducing the size of the tree and the required consideration of system or function dependencies.

System Interdependency

The countermeasures implemented upon the occurrence of an initiating event are performed by systems which, as a rule, are not mutually independent. In addition, the requirements on systems depend on the particular event sequence and on the mode and scope of the initiating event (e.g., in a LOCA, it depends on the location and size of the leak). For both reasons, we chose the designation "system functions" for the countermeasures and do not refer directly to the real systems themselves. In this regard, we refer to the coupling of functions Y and Z. If a leak occurs through a connecting pipeline of the reactor coolant system into the annulus of the reactor building, then the leak tightness of the containment has broken down and, at the same time, has influenced emergency cooling and decay heat removal.

The individual system functions are defined so that their physical effects on the event sequences are different. The reasons for this can be, for instance, differences in the system demand time point or in the required quantities of coolant.

System-Induced Subsequent Failures

The structure of the event sequences, i.e., the chain of events, corresponds to the chronology of the accident. Each event in the chain has to take into account the consequences of the previous events. For instance, if a transducer of the reactor protection system is broken by water exiting from a leak, then this would have to be considered accordingly in the system function K. Actually, the transducers of the reactor protection system are designed for conditions prevailing during a LOCA. In addition, in many cases of failure of a system function, the subsequent system is rendered inactive and thus has no further influence on the event sequence. In the selected example, upon failure of the reactor scram, the emergency cooling cannot prevent a core melt. The function Y is therefore not considered, and the branching point at this place in the diagram can be omitted.

Organization of the Event Trees

In preparing event trees there is a useful discrimination at the decision point "core melt," i.e., at the interface where the particular event sequence tells whether core melt occurs or not. The event sequences illustrated in Chapter 5 go as far as this interface. To determine the release of radioactive substances due to a core melt, the failure modes of the containment are examined and cataloged in accordance with a form useful to the calculation of release rates. In the study, six different failure modes of the containment are defined by decreasing leakage area α, β_1, β_2, β_3, η, δ (see Section 6.3.3 and 6.6.2). These failure modes are connected with event sequences that lead to the interface named above. Therefore, the release frequencies from the containment correspond to the particular failure modes. This is illustrated for the example by assuming that only two failure modes of the containment Z_1, Z_2 and only two release categories had to be considered. The system function Z would then be replaced by the secondary functions Z_1 and Z_2, and we would obtain the event tree shown in Figure 4-3. Since the failure modes Z_1, Z_2 represent mutually exclusive events, the corresponding branches can be omitted after the occurrence of Z_1. The event sequence represents the situation for proper operation of all system functions; the event sequences S-Z_1, S-Z_2 show the accidents mastered by the emergency cooling system. The event sequences

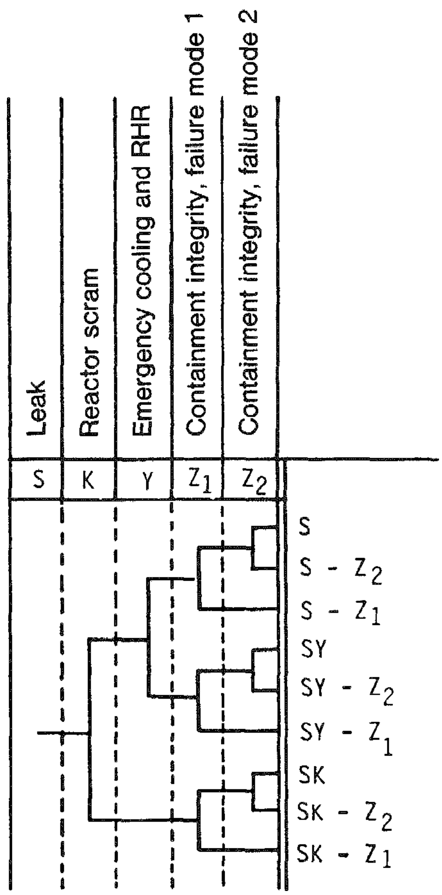


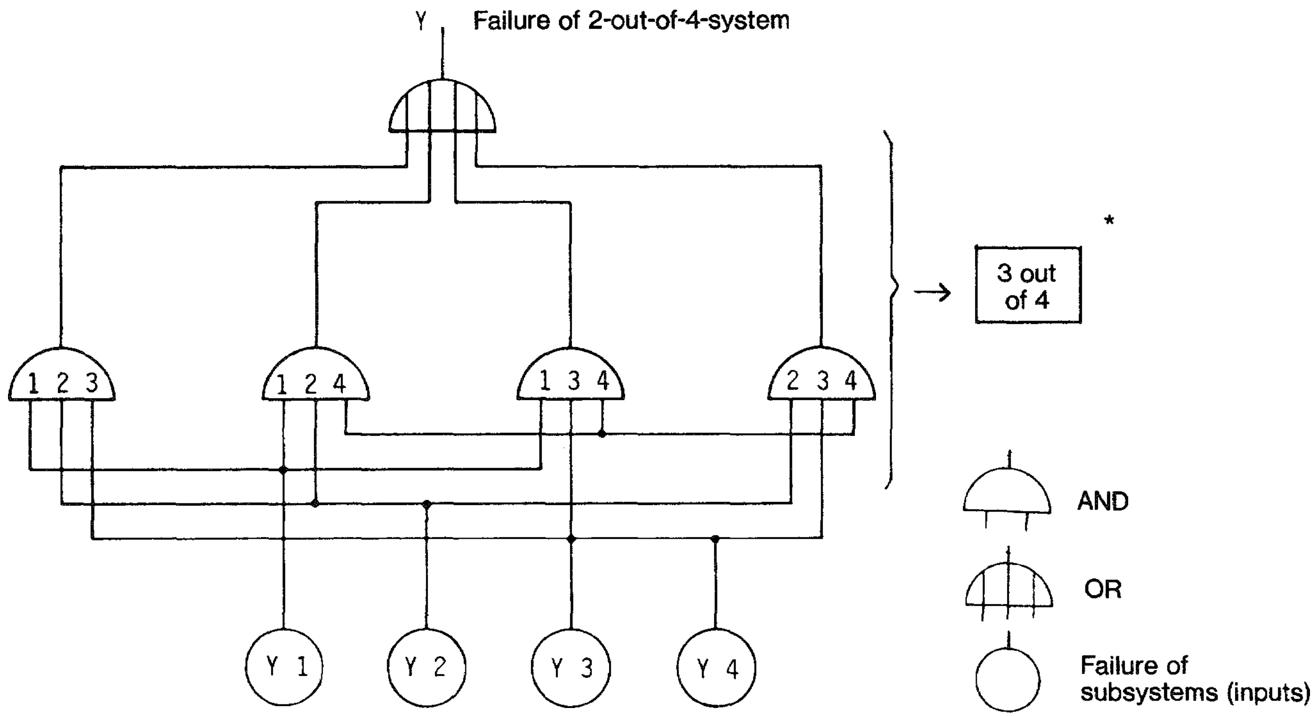
Figure 4-3. Consideration of failure modes of the containment in the event tree

SY and SK lead to core melt without failure of the containment. Actually, core melt always leads to a failure of the containment; this is taken into account in the study by a separate failure mode δ.
After preparation of the event trees, quantitative evaluation takes place by determining frequencies of the initiating events and the probabilities of failure of the needed system functions (unavailability or failure probability). In accordance with the interdependencies of system functions discussed above, we are dealing with conditional probabilities. Multiplication of the event frequency with the conditional probabilities for the corresponding system functions gives the frequency of the particular event sequence. Details on calculation of probability are found in Section 2.5.2.

4.5 METHODS OF FAULT TREE ANALYSIS

As explained in the previous section, for quantitative evaluation of event sequences, it is necessary to determine unavailability or failure probability of needed system functions. By unavailability, we mean the probability that a system function demanded at a given time, e.g., start of a pump, will fail. By a failure probability, we mean the probability of failure of a system function within a time period, e.g., the failure of a running pump needed to maintain the integrity of emergency cooling. To determine these probabilities, we use fault tree analysis. The top event of the fault tree is formed by the failure of the function required by the event tree, (e.g., reactor scram fails upon demand). Proceeding from this "undesirable event," all combinations of component failures are sought that lead to failure of the reactor scram or, expressed in general terms, which lead to the so-called undesired event. Therefore, in contrast to event tree analysis, we are dealing with a deductive method. Its application is important primarily because experience values are usually unavailable for reliability of systems, but are available for the various components. Linkage of the individual component failures in the fault tree takes place primarily by means of the logic operators And, Or, Not.

The system functions needed in the event tree are built up from redundant components or subsystems (loops), i.e., more loops are present than actually needed to fulfill these functions. For instance, the system function "emergency cooling and residual heat removal" is generally performed by a two-out-of-four system. The system consists of four loops, of which two are sufficient to perform the desired function. We also call this a 4 X 50% system since the function is 100% met when two loops are functioning. The fault tree of such a system, resolved into its



*Failure of any three of the four subsystems Y 1, Y 2, Y 3, Y 4 leads to failure of the two-out-of-four system. This simplifies the illustration of the fault tree for a 3-out-of-4 system.

Figure 4-4. Fault tree for a 2-out-of-4-system individual loops, which in turn are composed of a number of components, is shown in Figure 4-4. In the definition of "undesired event," it is of great importance to know how many redundant loops are needed to fulfill the safety obligation. We are speaking here of so-called effectiveness conditions.

The effectiveness conditions depend both on the initiating event and on the further event sequence. For certain leaks, one loop may be enough to pump in sufficient water. For certain types of leaks, one loop can fail from the beginning because it is feeding water directly into the leak. The effectiveness conditions of the study were based on how they were described in the licensing procedure.

The coupling between event tree and fault tree takes place in the manner shown in Figure 4-5. To determine the frequency of each event sequence, an AND coupling of the initiating event with the undesired event (failure of the system functions) is performed, as they occur in this sequence. In the illustrated example, the initiating event "leak" is linked to the failure of emergency cooling and residual heat removal and the failure of the containment (SYZ). The individual functions K, Y, Z have conditional probabilities because of the interdependence. This means that we must always note how the particular state in the event sequence (intact or failed) affects the down-stream functions. This must also be checked in preparing fault trees for the individual system functions, i.e., the fault trees must be rethought for each sequence and if necessary, modified. For instance, fault trees for the function Z are generally different in sequences SZ and SYZ.

In the study, the fault trees include the total interaction of governing systems (e.g., reactor protection system), energy supply (e.g., emergency power system), and processing systems (e.g., emergency cooling and residual heat removal system). So it is possible to identify and appropriately account for failures caused by the interaction of mutually dependent systems.

The important input data for quantitative evaluation of the fault tree analysis are failure rates or failure probabilities per demand and the uncertainties in these data, the time between function testing, and the unavailability due to maintenance (service and repair). Primarily by means of regular function testing of the different subsystems and components, it is possible to easily adapt the calculation of system reliability to the conditions occurring in operation.

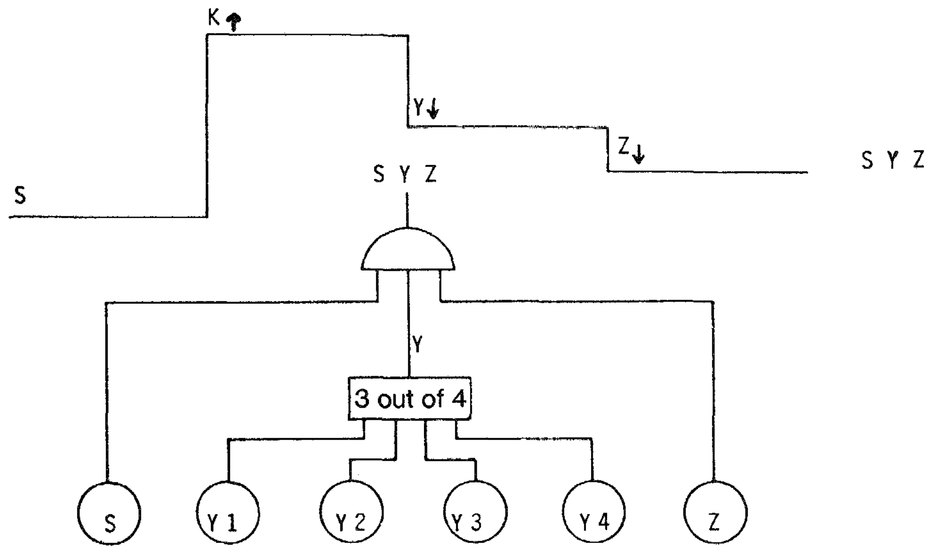


Figure 4-5. Coupling of event tree and fault tree

The numerical evaluation of the fault tree analysis takes place by means of the RALLY program. This consists primarily of a simulation program to determine the expected values of unavailability and of an analytic-simulation program to determine uncertainty in the mean unavailability due to the uncertainty in the failure rates.

4.6 RELIABILITY DATA

For a quantitative evaluation of event trees, the following data must be determined:

- the frequency of the initiating event, and
- the probability for the failure of system functions.

The expected frequency of initiating events is generally derived on the basis of observations: either estimated values of these frequencies are obtained directly from operating experiences (e.g., for the occurrence of a pipeline leak), or the initiating event is broken down into sub-events by a fault tree analysis for which operating experiences are available (e.g., power failure). The number of observed events of one type is related to a period of one year. The attendant frequency represents an average value for the expected occurrence of the event per year. Therefore the frequency can be much greater than one and should not be confused with the probability of an event, which by definition is between 0 and 1 (see Section 2.4.2).

The probability for failure of system functions is determined by means of fault tree analysis in which a probability is derived from the failure of components for the failure of system functions. The decisive statistical quantities are the failure rates λ or the failure probabilities per demand p of the individual components. In addition, information on maintenance (service and repair) of the components as well as on the time intervals of regular function testing are important. However, we will not further discuss this data because it is of moderate importance and is easy to determine.

The failure behavior of a component that has to perform a certain function can be described in one of the following two manners:

- By a failure rate λ. By failure rate we mean the relative decrease in the number of intact (unfailed) components occurring per unit time.

- By a failure probability per demand p. By failure probability per demand we mean the probability that the component will fail upon demand (the component fails during the time before the demand or at the latest, the moment of demand).

Both quantities are values derived from experience. Therefore, they are determined from the statistical evaluation of observations of the operational use of corresponding equipment (or, on a lesser scale, of laboratory tests.)
As a rule, we find a time behavior of the failure rate λ, which we call the "bathtub curve" (Figure 4-6). At the beginning of operational employment the possibility for premature failure exists: For example, faults due to manufacture that are not detected in spite of quality control and commissioning testing can result in increased failure rates. The number of defective components decreases continually with time until only satisfactory components remain. At the end of component service life, the failure rate can increase due to wear and age. During the majority of the use time, however, the failure behavior is not determined by this type of systematic failure cause; therefore a constant failure rate can be expected. We call this random failure. This is an exponential distribution, i.e., the distribution function or failure probability of a component is given by a function of time t as:

$$F(t) = 1 - e^{-\lambda t}$$

Although the occurrence of early and wear failures is counteracted in nuclear power plants by use of operationally proven components, quality control, and recurrent testing, the time dependence of component failure cannot be excluded. On the basis of operating experiences we obtain average values for failure rates or probabilities. These constant values are used in the fault tree analyses.

Of the two types of presentation of component failure behavior, the description generally uses the failure rate. If such a component is tested at regular intervals T, then the failure probability per demand of the component can be described by:

$$p = 1-e^{-\lambda T} \cong T \text{ for } T \ll 1$$

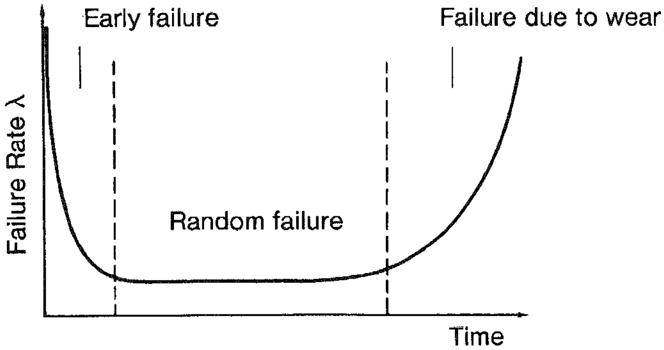


Figure 4-6. Time history of the failure rate

A description using a constant failure probability per demand is selected when the failure is caused as a result of the demand, like for instance, human interference.

From operating experiences, accurate values for the pertinent failure rates or probabilities cannot be obtained. Rather, uncertainties in these quantities exist that are partially attributed to the mentioned time dependencies. The primary cause of data uncertainty is that, as a rule, not enough observations of the particular event are available. So it is not possible, for instance, to give failure rates for each individual type of pump for the particular operating conditions. However, a series of failure rates does exist that were determined for different types of pumps under different use conditions. Naturally, uncertainties in failure rate are expected for an individual type of pump. These uncertainties can result from accidental errors in the manufacturing process, in operation, maintenance, etc. However, these rates are within a considerably narrower range than those resulting from the more general available experiences. Therefore, from experiences we can only give a range within which the failure rates will lie.

Uncertainties in the reliability data are taken into account by using a distribution instead of point estimation for the pertinent failure rates or probabilities. This type of presentation reproduces existing information: it indicates the probability according to available experience that the value of the particular quantity will lie in a certain range. To determine uncertainties in the estimation, the log-normal distribution is used (from WASH-1400, see Section 4.7.2).

These distributions of failure rates or probabilities are used as input data for the fault tree analyses. Accordingly, their results are not point values, but new distributions. Therefore, it is explicitly demonstrated that uncertainty in estimating the results is due to uncertainties in estimating input information.

Since the described methodological procedure corresponds to that of WASH-1400, it was originally intended to use the failure rates or probabilities found there. However, the reliability data in WASH-1400 could not be reproduced. For this reason, our own evaluations were performed under consideration of operating experiences of German power plants. In general, greater expected values (on medians) and greater uncertainty factors of failure rates or probabilities were found than in WASH-1400. Since instrumentation and control in German nuclear power plants differs significantly from that in WASH-1400, our own investigations were needed for the respective components in any case. In order to obtain a better foundation for failure rates and probabilities, operating experiences in nuclear power plants will be more thoroughly evaluated in phase B of the risk study.

In the fault tree analyses, interference by the operating personnel must be taken into account. Cases in which human activities were improperly performed can be identified, but a calculation of probability is very difficult. Since human activities can hardly be compressed into a rigid scheme, we must use estimations here. This type of overall evaluation is sufficient in many cases since we are trying to significantly reduce the influence of human error by means of design and operation.

For instance, in all safety systems the influence of human error is reduced by surveillance testing or other process system responses by means of control commands: if a safety system is triggered by reactor protection signals, then the most important valves equipped with a motor drive are triggered again and driven into the proper position as necessary.

In addition, a principle of German nuclear power plant design is that measures required within 30 minutes of the beginning of an accident will take place automatically without interference by the operating personnel.

In those cases where human error can nevertheless have an effect, at least a rough estimation of corresponding probabilities is needed. As much as possible, we proceeded as in WASH-1400. Thus for actions implemented after the beginning of an initiating event, the probability of human error is set higher the less time has elapsed since the beginning of the initiating event. In practice, only planned interference is considered, as prescribed in the operating manual. Unplanned interference, which can have both a negative as well as positive effect, is not quantified.

Common cause failures of components, subsystems, or systems represent a different problem. By this we mean interdependent failures of several components, subsystems, or systems due to a common cause, so that the failed states exist simultaneously.

Common cause failures are particularly troublesome if they affect redundant components, subsystems, or systems. Basically, we distinguish between:

- failures of two or more redundant components, subsystems, or systems due to a common cause that results in simultaneous failures or, at least, in the simultaneous existence of failure states
- simultaneous failures of two or more redundant components, subsystems, or systems that take place as the result of an individual failure (down-stream failures).

In WASH-1400, such common cause failures were also those where several systems failed simultaneously because they had the same components or auxiliary system or functional interdependencies. These failures are correctly included automatically in the present study by the fault tree analyses, and therefore they are not detailed separately. The contributions of such interdependencies to common mode failures found in WASH-1400 are slight, however, since in this report the system functions are established so that the corresponding systems contain only a few common components. (For instance, the power supply is defined as a separate system function.) Defining system functions by this method was not useful in the present study since the individual loops of the safety systems are generally separated in the reference plant; this would have made determination of probabilities for the failure of different system functions more difficult. The establishment of system functions took place here under different viewpoints.

To protect against a series of potential sources of common cause failures, German nuclear power plants avoid, as much as possible, linkages between redundant subsystems (loops) of safety systems. This means that failure of individual components will not have simultaneous effects on several subsystems. In addition, the loops are spatially separated and--especially where spatial separation is not possible--are protected by means of protective structures. Another measure taken against common cause failures is the principle of diversity, i.e., the use of different functional or construction principles for redundant safety features.

To quantify common cause failures, a subdivision according to types of detection is important. Here we differentiate between common cause failures that:

- occur or are detected only during an accident
- are detected during regular functional demands (in the framework of function testing or other regular system demands)
- are self-reporting.

Operating experiences primarily provide data for common cause failures that are detected during operation and primarily during function testing. Common cause failures occurring or detectable only during an accident can generally be determined only by analytical means. The common cause failures are very difficult to detect if requirements, both in operation as well as function testing, are not representative of requirements from components or systems occurring under accident conditions. In this regard, we mention again that the study assumes that this question is considered within the framework of the licensing procedure and that therefore such types of common cause failures do not play a dominant role. Quantification also proves very difficult for common cause failures detectable during operation and function testing, since observations are only conditionally applicable here. This is due to the following reasons:

- Only a fraction of component failures are common cause failures.

- The sources for occurring failures recognized as common cause failures and which have a great influence on system reliability have been eliminated. Therefore, similar failures are less likely (probable) to recur.

But in order to permit quantification of common cause failures, various methods are given in the literature that permit an estimation. These methods are described in the appendix. In the present study, common cause failures are quantified only if operating experiences are available on this failure or similar failures. A prerequisite for a numerical evaluation is that corresponding failures must have already occurred. Such failures are known for measured value acquisition, termination relay, diesel emergency power system, and pumps in long-term operation.

4.7 UNCERTAINTIES IN THE ANALYSIS

A large number of different accident sequences is simulated for a risk determination. Each accident sequence consists of:

- The plant-internal event tree, which runs from the initiating event to release, and

- The plant-external exposure sequence, which includes the dispersion and deposition of pollutants, local distribution of exposed persons, and harmful effects as well as protective actions and countermeasures (see Figure 2-6).

The results of the event trees are the starting points for the exposure sequences. The accident sequence model simulates the latter. The results of the event trees are compiled into release categories. The categories are characterized by:

- expected annual (c) frequency, and
- release characteristics important to assessment of injury.

The contribution of the simulated accident sequence to the determined annual risk consists of two components, namely:

- the expected annual frequency, and
- the expected consequence for the particular type of accident.

Both components are affected by uncertainties in the estimation. These result from:

- inaccurate knowledge of fixed or temporally fixed quantities like probabilities, expected frequencies, average values in general, etc.
- approximate functional description of regularities in the event and exposure sequences. Among these regularities are the laws of chance expressed by distribution functions or reduced to estimated values.

The expected consequence of a simulated accident sequence is determined through a sequence of functional relations that should describe the route to a core melt, the process of core melt, the radioactive release, dispersion and deposition of pollutants, and the exposure pathways under consideration of protective actions and countermeasures. The functional relationships are, in general, deterministic-mathematic models derived from physical models of the accident sequence. Section 4.7.1 deals with their validity.

The calculated results from deterministic-mathematic models depend on specific conditions of the simulated accident sequence. These specific conditions--like release category, weather situation, population distribution--are so-called random quantities since it cannot be definitely predicted which release, if any, will take place, what the weather will be, and what population distribution will be affected. By using probabilistic-mathematic models, the random behavior of these and other participating random quantities can be modelled. The calculated result is the expected frequency of the simulated accident sequence. Section 4.7.2 deals with the specific uncertainties in the probabilistic mathematic model.

4.7.1 Uncertainties in the Calculated Simulation of Accident Sequences

The simulation of accident sequences is based on physical-mathematical models of complex processes like, for instance, time- and location-dependent power distribution in the reactor core, thermohydraulic processes in the reactor coolant system and in the feedwater-steam loop, the load and failure mechanisms of the containment, or the dispersion of released fission products in the plant environment. As for almost all complicated technical systems, the model provides only approximate descriptions of the actual processes.

As a result of the above discussion, standard engineering practice is to design components of a system to cover uncertainties beyond the necessary "load capacity" with a safety margin ("conservative" design). This results in systems that can withstand not only expected normal stresses, but also more difficult conditions like, for instance, exceptionally high stress or reduction in normal load capacity due to failure of individual system components.

To determine the required load capacity, model calculations or experiments can be used. In nuclear engineering, however, the problem arises that experiments usually cannot be implemented on a full-scale model; only individual phenomena of a complex procedure can be simulated. Emphasis, therefore, is placed on a calculated simulation of the expected loads. Uncertainties in forming the model and in selecting parameters must be covered by assumptions that result in overestimation of loads ("pessimistic" assumptions).

The safety assessment in the licensing procedure is based on the above methods. The ability of systems to cope with designed accidents must be demonstrated by a calculated simulation in the licensing procedure. In appropriate guidelines (e.g., BMI safety criteria [1], reactor safety commission guidelines [2]), basic assumptions of the simulation are established to arrive at pessimistic results.

The risk study adopts information from accident simulations performed within the framework of the licensing procedure for the reference plant or comparable plants, primarily for the following purposes:

1. to determine the safety systems and the number of redundant loops of systems needed to cope with an accident; this provides prerequisites for a reliability analysis.

2. to determine the starting and boundary conditions for the study of phenomena in a core melt.

To study the phenomena during and after a core melt accident and from the core melt process itself down to the resulting health injury, models and parameters were taken primarily from WASH-1400 and partly modified. These models have necessarily been greatly simplified since we are normally dealing with complicated processes, all of whose details may not yet have been studied experimentally. Therefore, in numerous respects, uncertainties must be covered by pessimistic assumptions.

However, the problems are simplified because the plant system analysis of core melt accidents must determine those times when particular effects occur (e.g., beginning of core melt, contact between core melt and sump water, failure of the containment), whereas the detailed accident sequence is not of decisive importance. Effects that can influence the extent of fission product release and thus the radiological consequence can be identified with relàtive ease and determined in a pessimistic manner.

For the deterministic-mathematical model of accident sequence calculation, this is not easily possible. Therefore, an attempt is made here to quantify the influence of important parameters.

4.7.2 Methodological Treatment of Statistical Uncertainties

The sources of statistical uncertainties are naturally associated with the input quantities for the probabilistic-mathematical models. Calculated results of these models are the expected frequencies of the simulated accident sequences. They are determined as a product of:

- the expected frequency $h(k_i)$ of radioactivity release of a particular release category
- the probability $w(M_t)$ with which a weather situation occurs as assumed in the simulated exposure sequence, and
- the probability $w(B_v)$ that a population distribution is encountered as in the simulated exposure sequence.

To estimate the probabilities $w(B_v)$, $v = 1, 2, \ldots, n$, in the study, we divided the area around 19 nuclear sites into 36 main wind directions (see Chapter 7). Each main wind direction is thus assigned to a particular population distribution at each site.

If several units are at one site, then the attendant population distribution is counted an appropriate number of times. In this manner, we obtain a random sample of n = 25 x 36.

Basically, any number of population distributions is possible because the 19 sites can be interpreted as a random sample of the total potential sites. In addition, the wind direction at the actual site is variable, and the population distribution is not constant (daily cycle, long-term [yearly] trends). The number of possible population distributions can be imagined to be divided into classes (d). If n_j population distributions of the random sample fall within class j, then the probability that this class of population distribution will be affected in a real accident is estimated as n_j/n. In the simulation we are not using average population distributions of the individual classes, but real population distributions B_v, v = 1, 2, . . . n of the random sample, which are assigned the probability $w(B_v)$ = 1/n. In order to be accurate, each probability $w(B_v)$ would have to contain the expected relative frequency that the main wind direction will blow within the affected 10-degree sector. However, it is assumed that the average difference in frequency over all population distributions n_j of a class are compensating so that we can proceed in our calculation from a uniform distribution of wind direction. A methodical investigation of the uncertainties pointed up here was not performed because their influence was judged to be relatively small.

The probabilities $w(M_t)$, t = 1, 2 . . ., m are also estimated on the basis of random samples. The study divides the territory of the FRG into four siting regions of clearly differing meteorology and assigns the population distributions of the above random samples, in accordance with the sites, to these regions. From each region, we used a random sample of 115 weather profiles observed over several hours; the starting points of these observations were uniformly distributed over an entire year. The weather patterns come from records of a site considered typical for the particular meteorological region. The random weather sample thus includes a total of 4 X 115 real weather sequences.

Basically, in each of the four siting regions, any number of different weather sequences are possible. We can imagine the weather sequences to be divided into classes. If m_r of the 115 weather sequences lie in class r, then we estimate that in a real case the probability of a weather profile for this class is $m_r/115$. Here, also, we are not using average weather patterns of the individual classes for the simulation, but real weather profiles M_t, t = 1, 2, . . ., 115 of the random sample. Each has an assigned probability w (M_t) = 1/115.

If we had taken the weather profiles at a different site from a different observation year, then the probability estimations for the individual classes may have been different. Uncertainties in estimations of this type were considered to be small and were therefore not treated methodically. The question of whether 115 random sample elements is sufficient to satisfactorily characterize the quantity of weather profiles in a year for purposes of the study was discussed in (3).

The release category frequencies $h(K_i)$, $i = 1, 2, \ldots, k$ are the sums of expected frequencies of the assigned accident sequences. These, in turn, were determined through fault tree analyses and event trees from numerous frequencies and probabilities. All of these probabilities and frequencies are interpreted as estimations of average values, i.e., averaged over several nuclear power plants of the referenced type.

We estimated:

- expected frequencies of initiating events
- failure rates (or failure probabilities per demand) of components
- probabilities of human error
- probabilities needed in modelling so-called common cause failures and failure modes of the containment.

As a probability rule for the time-dependent failure behavior of components, we use a service life distribution without explicit consideration of so-called early failures and wear failures (exponential distribution). The uncertainty in the selection of the distribution type is not quantified; however, the uncertainties in estimating the distribution parameter, namely the failure rate, are treated methodically like the uncertainties in estimating the other frequencies and probabilities. If estimated values for one and the same failure rate, for probability (or due to insufficient detailed estimations on groups of failure rates or probabilities), or for expected frequency are different, then an empirical distribution is interpreted as an expression of the estimated uncertainty and is approximated by a log-normal distribution. The following priority order is distinguished for estimated values:

- estimated values from observations in nuclear power plants of the referenced type under accident conditions

- estimated values from general operating experiences in nuclear power plants

- estimated values from observations in comparable areas (coal power plants, etc.)

- estimated values from observations in other spheres (chemistry, laboratory, etc.)

- estimations by experts.

For instance, if only one estimated value is derived from observations, then in general the observation uncertainty is quantified by a method whose effect corresponds to application of the law of Bayes for noninformative, a priori distribution (uniform distribution over the potential value range). With simple expert assessments, citation of estimated uncertainty is naturally based on the judgement of the experts.

By using distributions to quantify estimation uncertainty, the presumed constant but inaccurately known quantities are assigned value ranges, together with subjective probabilities for the position of the particular value within a certain subrange. The general distribution type used is the log-normal distribution. The reasons for this selection are primarily:

- The log-normal distribution assigns zero probability to values of less than or equal to zero. Thus it considers the fact that the values of all pertinent quantities are positive.

- It is easy to adapt to many empirical distributions of existing estimated values.

- It is a suitable probability law for quantities which themselves are the product of many, possibly normally distributed quantities.

- It can be characterized by two parameters. Normally these are:

  --the median (X_{50}) whose name implies that the probability of values less than or equal to X_{50} is equal to the probability for values greater than X_{50}; that is, the probability is 50%, and

  --the uncertainty (or also called uncertainty factor) K_p, which has the property that the probability of values less than or equal to $X_{50} \times K_p$ is equal to the probability of values greater than $X_{50} \times K_p$; that is, it is (100-P)%.

  Because of these parameters and due to its relationship to the normal distribution, a log-normal distribution is quite easy to use in calculation.

- Because of its slope, it offers the possibility for excellent representation of quantities distributed primarily over low value ranges. Simultaneously, higher values of the distribution are better represented than would be the case, for instance, for a normal distribution with the same 5% and 50% fractile (median). However, a normal distribution with the same 5% to 95% confidence interval would on the average result in higher estimated values of the risk since its expected value lies above that of the log-normal distribution for the particular uncertainty range of interest here.

The expected (mean) value of a log-normal distribution is greater than its median. This expresses the property of more accurately representing the higher range of values than does a normal distribution with equal 5% and 50% fractiles (if the probability for values less than or equal to X_p is equal to P%, then X_p is called "P% fractile"). Between the expected value (X), median (X_{50}), and uncertainty factor (K_p) of a log-normal distribution, there is the following relationship:

$$X = X_{50} \times e^{s^2/2},$$

where $s = \ln K_p/u_p$ and "u_p" is the P% fractile of the standard normal distribution. Fractiles other than u_{95} about equal to 1.645 can also be found in standard tables (see, for instance, [4]) of statistical distributions.

The quantified uncertainties in estimating failure rates, probabilities and expected frequencies of initiating events are propagated according to the rules of probability calculation through the fault tree analyses and event trees down to the frequency of the particular release category. Thus, for these expected frequencies--which are to be considered as estimations of average values for several nuclear power plants of the referenced type--we have not one single value, but whole distributions. These distributions express the quantified estimation uncertainties of the expected release frequencies and are included in the subjective confidence intervals of the study results, together with other estimation uncertainties (see Section 8.2).

In order to be able to give a complementary cumulative distribution frequency of injury (representation form of risk in Section 8.1), a value must be selected from the subjective probability distributions of the expected release frequency, which can be called the "best" estimate. The question of which value is to be considered the "best" arises because, in principle, each value for which the subjective probability density function differs from 0 is a possibly correct choice. Depending on the particular problem, we make a selection between modal value (the probability distribution function assumes its maximum in this case), median, and expected value.

In this study the expected value is used. Only it represents the quantity, affected by uncertainties in estimation, in a manner such that the amounts of potential over- and under-estimations, weighted with the subjective probability of its accuracy, are kept in balance. In the case of the median, the potential over- or under-estimation only keeps the subjective probability in balance and therefore ignores any amounts due to incorrect estimation. Use of the median would result in under-representation of the risk, given the slope of distributions of the expected frequencies in the light of the quantified estimation uncertainties.

The expected value also has calculation advantages. For instance, the sum of expected value of different quantities is equal to the expected value of their sums, regardless of their distribution. The product of independent quantities behaves analogously. These properties do not apply in general for the modal value and median.

With normal distributions, the modal value, median, and expected value are the same. For log-normal distributions, the modal value is smaller than the median, and this in turn is smaller than the expected value.

The confidence interval in Section 8.2 is completely independent of the selection of the "best" estimated value of fixed, but inaccurately known quantities for which estimation uncertainties were quantified. Their determination is based on a complete, subjective probability distribution of these quantities.

Actually, the frequency of a simulated accident sequence would have to be expressed by the following product:

$$h(K_i, M_t, B_v) = h(K_i) \times w(M_t/K_t/K_i) \times w(B_v/K_i M_t),$$

i.e., instead of the above probability $w(M_t)$, the conditional probability would have to be used. This includes any dependencies between weather and radioactivity release. This condition will have an effect, for instance, when the frequency of certain initiating events or the probability of human error, etc., clearly depends on the weather. Uncertainties resulting from the use of $w(M_t)$ instead of $w(M_t/K_i)$

are judged negligible. The condition in $w(B_v/K_iM_t)$ is partially considered by the fact that only population distributions and weather patterns for the same meteorological site are included in the simulation.

The expected extent of injury and the expected frequency of the simulated accident sequences come from models of the accident event whose estimation uncertainties have been partially quantified or judged negligible by experts. In addition, estimated values of different quality stages are used in these models. Consequently, the confidence interval of the results given in Section 8.2 are subjective in nature.

FOOTNOTES

(a) We are dealing primarily with fission products generated in the fission of fuel in the reactor core. The fraction of activation products due to irradiation of initially inactive system components is comparatively small. Therefore, below we will discuss fission products explicitly.

(b) By residual heat we mean the total heat to be removed after shutdown.

(c) The "expected annual frequency" (see 2.4) should not be confused with the "frequency," which is a whole number (e.g., "frequency in year Y"). Below, for the sake of brevity, we often use the designation "frequency." But by this we always mean the "expected annual frequency."

(d) The class distribution was performed with reference to WASH-1400 on the basis of cumulative population in a 10-degree sector of the particular primary wind direction and the two neighboring sectors (up to Y km distance from the plant).

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Section 5

RESULTS OF THE EVENT TREE ANALYSIS

5.1 OVERVIEW

Event sequences that can develop from operational malfunctions in nuclear power plants are tracked down in risk analyses to extreme situations. For the determination of risk, event sequences of primary importance are those that lead to a melt of the reactor core. Other event sequences that affect the core or other radioactive inventory in the plant will not significantly affect the risk. They are therefore not discussed in the same detail as is devoted to core melt accidents.

As in WASH-1400, core melts are assumed for all cases where the reactor core is insufficiently cooled. This can only occur when safety features fail to the extent that they cannot perform their tasks.

In this chapter, we will study those processes and their frequency that lead to a reactor core melt. We will consider both internal and external initiating events. Event trees are prepared to quantitatively evaluate a large number of possible event sequences. The methods used are described in Chapter 4. Wherever simple plant system features are present, or where notable contributions to risk are not expected on the basis of estimations, preparation of event trees is unnecessary.

Section 5.2 is concerned with accidents initiated by internal system causes (internal accidents). External events that can cause accidents within the plant (external accidents) are discussed in Section 5.3.

5.2 INTERNAL ACCIDENTS

All internal accidents that can lead to overheating of the reactor core can be divided into two groups (Figure 5-1):

- accidents initiated by a loss of primary coolant

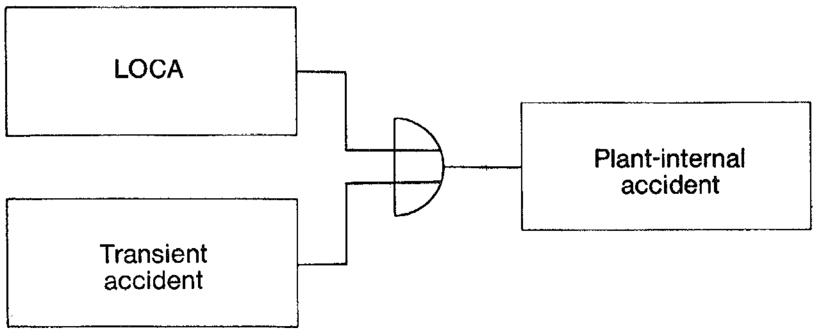


Figure 5-1. Plant internal accidents with consequences to the reactor core

- accidents that increase the power output in the core or that restrict heat removal from the core without any loss in primary coolant.

The first group is called loss-of-coolant accidents (LOCA) and the second group is called transient accidents.

Extensive safety systems are provided to cope with these accidents (see Chapter 3). Even though extensive provisions have been taken to assure reliable operation, a failure of these systems is assumed for the risk determination. Depending on whether safety systems function or fail, we obtain different event sequences. For a systematic determination of these event sequences, we use event trees; these are discussed in Section 5.2.1 for the LOCA, and in Section 5.2.2 for transient accidents. A postulated failure of the reactor pressure vessel (RPV) is discussed in Section 5.2.3. Section 5.2.4 deals with the effects of fire.

List of abbreviations used in Chapter 5:

ATWS	=	Anticipated Transients Without Scram
MS	=	Main Steam
HP	=	High Pressure
LP	=	Low Pressure
RPV	=	Reactor Pressure Vessel

In addition to accidents that can affect reactor core cooling, we must study those circumstances under which radioactive substances can be released from other parts of the system and whether a contribution to risk should be thereby expected. These will be discussed in Section 5.2.5.

The results of the completed studies for plant-internal accidents are summarized in Section 5.2.6.

5.2.1 Loss-of-Coolant Accidents

5.2.1.1 Initiating Events and Measures to Cope With Them. A LOCA occurs when the reactor coolant system leaks due to cracks or ruptures on any part of the reactor coolant system. These leaks are treated in the event tree analyses as initiating events. As in WASH-1400, the following leaks are studied in detail:

- leak in a primary coolant line

- leak in the pressurizer system
- leak through a connecting line to the reactor coolant system.

Event sequences caused by a leak in a primary coolant line are discussed in Sections 5.2.1.2 and 5.2.1.3.

A leak in the pressurizer system can occur either in the compensation pipeline that connects the pressurizer with the primary coolant line or at the pressurizer itself. A leak in the compensation pipeline should be treated like a leak in the primary coolant line and is therefore not discussed separately. A leak from the pressurizer can be caused by faulty opening or closing of pressurizer relief valves or safety valves (see Section 5.2.1.4).

In the case of a leak from a connecting pipeline to the reactor coolant system penetrating the containment, the exiting water does not collect in the sump. This water is thus no longer available for core cooling. If the connecting pipeline leads to the annulus, other failures in the annulus are possible. A LOCA through such connecting pipelines must therefore be treated separately (see Section 5.2.1.5).

A leak from the steam generator is mastered by the safety systems in the same manner as a leak in a primary coolant pipeline. The same is true for a leak from the RPV up to a certain fracture cross-section. Because of the extensive quality control measures undertaken during planning, manufacture, and operation of the pressure vessel, it is assumed as in WASH-1400, that leaks in the pressure vessels are far less likely than leaks in the pipelines and that no notable contribution to risk is to be expected from them. More extensive analyses of leaks not studied in detail must await phase B of the risk study.

In order to prevent overheating of the reactor core during a LOCA, the following measures are needed:

- accomplishment and long-term assurance of subcriticality of the reactor core
- assurance of a sufficient coolant inventory in the reactor coolant system
- heat removal from the reactor coolant system.

These measures are implemented by means of the reactor scram system, the emergency and RHR system, and the feedwater-steam system. For very small leaks, the volume control and chemical injection system may come into use. The systems presented above have to perform different tasks. Called system functions below, they are:

- reactor scram
- HP injection
- accumulator injections
- LP injection for reflooding
- LP injection with sump recirculation
- main feedwater supply and main steam relief
- emergency feedwater supply and main steam relief.

The HP, accumulator, and LP injections are implemented by means of the emergency cooling and RHR system. The main feedwater supply is needed for full power operation and is assured by the main feedwater pumps. In order to remove residual heat and to shut down the plant (i.e., to reduce the coolant temperature), the emergency feedwater supply is sufficient. The emergency feedwater supply is basically assured by the emergency feedwater system and the emergency system; the emergency system may be used to shut down block B only after block A has already been shut down. The main steam relief can take place through the main steam bypass mechanism and the condenser, as well as by direct venting to the outside. The above system functions have been described in Section 3.4.

To cope with a LOCA, it is important to know which functions are required from the individual systems. In general, it is assumed here that the system is at full load operation at the beginning of the accident, since this places the greatest demands on the system functions. In addition, location and size of the leak have a distinct influence on the requirements of system functions.

Leaks in a primary coolant pipeline are therefore studied separately according to several sizes of rupture cross-section (see Table 5-1, column 2). The subdivision of the regions considers the fact that for different break (leak) sizes, different system functions are needed to assure sufficient core cooling (refer to columns 3-7). For example, for a large leak, the pressure in the reactor coolant system drops off so quickly that the HP injection cannot be used. For large- and medium-sized leaks, the residual heat is carried away by the emergency cooling and RHR system. By residual heat, we mean all heat to be carried away after shut-down. For residual heat removal in the case of small leaks, a feedwater supply is

Table 5-1. Minimum requirements for system functions to remove residual heat during leaks in a cold primary coolant pipeline (2 out of 4 means that of four existing redundant systems, only 2 are needed.)

LOCA	Rupture cross sectional area (cm^2)	System functions				
		High pressure injection	Accumulator injection	Low-pressure reinjection for flooding	Low-pressure injection with sump recirculation	Feedwater supply: a) Main feedwater b) Emergency feedwater
Large leak	> 400	--	hot 3 out of 4 cold 2 out of 4	hot 2 out of 4 cold 1 out of 4	hot 2 out of 4	--
Medium leak	80 - 400	2 out of 4	hot 2 out of 4 cold 2 out of 4	hot 2 out of 4 cold 1 out of 4	hot 2 out of 4	--
Small leak	2 - 80	2 out of 4	--	hot 2 out of 4 cold 1 out of 4	hot 2 out of 4	a) 1 out of 4[1] or b) 2 out of 4
Very small leak	< 2	--	--	--	--	a) 1 out of 4[1] or b) 1 out of 4

[1]One out of 4 main feedwater lines are needed.

first needed (main feedwater supply or emergency feedwater supply); a heat sink is a further requirement. For leaks under 2 cm^2 cross-section, the coolant losses can be made up by the volume control system. In addition, the same systems are available as for coping with small leaks. Leaks of less than 2 cm^2 cross-section, therefore, do not contribute any significant quantity to the risk, in spite of their greater frequency, and are left out of the discussions below. For leaks below 1000 cm^2, however, it is pessimistically assumed that the system function of reactor scram is needed. For leaks greater than 1000 cm^2, a reactor scram must not occur. Upon failure of the reactor scram, the loss of coolant causes a rapid shutdown of the reactor through physical effects.

In general, several equivalent subsystems are available (redundancy) to perform the individual system functions. In Table 5-1, columns 3-7, we see how many of the subsystems are used for injection to assure satisfactory system function. Accordingly, for a large leak, three of the four existing accumulators will feed into the hot, and two of four into the cold legs of the primary coolant lines. In addition, during this accident low-pressure injections for flooding and for sump recirculation are needed. For reflooding, of the four existing LP injection systems, two will feed into the hot and one into the cold primary coolant lines. For sump recirculation, LP injections into two hot primary coolant lines are needed. The table pertains to leaks in a cold primary coolant line; for a leak in a hot primary coolant line, merely reverse the information above for "hot" and "cold."

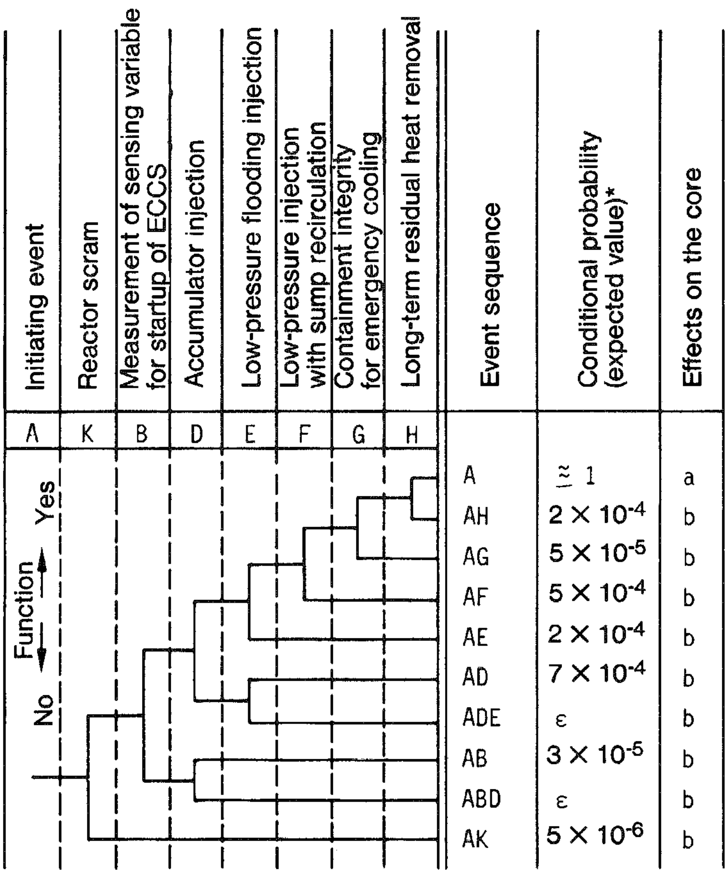
The minimum requirements taken as a basis here have been derived largely from the licensing procedures for the reference plant or comparable plants. This corresponds to the practice used in WASH-1400. If fewer subsystems are available, then the system is considered to be completely failed. This means that a partial operation of the system which might be sufficient to prevent core melt is not considered. The same is true for the different required system functions: upon failure of one system function it is normally assumed that additional system functions cannot prevent core melt. In addition, it is assumed that the named system functions must be available on a constant basis from the moment of demand. This means for a delayed use or temporary failure of these functions that core melt is assumed.

For leaks at the pressurizer, the same minimum requirements are taken as a basis as for leaks in the primary coolant pipeline. Thus, the number of subsystems required for the individual system functions may be over-estimated in comparison to leaks of the same size in the primary coolant pipeline. Removal of residual heat is simplified here by more favorable thermodynamic conditions. It is believed that decay heat removal by the vaporization of water in the steam generator and of the reactor coolant system is tolerable for a limited time, i.e., that a delayed operation of the feedwater supply is sufficient.

5.2.1.2 Large and Medium Leak. In Figures 5-2 and 5-3, event sequences are illustrated for large and medium leaks in a primary coolant pipeline. Merely for the sake of simplicity, in large leaks we do not distinguish between fracture cross-sections greater or less than 1000 cm^2. For a better understanding of the occurring sequences, we first explain the event sequence for proper functioning of systems. As an example, let us consider the accident "medium leak," because in this case more system functions are demanded (event sequence S1 in Figure 5-3).

After the occurrence of a leak, the pressure in the reactor coolant system and the pressurizer water level both decrease, whereas the pressure in the containment increases. This automatically triggers the reactor protection system for a reactor scram; additional energy generation due to fission processes is terminated except for residual heat. Once the limit values from the instrumentation for emergency cooling start-up signals are reached, emergency cooling and closure of penetrations through the containment (containment isolation) are automatically initiated. The high-pressure injection starts up after pressure in the reactor coolant system drops to 110 bar. The accumulator injections begin automatically when the pressure in the reactor coolant system drops below 25 bar. At a pressure of 10 bar, switching takes place automatically from the high-pressure to the low-pressure injections for reflooding. Here, the RHR pumps move water from the storage tanks into the reactor coolant system. The water exiting from the leak collects on the bottom of the containment, in the building sump. If the storage tanks have been emptied to a minimum level, then automatic switching to low-pressure injection for sump recirculation takes place. The RHR pumps draw water from the sump and pump it through the residual heat exchanger back into the reactor coolant system. Thus, water exiting from the leak is used for further cooling of the reactor.

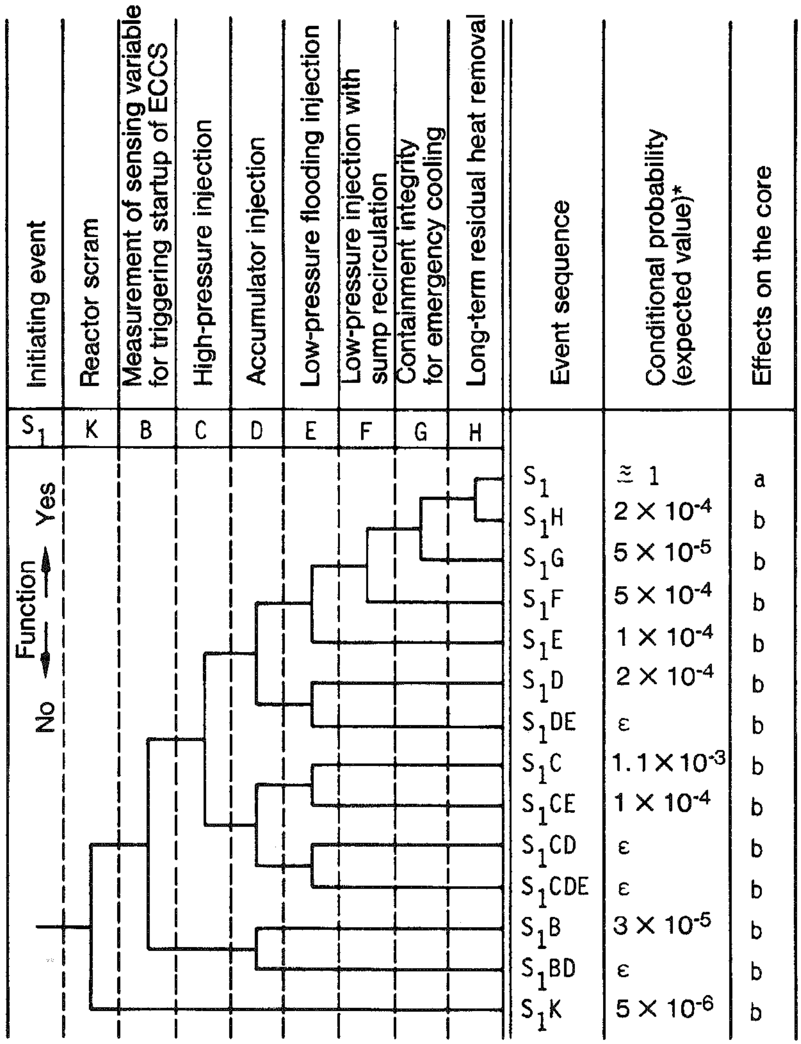
It is important for the assurance of emergency cooling that there be no large losses of water or steam from the containment. This means that containment integrity must be assured for emergency cooling. If there is outlet of water or steam from the containment, there could be:



a No core melt
b Core melt

*Probability of the individual event sequence assuming that the initiating event occurs. The frequency of the individual event sequence is obtained by multiplication with the frequency h of the initiating event. h(A) = 2.7 × 10^{-4}/year (expected value)

Figure 5-2. Event tree for a "large leak"



a No core melt
b Core melt

*Probability of the individual sequences assuming the initiating event has occurred. The frequency of the individual event sequences is obtained by multiplication with the frequency h of the initiating event. $h(S_1) = 10^{-4}$/year (expected value)

Figure 5-3. Event sequence for a "medium leak"

- a failure of components in the annulus needed for emergency cooling, due to temperature, humidity, or pressure
- so much water lost from the sump that sufficient residual heat removal would no longer be assured
- such a drop in the containment over-pressure that the RHR pumps pulling water from the sump may fail due to cavitation.

Because of contamination of the containment after a large or medium leak in the reactor coolant system it is assumed that under some circumstances no work can be performed in the containment for several months. During this time, a long-term RHR cooling must be maintained.

In contrast to a medium leak, to cope with a "large leak," high-pressure injections are not necessary (Figure 5-2). Here, the pressure in the reactor coolant system drops off so quickly that the accumulator injections and low-pressure injections spring into action within a very short time. For leak sizes greater than 1000 cm^2, functioning of the reactor scram system is unnecessary since the core becomes subcritical through boiling of the coolant, rapid drop in water level, and the injection of borated water without insertion of the control rods.

The expected values of the occurrence frequencies of large and median leaks were estimated according to WASH-1400 as 2.7×10^{-4}/year or 8×10^{-4}/year. In Figures 5-2 and 5-3 we find the calculated probability for the various event sequences under the assumption that a large or medium leak has occurred. There, ε generally stands for probabilities less than 10^{-5}, provided they contribute less than 1% to a core melt accident for the particular initiating event under discussion. Lower probabilities are given only if they are of particular interest.

Event sequences with a failure of system functions needed to cope with a large or medium leak lead to core melt. As shown above, the summed probability is 2×10^{-3}, approximately equal for both accidents. The frequencies of a core melt result by multiplying this value times the frequency of the initiating event (5×10^{-7}/year for the accident "large leak" and of 2×10^{-6}/year for the accident "medium leak").

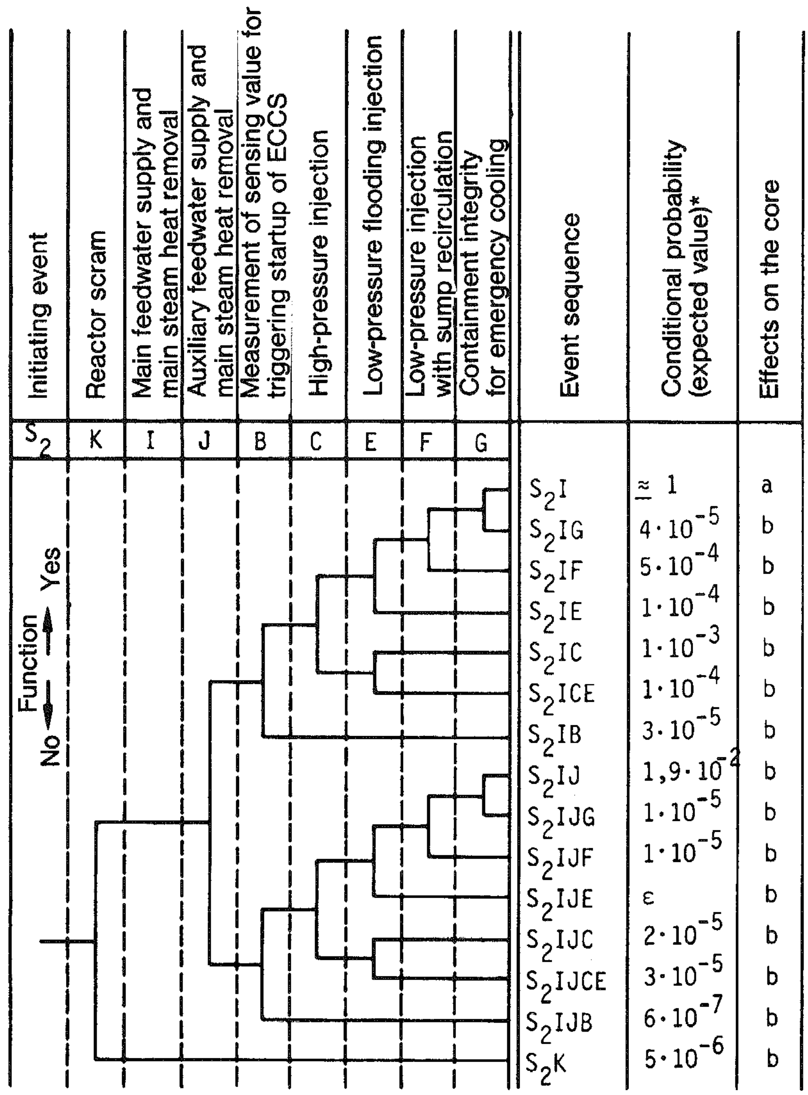
In the comparative discussion of results given below, it must be remembered that different system functions are needed to cope with a large and medium leak, and the minimum requirements of the individual system functions are sometimes not the same.

For a large leak, the event sequence AD gives the greatest contribution: 42%. Each of the four accumulators feeds through one pipeline equipped with check valves into the hot and cold leg of one of the four primary coolant pipelines. If a large leak exists in a cold primary coolant pipeline, then at least three out of four hot, and two out of four cold accumulator injections must take place into the intact legs (see Table 5-1). If the check valves do not open in the two loops through which injection into hot primary coolant pipelines is required, or if the check valves fail in two loops that are pumping into intact cold primary coolant pipelines, then this means that the minimum requirements have not been met. Because of the other minimum requirements, the influence of failure of the accumulator injections for a medium leak (9%) is much smaller. Here, failure of the HP injections, which are not needed for a large leak, contributes 52%. Failure of the 3-way valve and human error, namely, common cause failures due to incorrect calibration of measurement channels that automatically trigger the HP injections play a decisive role. Failure of the switching of the 3-way valve would mean that coolant is pumped into the broken primary coolant pipeline. The appropriate HP injection would in this case no longer be available for emergency cooling.

A contribution of 30% for a large leak and 21% for a medium leak means that failure of the LP injections makes a considerable contribution to the total result for sump recirculation. Here also, common cause failures play an important role through incorrect calibration of measurement channels.

The event sequences with failure of containment integrity for emergency cooling (AG, S_1G) are of special importance. In this case, core melt is assumed for a loss of containment integrity (see Chapter 6).

5.2.1.3 Small Leak. In Figure 5-4 the event sequences are illustrated for a small leak in a primary coolant pipeline. A "small leak" is present when the residual heat cannot be removed through the leak itself. Therefore, an additional heat removal via the feedwater-steam system is needed. That is, after the reactor scram (and successful turbine trip), feedwater must be made available for the steam-generator from the main feedwater or emergency feedwater supplies. In addition, the generated steam must be released into the turbine condenser or through the relief valves (main feedwater supply and main steam relief or emergency feedwater supply and main steam relief). After sufficient pressure reduction, the loss of coolant from the reactor coolant system is compensated by



a No core melt
b Core melt

*Probability of the individual event sequences assuming the initiating event has occurred. The frequency of the individual event sequences is obtained by multiplication with the frequency of h of the initiating event $h(S_2) = 2.7 \times 10^{-3}$/year (expected value)

Figure 5-4. Event tree for a "small leak" high-pressure injections. The HP safety injection pumps draw water from the borated-water storage tanks. Because of the limited supply of the storage tanks, low-pressure injections for sump recirculation must be switched on at the proper time. For the above reasons, a shutdown of the plant is needed. The remainder of the event sequence generally corresponds to that of a medium leak. The accumulator injections are not needed, however.

The expected value of the frequency of a small leak in a primary coolant pipeline was estimated in WASH-1400 as 2.7×10^{-3}/year. In Figure 5-4, we find the calculated probabilities for the various event sequences under the assumption that a small leak has occurred. The sum of probabilities for failure of system functions needed to cope with a small leak in a primary coolant pipeline thus becomes about 2.1×10^{-2}. By multiplication with the frequency of occurrence, we thus obtain a frequency of 5.7×10^{-5}/year for a core melt accident due to a small leak in a primary coolant pipeline. This value is more than one order of magnitude greater than the corresponding values for the large and medium leak. This is attributable to the high initiating event frequency and to the greater unavailability of system functions needed to cope with the accident.

Since, in the reference plant, manual interference in connection with shutdown yields a considerable contribution to the frequency of core melt accidents, this will be discussed in detail below.

After the reactor scram, turbine trip and automatic opening of the bypass system through which the main steam is fed directly to the condenser, shutdown of the system must take place with the aid of the MS relief. Shutdown is initiated by hand. The measures to be performed are found by the control room personnel in the operating handbook. By manual control of the MS bypass valves or--if the condenser is not available--by manual control of the relief valves, the main steam pressure and temperature are reduced accordingly. In order to achieve sufficient delivery from the HP safety injection pumps and the timely switching of low-pressure injection for reflooding, before emptying the borated water storage tank to low-pressure injection for sump recirculation, a downward gradient of MS temperature of 100°C/h is needed on the basis of our pessimistic estimations. If we proceed from simple personnel redundancy, i.e., performance of the measures by a reactor operator and monitoring by a shift supervisor, then for the failure of preparations for shutdown we obtain an expected value of 1.6×10^{-2}. This is about 75% of the total result for a small leak.

If preparations for shutdown occur according to instruction, then an incorrect, excessively fast shutdown of the plant is unlikely. If the shutdown were to occur for a short time period at a gradient of 200°C/h, then the reactor protection signal for leak recognition in the main steam system ($\Delta p/\Delta t$-Signal) would be triggered and the MS quick-closing valve and the MS relief mechanism would close. However, the relief mechanism can be reopened after 15 minutes and the shutdown continued. Even a second triggering of the $\Delta p/\Delta t$ signal does not result in an uncontrolled accident.

When determining probabilities, it must be remembered that the main feedwater control is not designed for the conditions prevailing during a LOCA. For this reason, the main feedwater supply is assumed to be not available, and only the emergency feedwater supply is considered in the analyses. The unavailability of hardware for the emergency feedwater supply and MS relief amounts to 3×10^{-3} and thus contributes about 15% to the total result. Conversely, the unavailability of HP and LP injections is 8% of the total result.

With regard to the event sequence upon failure of containment integrity for emergency cooling (event sequence S_2G), the same is true as presented in the preceding paragraph.

As these results show, the frequency of a core melt accident due to a "small leak" could be significantly reduced, especially by simplification or elimination of preparatory manual measures for shutdown. This could be achieved by improving instrumentation or by automation of the shutdown process. The latter has been achieved in new plants.

By modification of feedwater control by a differential pressure transducer designed for operating conditions in a LOCA, we would also have the main feedwater supply available to cope with the accident. Together with the emergency feedwater supply, two feedwater supplies would thus be available. Failure of both feedwater supplies would thus be negligible with respect to their contribution to the total result.

Overall, the frequency of a core melt accident due to a "small leak" could be reduced by almost one order of magnitude.

5.2.1.4 Small Leak in the Pressurizer. For several of the transients discussed in Section 5.2.2, the pressure in the reactor coolant system increases so much that the pressurizer valves open. The most important of these transients are:

- power failure
- turbine trip without opening of MS bypass mechanism
- turbine trip without rod insertion
- failure of all main coolant pumps
- failure of the coolant pressure control
- ATWS accidents.

In addition, other transients can cause a response from the pressurizer valves if they occur immediately after a rapid change in power output. Opening of pressurizer valves is also possible if the control units triggered during transients are not intact, or if the first triggering of a needed reactor scram fails.

Each of the pressurizer valves is adjusted to a different response pressure; the pressurizer relief valves are adjusted to lower pressure values than the pressurizer safety valves. The increase in coolant pressure is generally limited by opening one or two pressurizer valves; for a majority of transients only one pressurizer relief valve opens. The ATWS accidents (anticipated transients without scram) represent an exception. All pressurizer valves respond to these highly unlikely accidents.

If the pressure in the reactor coolant system continues to drop after the pressurizer valves are open, then after the particular response pressures are met, the pressurizer valves close again. If a pressurizer relief valve does not close, redundant blocking measures are provided. If these also fail, then the result is a "small leak at the pressurizer" in accordance with the valve cross-section. This type of LOCA also occurs when a pressurizer safety valve does not close after the minimum pressure for its response is reached.

In Section 5.2.1.1 we pointed out that the same minimum requirements are assumed for a leak from the pressurizer as for a corresponding leak in the primary coolant pipeline. Because of these minimum requirements, for small leaks at the pressurizer the pessimistic assumption is that a failure of the reactor scram leads to core melt. ATWS accidents, where one of the pressurizer valves no longer closes, are accordingly treated as core melt accidents. Evaluation of small leaks at the pressurizer under anticipated transients is presented in Section 5.2.2.4; ATWS accidents are discussed in Section 5.2.2.5.

5.2.1.5 Leak From a Connecting Line to the Reactor Coolant System. The reactor coolant system is connected to various systems (e.g., emergency and RHR system) via connecting lines situated outside the containment. Even though the connecting lines are provided with sequential (series of) blocking valves, it must be checked whether a loss of coolant can occur through these connecting lines and result in loss of primary coolant from the containment. In such a case the exiting water would not collect in the sump and thus would no longer be available for emergency cooling. In addition, it must be considered that the components situated where there are leaks in the annulus may be involved. The various possibilities are discussed below.

In the emergency and RHR system, a loss of coolant in the annulus can only occur at full power operation when two check valves fail in one loop. This is the case when:

- there is an internal break with failure of the check action of both check valves
- there is an internal break of one check valve in connection with improper opening of the second check valve.

Failure of check action due to internal break is very unlikely. An incorrect setting of the check valves is generally impossible because of the various surveillance measures (e.g., position monitoring of valves). Since only the above failure combinations can lead to a loss of coolant to the annulus, analysis of these event sequences will not give any notable contribution to risk. A LOCA during RHR operation through a leak in the emergency and RHR system also does not significantly contribute to the risk.

In the first place, the probability of a pipeline fracture during the limited time of RHR operation is very low. Secondly, such a fracture, even if it should occur, would, in most cases, be blocked off very quickly. After successfully blocking the fracture, the emergency and RHR system would again be fully functional.

Pipelines of the volume control and chemical injection system, which are also linked to the reactor coolant system, normally carry only water in the region of the annulus which is at a temperature of less than 100°C and a pressure of

10 bar. Thus, for a pipe fracture in the annulus, no steam atmosphere will form. Large area flooding is negligible because of the event frequency; in addition, redundant blocking valves are automatically closed. Furthermore, a flooding of the annulus with about 300 m^3 water could be mastered by the design of the structures themselves.

A loss of primary coolant from the volume control and chemical injection system would also be possible by improper operation of the pressurizer water level control. The pressurizer water level is monitored from the power plant control room so that control room personnel can intervene with high probability. A reduction in pressurizer water level is also reported and a further reduction is prevented by reactor protection signals. Therefore, no contribution to risk is expected from such operational malfunctions.

5.2.2 Transients

5.2.2.1 Initiating Events and Measures to Cope With Them. Operational malfunctions that cause long-term inequilibrium between heat generation and heat removal without loss of coolant are called transients. The predominant number of transients are intercepted by operating systems. In those few cases where the operating systems are insufficient or where they fail, intervention of the safety systems is necessary. We speak of transient accidents only in those accident sequences where operation of the plant cannot be continued for safety reasons (Section 3.3.2.1). Transient accidents generally occur when triggered safety systems do not perform their functions.

During transients or transient accidents which are coped with by the operating and safety systems as designed, no notable release of fission products takes place; therefore, to determine risk, those transients should be studied where safety systems failure can lead to a core melt.

There are numerous reasons for transients. These initiating events and their effects cannot all be discussed individually. In order to obtain the most wide-ranging determination of the events contributing to risk, we shall proceed here as outlined below. In an initial step we determined which principal possibilities can lead to a loss of equilibrium between heat generation and heat removal. These are:

- change in power generation
- change in power withdrawal (feedwater input or steam removal)

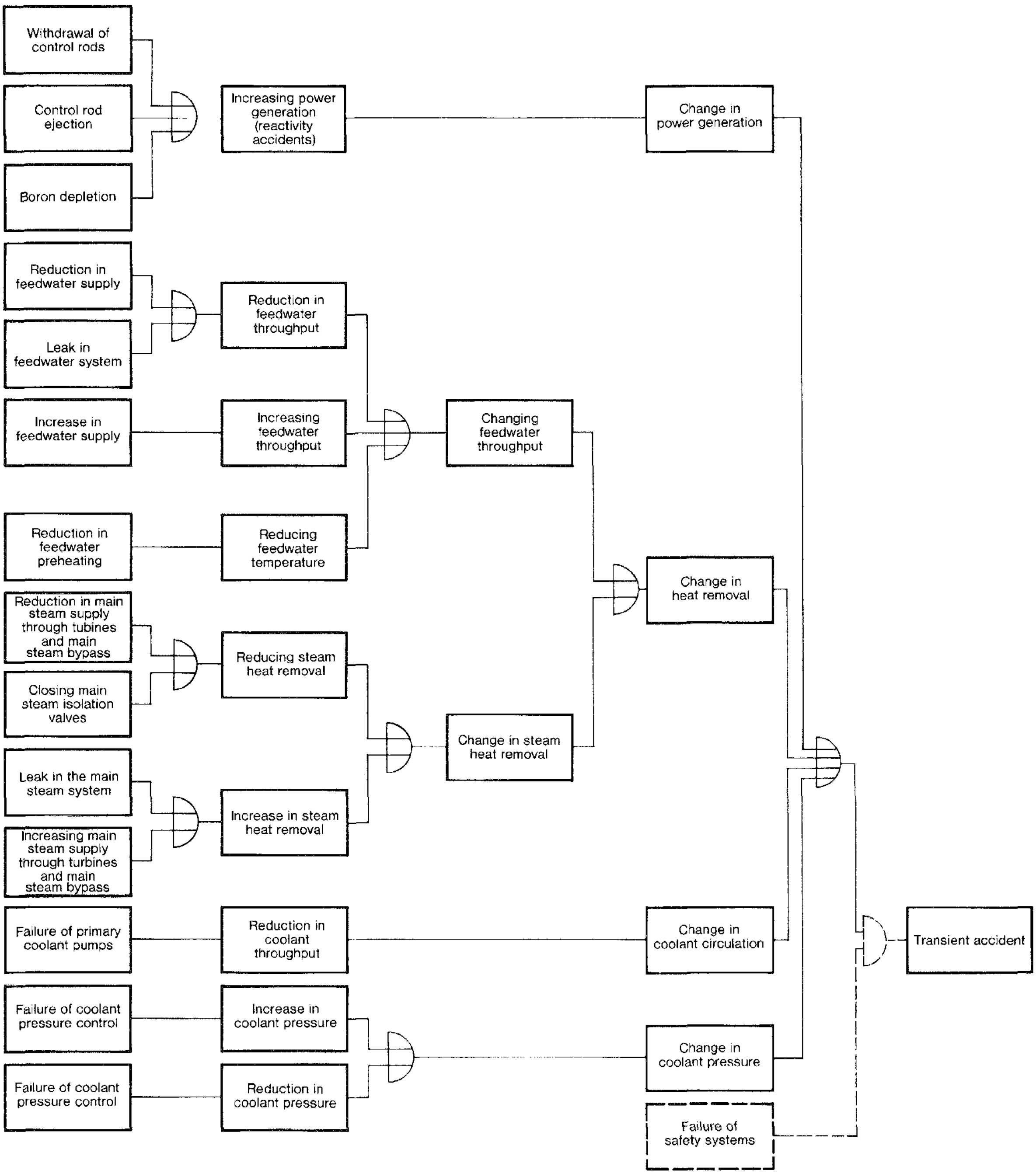


Figure 5-5. Causes of transients

- change in coolant circulation
- change in coolant pressure.

Changes in coolant temperature are only indirectly possible, as a consequence of the above named changes. Changes in coolant quantity (due to improper function of pressurizer water level control) are discussed in Section 5.2.1.5.

In the next step, these four potentials are further differentiated. The method is pursued until the important initiating events are found. Figure 5-5 shows the first step of this study.

In the determination of important initiating events, a simplification results from the fact that initiating events can be neglected if their effects are smaller in comparison to other events and if their frequency is unimportant.
The initiating events are divided into two groups with regard to frequency of occurrence analogous to WASH-1400:

- likely events whose frequency lies between about 10^{-2}/year and 10/year, but which is generally above 10^{-1}/year
- unlikely events with a frequency less than 10^{-2}/year, whereby the frequency is usually much smaller.

For the likely events, the frequency can often be estimated by analyzing the operating experiences of German nuclear power plants. If this is not possible, then the frequency is determined by means of fault tree analysis.

For the sum of frequencies of all initiating events of transients that require intervention by the safety systems, a value can also be estimated from operating experiences. If, in the course of a transient, a process quantity reaches a predetermined limit value, then reactor scram is triggered. The limit values are adjusted to prevent an overload of the reactor core. Reactor scrams thus occur more frequently than necessary to prevent damage to the reactor core. The frequency of initiated reactor scrams in nuclear power plants thus provides an upper limit for the sum of frequencies of all initiating events to be studied. On the basis of operating experiences in German plants, a value of five reactor scrams per plant per year is estimated.

"Unlikely" events are transients whose frequency is so small that they are not anticipated during the operating life of the plant. These unlikely events include, for instance, a break in the feedwater pipeline, a break in the main steam pipeline, or ejection of a control rod by the breakage of a nozzle at the reactor pressure vessel lid.

As was assumed in WASH-1400, the risk contribution of rare transients is small in comparison to the contribution of likely transients. The unlikely transients will therefore be studied in phase B of the risk study. In this regard, a rupture in the main steam pipeline in front of the main steam gate valves is a special case. Frequency and effects of such fractures therefore require urgent analysis in phase B.

To reduce damage to the reactor core during transients that require intervention by safety systems, various measures must take place:

- accomplishment and long-term assurance of subcriticality of the reactor core
- pressure limitation of the reactor coolant system
- assurance of a sufficient coolant inventory in the reactor coolant system
- heat removal from the reactor coolant system.

These measures are implemented by the reactor scram system, the volume control and chemical injection system, the reactor coolant system with its pressurizer system, the feedwater-steam system, and possibly the emergency cooling and RHR system. The following system functions are needed within the first hours from among the tasks to be performed by these systems according to the type of transient:

- reactor scram
- initiating of pressure relief of the reactor coolant system
- terminating of pressure relief of reactor coolant system
- main feedwater supply and main steam relief
- emergency feedwater supply and main steam relief
- delayed feedwater supply and main steam relief.

Pressure relief of the reactor coolant system takes place through the pressurizer relief valves and safety valves. If these valves do not open sufficiently, an overpressure failure of the reactor coolant system is possible. If valves have opened, but if some of them fail to close on demand, then the transient becomes a LOCA. With regard to measures taken in this case, see Section 5.2.1.1.

The main feedwater supply needed for full power operation is assured through the main feedwater pumps. The emergency feedwater supply is sufficient to remove decay heat and to shut down the plant (i.e., to reduce coolant temperature). This is quite possible by means of the emergency feedwater system and the emergency system. Main steam can be released through the main steam bypass mechanism and the turbine condenser, as well as by venting the steam directly to the outside.

Of these system functions, we distinguish between delayed feedwater supply and main steam relief. This distinction is needed if we succeed in resupplying feedwater to the steam generator only after evaporization of the secondary side of the steam generator. In the meantime, the pressure in the reactor coolant system increases so that opening of the pressure relief valve of the reactor coolant system is required.

The minimum requirements of individual system functions need to be discussed only for transients that do not result in LOCA. In order to prevent overheating of the reactor core during these transients, it is enough to keep the plant in the hot state; shutdown of the plant is not necessary. The reactor scram has a decisive influence on the minimum requirements. If the reactor scram should fail, then greater demands are placed on the other systems. In Table 5-2 we distinguish between likely transients with successful reactor scram and likely transients without scram (ATWS accidents).

Likely transients would contribute to risk if one of the above system functions had already failed due to the initiating event, or if the reliability of the required system functions had been diminished. Therefore, the maximum reduction of feedwater inlet, that is, the "failure of the main feedwater supply," is of particular interest as an initiating event (see Section 5.2.2.3). Power failure is not as frequent, but a distinction must be made because in addition to the main feedwater supply failure, main steam relief through main steam bypass mechanism, turbine condenser, and coolant circulation in the reactor coolant system have all failed.

Table 5-2. Minimum requirements of system functions during transients that require intervention by the safety systems (2 out of 4 means that of 4 existing redundant subsystems, two are needed.)

Transients	Opening of pressure relief valve in reactor coolant system	Closing of pressure relief valve in reactor coolant system	Feedwater supply a) Main feedwater b) Emergency feedwater c) Delayed feedwater
Anticipated transients with reactor scram	--*	Eventually 1 out of 4 or 2 out of 4**	a) 1 out of 4## or b) 1 out of 4 or c) 1 out of 4
ATWS accident, power failure	2 out of 3#	4 out of 4**	b) 2 out of 4
ATWS accident, failure of main feedwater supply	3 out of 3#	4 out of 4**	b) 2 out of 4
Other ATWS accidents	2 out of 3#	4 out of 4**	a) 2 out of 4##

* The pressurizer valves open in some transients; however, this opening is not necessary to prevent overpressure failure of the reactor coolant system except when feedwater supplies a and b fail.

** If any pressurizer valve fails to close, the transient becomes a LOCA.

\# Of interest here are the three pressurizer valves with the large valve cross-section.

\## One out of 4 or 2 out of 4 injections via the main feedwater lines are necessary.

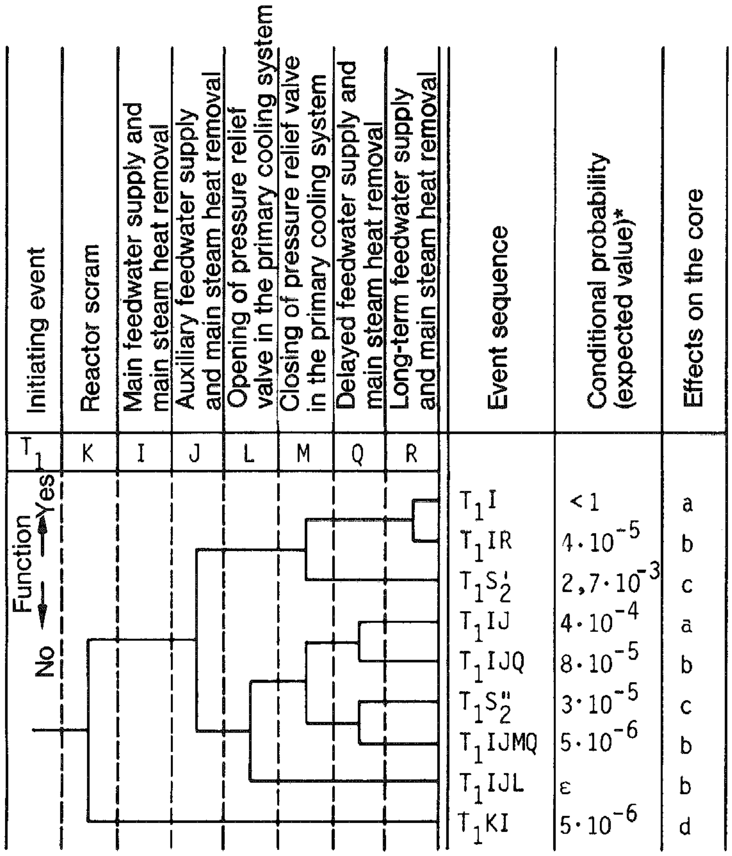
Furthermore, the unavailability of some of the system functions needed to cope with the power failure is greater than for other transients because only the emergency power system is available to supply electricity. Power failures are discussed in detail in Section 5.2.2.2.

These two transients place special demands on system functions for feedwater supply and main steam relief. A contribution to risk is also possible based on the failure of other system functions. For instance, the reactor scram is initiated for all likely transients that require activation of safety systems. If they fail, then ATWS accidents result; according to Table 5-2, these place the greatest demands on the opening and closing of the pressure relief valve of the reactor coolant system (see Section 5.2.2.5). Opening of the pressure relief valve is also needed in cases of power failure and main feedwater supply failure transients if the emergency feedwater supply also fails. Since this is discussed in the event tree analyses for these transients, a special study is not needed at this time.

Closing of the pressure relief valve of the reactor coolant system is required not only for ATWS accidents, but also for various likely transients after successful reactor scram where pressurizer valves open. The minimum requirements for the closing of the pressure relief valve are smaller, but the transients are much more frequent than ATWS accidents. If one of the pressurizer valves does not close, then a "small leak in the pressurizer" results (see Section 5.2.2.4).

5.2.2.2 Power Failure. In normal operation, power supply to the nuclear power plant takes place through the four 10-kV busbars of the auxiliary power system (see Section 3.4.7). A power failure exists when the voltage to more than one of these 10-kV bus-bars fails. Since the failure of all four 10-kV busbars of the auxiliary system, that is, the complete failure of the auxiliary power supply, is more frequent than the failure of two or three of the 10-kV busbars, we always assume a complete failure of the auxiliary power supply.

During power failure, electricity supply to the 10-kV emergency power busbars takes place through the emergency diesel generators. In order to limit operating time of this emergency power generators, one must try to draw at least the power needed to supply emergency consumers from the interconnecting grid or from the connecting line to unit A as soon as possible.



a No core melt
b Core melt
c Continuation "small leak at pressurizer during power failure"
d Continuation "ATWS" accidents

*Probability of the individual event sequence assuming that the initiating event has occurred. The frequency of the individual event sequence is obtained by multiplication with the frequency h of the initiating event $h(T_1) = 0.1$/year (expected value)

Figure 5-6. Event tree for "power failure"

The event sequences for power failure are illustrated in Figure 5-6. Failure of the auxiliary power supply leads to a reactor scram--triggered initially by unacceptable decrease in RPM of the main coolant pumps. Because of the failure of the 10-kV busbars of the auxiliary power supply, the main feedwater supply is not available, and valves of the main steam bypass mechanism do not open.

Heat is removed via the emergency feedwater supply and steam relief system. The relief valves and safety valves are used to release steam. Opening of the pressure relief of the reactor coolant system is necessary during temporary failure of the emergency feedwater supply, in order to limit pressure increase in the reactor coolant system after secondary vaporization of water from the steam generator. Even if the emergency feedwater supply is operating, the first pressurizer valve reaches the response pressure so that closing of the pressure relief valve of the reactor coolant system is needed to prevent the power failure from turning into a LOCA. If a pressurizer valve fails to close, then the result is a "small leak in the pressurizer" (see Section 5.2.2.4). Upon failure of the emergency feedwater supply, a delayed feedwater supply and main steam relief system must be taken into operation for up to about 75 minutes after the incidence of the power failure in order to prevent the reactor core from overheating. During this time, a feedwater supply from unit A must be established by means of the emergency system. If the water supply in the feedwater tank is used up, then operation of the long-term feedwater supply and main steam heat sink system is necessary. Water is pumped by the deionate pumps and the deionate pressurizing pumps from the deionate tanks into the feedwater tanks. If this is not possible, then the emergency feedwater pumps are switched over to draw water directly from the deionate tanks. In this case, the deionate pumps and deionate pressurizer pumps are needed for emergency feedwater loops one and two.

The following possibilities exist for initiation of a power failure:

- failure of the auxiliary transformer
- failure of the 27-kV busbars
- failure of both AC mains and failure of shutdown to auxiliary power
- failure of one AC main and failure of shutdown to partial load and auxiliary operation
- failure of the conventional secondary cooling system
- turbine trip and failure of the generator circuit breaker

- turbine trip and failure of the auxiliary power supply from the interconnecting grid.

An estimated value of 0.1/year is given for the frequency of power failures. This is based on the failure of individual components in the auxiliary power system and with regard to mains failures. The auxiliary power system components provide the major contribution; for instance, failure of the auxiliary power transformers is set at 4×10^{-2}/year (failure of one of the two transformers is already enough to cause blackout of the auxiliary power supply).

The probabilities determined for the various event sequences are shown in Figure 5-6 under the assumption that a power failure has occurred. Probability for the failure of system functions needed to cope with a power failure was set at a total of 1.3×10^{-4}. The frequency for a core melt accident due to a power failure is thus (0.1/year) $\times 1.3 \times 10^{-4} = 1.3 \times 10^{-5}$/year.

The sequence of power failure is determined primarily by the operation or failure of the emergency feedwater system and main steam relief. If systems function, only the long-term feedwater supply is needed to cope with the power failure. Failure of the long-term feedwater supply contributes about 30% to the total result for the power failure. By simple measures with regard to deionate injection, a significant improvement can be achieved here.

When the emergency feedwater supply and main steam relief fail, the power failure can be coped with by establishing a feedwater supply from unit A within 75 minutes after the beginning of the accident (event sequence T_1IJ). If this delayed feedwater supply also fails, then with regard to the frequency of power failure, we obtain a frequency for this event sequence of (0.1/year) $\times (5 \times 10^{-4}) \times (1.6 \times 10^{-1}) = 8 \times 10^{-6}$/year.

This value represents 64% of the total results for power failure. Unavailability of emergency water supply of 5×10^{-4} up to 80% is caused by common cause failures of the emergency diesel generators. The unavailability of the delayed feedwater supply and main steam heat sink is 0.16. It is important here that manual intervention be taken in the annulus of unit A. In addition, hardware failures of the emergency system may make significant contributions.

The risk amount due to power failure can be considerably reduced by a return to grid power. The emergency power busbars were originally so biased that when the emergency diesel generator failed, the attendant 10-kV emergency busbars could no longer be switched back to the corresponding 10-kV auxiliary busbars, even if voltage was again available there. For a return to grid power, the connecting lines to train A of the Biblis plant are quite suitable. During the refueling of June-August 1978, the possibility for a return to grid power was established. Since corresponding documentation has not yet been completed, only a rough estimation can be made. Accordingly, the frequency of a core melt accident as a result of a power failure would be reduced to about 10^{-5}/year.

5.2.2.3 Failure of the Main Feedwater Supply. Failure of the main feedwater supply can have various causes. To determine the contributions to the risk, we must distinguish whether the cause is a power failure or not. The power failure was discussed in detail in Section 5.2.2.2. Therefore, we need only to determine whether a simultaneous failure of main feedwater pumps occurs without failure of the auxiliary power supply. Whereas a frequency of power failure of 0.1/year was estimated, we obtain from operating experiences in Germany a frequency for the failure of the main feedwater supply without power failure of 0.8/year.

If the auxiliary power supply is available to cope with the accident, then this has a favorable effect on the system functions demanded. However, a reactor scram occurs later than a power failure. It is usually triggered by an unacceptable drop in secondary steam generator water level. When the main feedwater supply fails, an emergency feedwater supply and main steam relief must be established, or the main feedwater supply and main steam relief must be restarted briefly. If no secondary feed to the steam generators is established within a few minutes, then a reactor protection signal prevents reestablishment of the main feedwater supply. If there is no feed to the steam generators, the steam generators are dried out sooner because of the later timing of reactor scram in comparison to a power failure. Up to about 40 minutes after the beginning of the accident, a delayed feedwater supply and main steam relief must be generated to prevent unacceptable overheating of the reactor core. This results in an increase in unavailability of emergency feedwater supply and delayed feedwater supply.

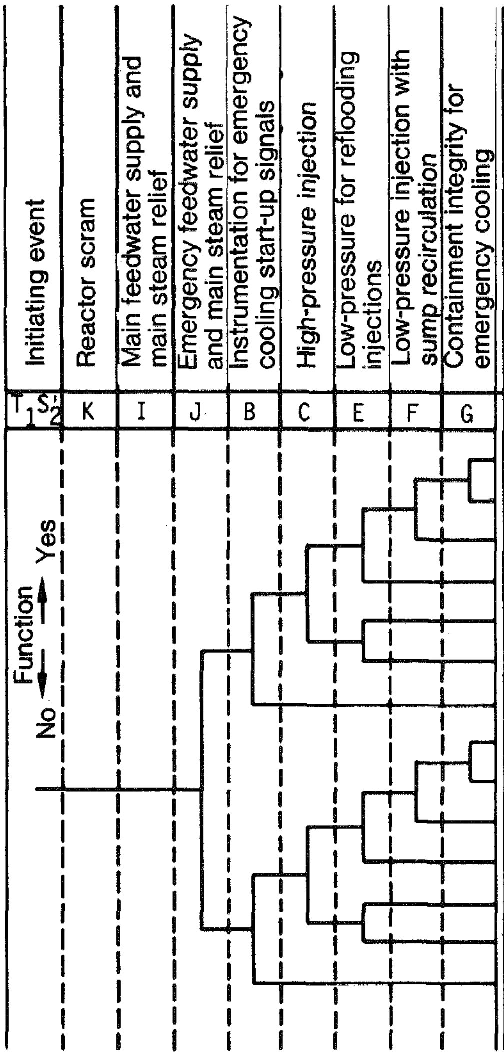
The accident "failure of main feedwater supply" leads to a core melt frequency of 3×10^{-6}/year for successful reactor scram. The contribution to core melt frequency upon failure of reactor scram is not discussed here but is treated jointly with other ATWS accidents (Section 5.2.2.5).

5.2.2.4 Small Leak in the Pressurizer. The reference plant design data as well as the instrumentation and control differ (especially with regard to control rod insertion) in part from other German nuclear power plants. Therefore, there are differences in the transients that result in opening of the pressurizer valves. The previously most frequent cause for opening the pressurizer valves (partial failure of rod insertion) has been eliminated and is no longer anticipated. From previous German operating experiences, the frequency of the opening of pressurizer valves can therefore be roughly estimated as 0.5/year for the reference plant.

The transients that lead to opening of pressurizer valves in the reference plant are summarized in Section 5.2.1.4. Basically, we must distinguish between power failure and transients without power failure. Power failure has a considerable influence on the probability of failure of other system functions. Small leaks at the pressurizer, which arise from a power failure through the failure of pressurizer valves to close, are discussed in detail below. Small leaks at the pressurizer due to all other transients provide a much lower contribution to risk. An average frequency of core melt accidents has been estimated for them in the range of 2 X 10^{-6}/year.

The event sequences for a small leak at the pressurizer caused by power failure are illustrated in Figures 5-7 and 5-8. The two event trees are directly connected to the event sequences T_1S_2', or T_1S_2'' of the power failure event tree (Figure 5-6). The structure of the diagram is similar to the diagram for a small leak in a primary coolant pipeline (Figure 5-4). The reactor scram is no longer indicated since it was already triggered by the power failure. The emergency feedwater supply and main steam relief must be reconsidered, even though they are already contained in the diagram for power failure. In contrast to the power failure, where feedwater supply must occur within 75 minutes, 2 to 3 hours are available for a small leak in the pressurizer. After this period of time, the plant must be shut down. Here the same minimum requirements apply as were found for emergency feedwater supply and main steam relief for a small leak in a primary coolant pipeline.

The occurrence probabilities for small leaks in the pressurizer due to a power failure are determined from the event tree for the power failure (see Figure 5-6) as (0.1/year) X 2.7 X 10^{-3} = 2.7 X 10^{-4}/year for the case T_1S_2' or (0.1/year) X 3 X 10^{-5} = 3 X 10^{-6}/year for the case T_1S_2''. The sums of probabilities for failure of system functions needed to cope with a small leak at the pressurizer are obtained from Figures 5-7 and 5-8. With regard to rounding errors, we obtain

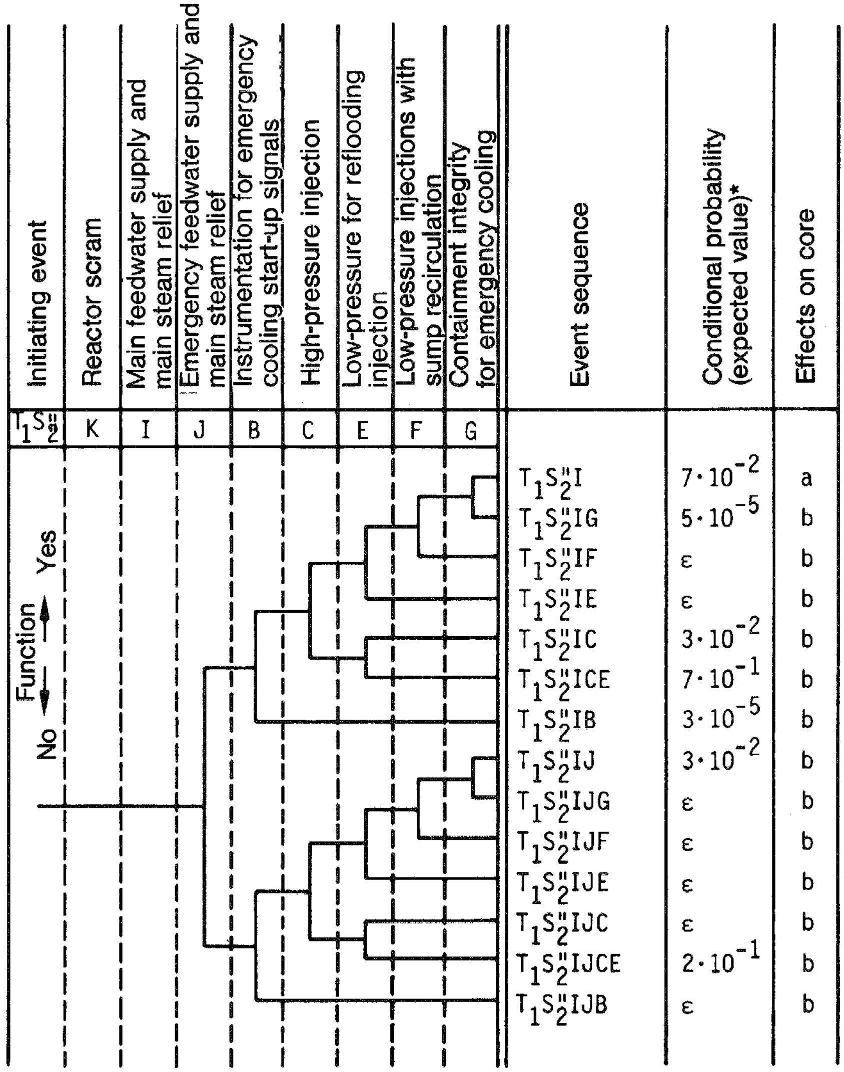


Event sequence	Conditional probability (expected value)*	Effects on the core
$T_1S_2'I$	≈ 1	a
$T_1S_2'IG$	$5 \cdot 10^{-5}$	b
$T_1S_2'IF$	ε	b
$T_1S_2'IE$	ε	b
$T_1S_2'IC$	$2 \cdot 10^{-3}$	b
$T_1S_2'ICE$	$1 \cdot 10^{-2}$	b
$T_1S_2'IB$	$3 \cdot 10^{-5}$	b
$T_1S_2'IJ$	$7 \cdot 10^{-4}$	b
$T_1S_2'IJG$	ε	b
$T_1S_2'IJF$	ε	b
$T_1S_2'IJE$	ε	b
$T_1S_2'IJC$	ε	b
$T_1S_2'IJCE$	$2 \cdot 10^{-3}$	b
$T_1S_2'IJB$	ε	b

a No core melt
b Core melt

*Probability of the individual event sequences assuming the initiating event has occurred. The frequency of the individual event sequence is obtained by multiplication with the frequency h of the initiating event $h(T_1S_2') = 2.7 \times 10^{-4}$/year (expected value)

Figure 5-7. Event tree "small leak at pressurizer during power failure" T_1S_2'



a No core melt
b Core melt

*Probability of the individual event sequences assuming the initiating event has occured. The frequency of the individual event sequence is obtained by multiplication with the frequency h of the initiating event. $h(T_1S_2'') = 3 \times 10^{-5}$/year (anticipated value)

Figure 5-8. Event tree "small leak at pressurizer during power failure" T_1S_2''

1.5×10^{-2} or 0.93. Probabilities less than 1% of these values are designated in the figures by ϵ, provided they are of no particular interest. From the occurrence probabilities for small leaks in the pressurizer and the sums of probabilities for the failure of needed system functions, we obtain frequencies of 4.1×10^{-6}/year or 2.8×10^{-6}/year for a core melt accident. Overall, a core melt will result from a "small leak in the pressurizer with power failure" at a frequency of 7×10^{-6}/year.

Both in the case T_1S_2' as well as T_1S_2'', the unavailability of the power supply for high-pressure injection and low-pressure injection is an important factor. For instance, if we follow event sequence T_1S_2', a common cause failure of the diesel generators is found in about 45% of the failure combinations that lead to a small leak in the pressurizer as well as failure of the needed high-pressure injection. For T_1S_2'' this figure is almost 90%. Whereas for T_1S_2' a common cause failure of the diesel generators, together with an independent failure of hardware, results in an uncontrolled accident, additional human error is needed for T_1S_2''.

5.2.2.5 ATWS Accidents. A reactor scram is needed in German PWR power plants with an average frequency of 5/year. As a result of this requirement, potential damage to fuel elements is supposed to be avoided. If the reactor scram should fail, the pressure relief of the reactor coolant system would have to limit any possible increase in coolant pressure. In addition, a sufficient heat removal through the feedwater-steam system would be needed. Such ATWS accidents do not result in any important contribution to core melt frequency because of the high reliability of the reactor scram (the average unavailability is 5×10^{-6}). Overheating of the core can only result from such accidents when additional failures of needed subsystems occur.

The greatest coolant pressure rise would occur for an ATWS accident "failure of the primary coolant supply" (frequency of initiating event 0.8/year, see Section 5.2.2.3). For this accident, it is assumed that a satisfactory opening of the pressure relief of the reactor coolant system occurs only when the three pressurizer valves of larger valve cross-section open (Table 5-2). For failure of one of the three valves to open, we estimate a probability of 1.2×10^{-1}. On the basis of this event sequence (designated as T_2KL), a contribution of (0.8/year) $\times (5 \times 10^{-6}) \times (1.2 \times 10^{-1})$ equal to 5×10^{-7}/year to the frequency of core melt accidents can be expected.

During ATWS accidents, all four pressurizer valves normally open. If the coolant pressure continues to drop, then closing of the pressure relief of the reactor coolant system, i.e., all pressurizer valves, is required. Otherwise, the ATWS accident becomes a LOCA "small leak at pressurizer." In Section 5.2.1.1, it was pointed out that the same minimum requirements for a leak in the pressurizer were assumed as for a corresponding leak in a primary coolant pipeline. Therefore, the pessimistic assumption is made for such a leak in the pressurizer that a failure of reactor scam results in core melt.

The probable failure of one of the four pressurizer valves to close is determined as 2.5 X 10^{-2}. The event sequence TKM (T stands for the sum of all anticipated transients) thus leads to a core melt frequency of (5/year) X (5 X 10^{-6}) X (2.5 X 10^{-2}) equal to 7 X 10^{-7}/year.

The two event sequences discussed above provide the primary contributions of ATWS accidents to core melt frequency. Other event sequences consequently are of no importance.

5.2.3 Failure of the Reactor Pressure Vessel

At the end of 1977 there were 206 high-power reactors in operation worldwide; at that time they had collectively operated for about 1500 reactor-years (1). The rupture or burst of a reactor pressure vessel, abbreviated below as the RPV, has not occurred. Therefore, it is not possible to determine the failure frequency of the RPV on the basis of available operating experiences.

A significantly more reliable statement about failure probability can be made for the RPV in other technical spheres as, for instance, in boiler operation and in the chemical industry, with their years of operating experience. However, failure frequencies obtained from nonnuclear pressure vessels cannot be directly applied to a RPV, since the two groups of vessels are not entirely comparable. The reasons are:

1. There are significant differences between the rated data of a nonnuclear pressure vessel and that of a RPV. This indicates a statistical evaluation of various characteristics of nonnuclear pressure vessels under surveillance by the Technical Inspection Agencies, according to which 95% of the nonnuclear pressure vessels have a wall thickness of less than 20 mm (RPV: 350 mm for the PWR in the cylindrical part), the stored energy is lower by 95% than in the RPV and the nominal stress of 85% is only 100 N/mm^{-2} and of 50%, only 70 N/mm^2 (RPV: 180 N/mm^2) (2).

2. There are considerable differences between nonnuclear pressure vessels and a RPV in design and construction, as well as in other parameters affecting quality, including monitoring and testing measures.

To improve quality of the RPV, we must go beyond the demands of nonnuclear pressure vessels:

- determination of total loads from all realistically anticipated operating and accident conditions, including rare, extreme states
- stress analysis for all components based on the above loads
- optimized construction with regard to load transmission processing, and testing
- purity of material
- strength of material
- good processing properties
- surveillance of welding
- surveillance of heat treatment
- multiple, independent ultrasonic testing of welded seams after heat treatments and after the first pressure test
- recurrent, nondestructive testing during operation
- surveillance of long-term behavior by suspended lead test specimens.

An RPV therefore differs in design and quality from a nonnuclear pressure vessel. A factual evaluation of these differences leads to the conclusion that the RPV has a significantly greater reliability than the nonnuclear pressure vessel because of standards and code requirements in nuclear engineering.

The estimated failure probabilities of the RPV will apply only to those failure modes whose origin is found in the design, construction, material, and processing--that is, in the materials themselves (including improper sizing and incorrect material selection).

The most important measures implemented on Biblis class RPVs to assure the high quality needed for reactor operation are discussed briefly below.

Design and Construction

All anticipated loads occurring during operation were examined within the framework of a stress analysis with respect to their effects on the RPV. The result is accurate information about occurring stresses which are limited in regard to load type and material behavior. In addition, the RPV design exhibits a structure capable of load transmission.

The permissible primary liner stress in the RPV, which is about 180 N/mm^2, is thus greater than that normally found in nonnuclear vessels. The assured tensile strength safety margin at room temperature (568 N/mm^2) is a factor of three. As a result of the detailed stress analysis, the stress capacity measured for the material is unquestionably satisfactory. In addition, the optimized construction has a favorable effect on the following conditions:

- no wall penetrations beneath the vessel flange
- extra reinforced vessel flange with mounted nozzles
- particularly reinforced lid in the region of the control rod penetrations
- no longitudinal welded seams as a result of the use of seamless forged rings
- limitation of radiation dosage to vessel wall as a result of a sufficiently large water gap.

Design and construction of the RPV were rechecked and reevaluated by independent experts.

Material Selection

The base material of the RPV is 22 NiMoCr 37. The properties of this material are known and statistically confirmed by a large number of tested components. Chemical analysis, melting, processing, and heat treatment are coordinated so that the established minimum values for strength and ductility will be maintained at all points, even in the weld metal and in the heat-influenced zone; in addition, these values do not fluctuate unacceptably over wall thickness or total volume. This was demonstrated by extensive, destructive material testing.

The integral ductility achieved and the proven, limited number of flaws assure sufficient resistance against anticipated loads, even if, with regard to base material and heat influenced zone, no exceptionally optimum steel is available. In terms of the initial operating phase of about 10-20 years, we thus have no reservations about safety against catastrophic failures. Even for a second operating phase of an additional 20 years, according to current knowledge, an unacceptable reduction in the safety margin is not anticipated. Final confirmation for this confidence must be found in the constantly improving status of science and technology at each particular time of evaluation. The assurance of sufficient knowledge is in turn attained by evaluation of operating experiences and vigorous continuation of efforts in research and development.

In principle, the same quality requirements as for the vessel shall apply to the bolts of the lid as well.

Manufacture

Important manufacturing processes like welding and heat treatment are monitored constantly during their implementation and inspected by independent experts. By means of simulations and by test specimens accompanying manufacture it was demonstrated that the necessary material properties are retained even after processing and manufacture.

Microcracks and cracks measuring in the millimeter range are possible in coarse-grain 22 NiMoCr 37 material in superheated regions of the heat-affected zone. As a result of the multi-layer welding techniques, the expansion of these cracks into the interior of the vessel walls is limited so that during extensive, mechanical testing, no reduction in resistance could be found. For the first operating phase, therefore, no reservations about safety statements are needed. Furthermore, for future operating phases the same statement applies as was made above with regard to the material.

Component Testing and Initial Pressure Testing

During manufacture, quality was checked continuously by means of nondestructive test methods. The most important of these is ultrasonic testing. This was used on welded seams immediately after intermediate heat treatment, after final heat treatment, and after the initial pressure test, in the form of triple testing performed by manufacturer, system deliverer, and independent experts. By means of this multiple testing, changes in the manufacture could be followed and human error could be eliminated. After the entire pressure vessel was assembled, it was subjected to an initial pressure test at 1.3 times the design pressure to demonstrate that the strength to accomodate the operating conditions is present.

Recurrent Testing During Operation

The RPV is recurrently checked by means of generally automated ultrasonic testing and hydraulic pressure testing to determine its status, so that it remains under control with advancing service life in accordance with operating conditions.

In-Service Inspection

The operating behavior and functioning of the entire reactor coolant system are tested within the framework of commissioning. During subsequent stages of full power operation, temperatures and pressures are monitored and recorded, in addition to frequencies of the various load states. This assures that stress of the reactor pressure vessel is monitored during operation and can be checked at all times.

As a result of the neutron irradiation, the ductility of the base material changes in the region of the reactor core. In order to keep this change as small as possible, the neutron flux integrated over the service life is limited by a water gap between core and vessel wall to $1 \times 10^{19}/cm^2$. The change in ductility is monitored through so-called suspended specimens--these are lead material samples exposed to operating conditions and to neutron irradiation, which are removed successively during power plant operations--the change in strength is monitored. This assures that the ductibility is known with advancing age and that a limit corresponding to operating conditions is not exceeded.

After a summary evaluation of all measures implemented to determine and protect against stresses and to generate and demonstrate quality, the potential for a catastrophic failure of the RPV due to inherent causes cannot be recognized. Experts in the area of pressure vessel engineering agree with this statement. Performance and quality of the RPV are such that a rupture due to unsatisfactory design, materials, or manufacture is practically precluded, according to expert opinion, under the conditions outlined above.

However, these statements also show how problematic it is to make a quantitative determination of RPV reliability. For this reason, with reference to WASH-1400, we accepted the failure probability of 1 X 10^{-7} per RPV year of operation as a calculated value for phase A of this German study for the assumed accident rupture of the reactor pressure vessel due to a defect in design, material, or processing.

In Great Britain estimations performed on the basis of probabilistic fracture mechanics lead to values of the same order of magnitude (4).

We should mention that not every failure of the RPV will result in immediate damage to the containment and thus cause a rapid release of fission products.

The judgment of the RPV made in WASH-1400 is based on evaluations of operating experiences from nonnuclear pressure vessels (3). Operating experiences for nonnuclear pressure vessels in the FRG were evaluated to recheck and validate the failure frequencies determined there for nonnuclear pressure vessels. The evaluation extended from 1959 to 1976 and included available statistics on numbers and damage of such vessels. This gave an average failure frequency for rupture of a pressure vessel of about 10^{-5} per year of pressure vessel operation (2).

An important result of these studies was that the failure probability of pressure vessels in the time period under consideration decreased by nearly one order of magnitude. This increase in reliabiiity can be attributed primarily to expanded and improved manufacturing and to operational monitoring implemented in the meantime. The failure probability determined for nonnuclear pressure vessels can be applied only conditionally to RPV evaluation. It can be used initially to verify the high reliability achieved in the meantime by nonnuclear pressure vessels. Results confirm the reliability of the assumed failure frequency for the RPV which--as presented above--has a significantly greater reliability because of better construction and quality.

For the type of RPV design under consideration here, it can be maintained that no penetration of the pressure retaining wall occurs beneath the primary coolant nozzles. Estimations of leak cross-sections by means of fracture mechanical methods gave cross-sections of less than 30 cm^2 for the greatest, subcritical crack length in the cylindrical wall.

A LOCA through such a leak cross-section can be coped with at any time by means of the emergency and residual heat removal system. With regard to the measures and, in particular, to the recurrent testing (hydraulic pressure testing and nondestructive ultrasonic testing), taken with regard to the RPV, crack lengths of this size are significantly less likely than leaks in the primary coolant pipeline. Leaks in the area of the flange connection or control rod penetrations are covered by the measures taken in a LOCA. With the frequency of 10^{-7} per RPV operating year for the rupture of a RPV, no relevant contribution to risk results from an assumed failure of the pressure vessel.

5.2.4 Effects Due to Fire

In the present study the effects of fire on the risk are treated qualitatively, like the procedure used in WASH-1400.

Within the framework of licensing procedures for nuclear power plants, fire protection is discussed in detail. Reference is made to the various measures to counteract fires in Section 3.4.12.

In order to permit a quantification of risk in phase B of the study, the following work has been performed:

- determination of potential fire sources in all buildings of the nuclear power plant with reference to potential damage
- determination of passive and active fire prevention measures
- working out methodological approaches for a systematic, quantitative analysis.

The individual work is described in detail in the appendix. Previous work indicates no significant contribution to core melt frequency.

5.2.5 Other Plant-Internal Accidents

Whereas previous sections dealt with accident sequences in the reactor core, which could result in release of radioactive fission products from the reactor coolant system, other important activity retaining components of the nuclear power plant will be studied below with regard to their radioactive inventory and potential radionuclide release.

Important radioactive components outside the core are spent fuel elements unloaded from the reactor core and the radioactive auxiliary systems of the nuclear power plant. The nuclide inventory contained here is important both with respect to nuclide composition and to total radioactivity. In an estimation of the potential consequences of accidents, fission product barriers other than for the reactor core must be taken into consideration. For example, spent fuel elements are always handled under several meters of water within the containment. The corresponding fission product barriers are: the fuel cladding, the water layer of the fuel storage pool, and the containment with its ventilation system. During removal of spent fuel elements from the system, the fission product barriers of cladding and fuel transport container are present.

In phase A of the risk study, only those accidents are carefully studied which would result in the release of radioactive substances outside the reactor core and would simultaneously allow an extrapolation of conditions from the PWR reference plant used in WASH-1400 to reference plant Biblis B. Such accidents include destruction of fuel cladding during handling of spent fuel elements, with subsequent release of radionuclides into the containment, and damage to a transport container for spent fuel elements and of the fuel elements being transported therein, with subsequent release to the environment outside the plant.

Fuel elements are removed from the reactor core after reaching their maximum burnup in the annual refueling. They are then placed in the fuel storage pool. After one-half year decay time, at the earliest, spent fuel elements are removed from the plant in transport casks. Damage to the individual fuel elements, possible during handling and returning them into the storage pool, can result in a release of radionuclides. Representative accidents that could result in destruction of fuel elements would be a fuel element handling accident within the storage pool, and dropping outside the reactor building a transport cask held by the portal crane at maximum height and loaded with spent fuel elements.

Fuel Element Handling Accident

In this accident, it is assumed that a fuel element falls from the grab of the refueling machine. The fuel element is damaged upon impact so that in the worst case, all claddings lose their integrity. Thus, fission products collected in the fission gas plenum can be released through the pool water to the atmosphere of the containment.

Table 5-3. Summary of the results of the event tree analyses

LOCA Transient	Frequency of initiating event h (expected value) per year	Conditional probability of failure of the required system functions w (expected value)	Frequency of core melt accidents h x w (expected value) per year
Large leak in a primary coolant pipeline	2.7×10^{-4}	1.7×10^{-3}	5.0×10^{-7}
Medium leak in a primary coolant pipeline	8.0×10^{-4}	2.3×10^{-3}	2.0×10^{-6}
Small leak in a primary coolant pipeline	2.7×10^{-3}	2.1×10^{-2}	5.7×10^{-5}
Loss of power	1.0×10^{-1}	1.3×10^{-4}	1.3×10^{-5}
Failure of the main feedwater supply	8.0×10^{-1}	4.0×10^{-6}	3.0×10^{-6}
Small leak in pressurizer during power failure	2.7×10^{-4}*	2.6×10^{-2}	7.0×10^{-6}
Small leak in pressurizer during other transients	1.0×10^{-3}*	2.0×10^{-3}	2.0×10^{-6}
ATWS	3.0×10^{-5}	3.0×10^{-2}	1.0×10^{-6}

*The frequency of the small leak in the pressurizer is obtained from the frequency of opening a pressurizer relief valve (0.1 per year during power failure, 0.4 per year for all other transients) by multiplication with the conditional probability 2.7×10^{-3} that the relief valve and its redundant block valve do not close.

Dropping of a Transfer Cask Containing Spent Fuel Elements

The element transport casks are designed so that deformation may occur upon impact from 9 m drop height to smooth concrete, but their seal will not be affected. During removal of spent fuel elements from the Biblis B nuclear power plant, the maximum lifting height (potential drop height) is 21 m and thus is above the corresponding cask specification. For a potential drop from this altitude a loss of cask integrity, connected with damage to fuel cladding, and subsequent release of radionuclides to the environment is possible.

Studies previously performed on the release of fission products from radioactive components outside the reactor coolant system do not indicate any dominant contribution to risk. This is attributed primarily to the fact that the components under discussion have a radioactivity that is small compared to the core radioactivity (see also Table 4-1).

5.2.6 Discussion of Results for Plant-Internal Accidents

In the previous sections we discussed event sequences proceeding from the initiating events. The physically different event sequences were pointed out and assigned probability values. Thus, the contribution to the frequency to core melt accidents could be determined.

To determine the frequencies of fission product release from the reactor building, we also need to consider the failure potentials and attendant probabilities for the containment. These probabilities depend on the particular event sequences, so that a dominant value for core melt frequency will not necessarily result in the greatest frequency of fission product release. In Section 6.6.3 a separate discussion of frequencies of fission product release to the environment will be given.

The frequencies of core melt determined by the event sequence analyses are compiled in Table 5-3. The sum of core melt frequencies is thus 9 X 10^{-5}/year. The parameters of the attendant distribution functions are given in Table 5-4. The greatest contributions come from small leaks in the primary coolant pipeline and from power failure. In the former case, this is attributable both to the greater frequency in comparison to the other leaks in a primary coolant line, as well as to the unavailability of system functions, whereby manual efforts determine the results. In the second case, the cause is the relatively high frequency of the initiating event. A power failure becomes a "small leak in the pressurizer during power failure" accident if the pressurizer valves fail to close; this also provides an important contribution to the total core melt frequencies.

Table 5-4. Sum of frequencies of core melt accidents

Expected value	9×10^{-5}
Median (50% fractile)	4×10^{-5}
Lower limit (5% fractile)	1×10^{-5}
Upper limit (95% fractile)	3×10^{-4}

The values listed are frequencies per year of operation.

The LOCA and transients that were discussed in more detail on the basis of event trees, lead to about 93% of the total core melt frequency. The attendant fractional contributions of failures of system functions are seen in Table 5-5. The reference value is the frequency of core melt due to the particular initiating event. The influence of one and the same system function is often different because:

- different system functions are needed to cope with different accidents
- the minimum requirements can differ
- the probability of failure of system functions depends on the particular event sequence.

This is quite clear, for example, in comparing accumulator injections for large and medium leaks (see Section 5.2.1.2).

An overview of the influence of independent hardware failures, common cause failures of hardware, and human error is compiled in Table 5-6. Since not all of these failure modes necessarily result in failure to cope with the accident, those combinations were also included, e.g., independent failures and common cause failures. The AND connections found in the table are thus to be taken as logic connections; they result directly from the fault tree analyses.

Individual failures of hardware or human error do not result in failure to cope with an accident for any of the initiating events. Double failures, however, contribute to the result in a few cases: for a large and medium leak, double failures of accumulator injection cause an uncontrolled accident; for small leak in a primary coolant pipeline, the manual efforts to prepare shutdown at 100°C/h determine the result. It was assumed here that implementation of manual efforts is monitored. Double failures of hardware, which prevent shutdown of the plant at 100°C/h, are of no significance.

For large and medium leaks in a primary coolant pipeline, the safety systems needed to cope with the accident are automatically set in operation. Human error therefore plays a role in these accidents only with respect to calibration of measurement channels (common mode failures due to human error). Of the common cause failures of hardware, failures of the RHR pumps during long-term emergency residual heat removal are important.

For a leak in a primary coolant pipeline, incorrect manual efforts to prepare shutdown contribute about 78% to the risk (see Section 5.2.1.3). Additional manual efforts are of subordinate significance by comparison. Common cause failures due to incorrect calibration of measurement channels amount to only 3%. The 1% common cause failures of hardware also affect the measurement channels.

If a power failure occurs and if the emergency feedwater supply fails due to independent failures or common cause failures of the emergency diesel generators, then it is possible to manually operate the emergency system. Thus, an emergency feedwater supply from unit A can be established. Common cause failures of hardware alone, therefore, provide no contribution to the risk. Human error alone also plays no role since, in case of power failure, normally all measures are implemented automatically. Failures due to incorrect calibration can be neglected

Table 5-5. Contributions of system functions to the frequency of failure to cope with an accident.

LOCA transients	Frequency of failure to cope with the accident	System function	Accident sequence	Contribution to the frequency of failure to cope with an accident
Large leak in a primary coolant pipeline	5.0×10^{-7}/a	Accumulator injections, low-pressure flooding injections, low-pressure injections with sump recirculation, long-term decay heat removal, other	AD AE AF AH	42% 12% 30% 12% 4%
Moderate leak in a primary coolant pipeline	2.0×10^{-6}/a	High-pressure injections, accumulator injections, low-pressure flooding injections with sump recirculation, long-term decay heat removal, other	S_1C S_1D S_1E S_1F S_1H	52% 9% 4% 21% 9% 5%
Small leak in a primary coolant pipeline	5.7×10^{-5}/a	Auxiliary feedwater supply and main steam heat removal, high-pressure injections, other	S_2IJ S_2IC	90% 5% 5%
Loss of power	1.3×10^{-5}/a	Auxiliary feedwater supply and main steam heat removal, delayed feedwater supply and main steam heat removal, long-term feedwater supply and main steam heat removal	T_1IJQ T_1R	67% 33%
Small leak at the pressurizer during power failure	7.0×10^{-6}/a	Auxiliary feedwater supply and main steam heat removal, high-pressure injection, other	$T_1S_2'IJ, T_1S_2'IJCE$ $T_1S_2''IJ, T_1S_2''IJCE$ $T_1S_2'IC, T_1S_2'ICE,$ $T_1S_2''IC, T_1S_2''ICE$	20% 78% 2%

Table 5-6. Contributions of the different failure modes to the frequency of failure to cope with an accident

LOCA, Transients	Frequency of failure to cope with accident	UA**	CMA#	M##	UA & CMA	UA & M	CMA & M	UA & CMA & M
Large leak in a primary coolant pipeline	$5 \times 10^{-7}/a$	73%	15%	12%	--	--	--	--
Moderate leak in a primary coolant pipeline	$2 \times 10^{-6}/a$	62%	11%	27%	--	--	--	--
Small leak in a primary coolant pipeline	$5.7 \times 10^{-5}/a$	13%	1%	85%	--	1%	--	--
Loss of power	$1.3 \times 10^{-5}/a$	26%	--	--	29%	27%	18%	--
Small leak at the pressurizer during power failure	$7 \times 10^{-6}/a$	33%*	--	--	26%*	4%*	--	37%*
Total	$8 \times 10^{-5}/a$	18%	1%	63%	7%	5%	3%	3%

* The percentages for failure combinations resulting in failure to cope with a small leak at pressurizer are given under the assumption that a power failure has occurred.

**UA: Independent hardware failures, including maintenance interval.

#CMA: Common mode failure of hardware.

##M: Human error, including common mode failure due to incorrect calibration of instrumentation.

in a power failure. In order for a LOCA to occur by means of a pressurizer relief valve, an independent failure of hardware is always needed (e.g., failure of the relief valve). For small leaks at the pressurizer resulting from a power failure, both common cause failures of hardware as well as human error will lead to an uncontrolled accident only in connection with independent failures.

If we consider the total frequency of core melt, then about 2/3 of this amount is due to human error. But only 1/20 of this amount, i.e., about 3% of the result, is comprised of common cause failures due to human error (incorrect calibration of measurement channels). If we add all contributions in which common cause failures of hardware or human error play a part, then we obtain about 15% of the total determined core melt frequency. Reasons for this relatively low percentage are the dominant contribution of manual efforts to cope with small leaks in a primary coolant pipeline, as well as the potential for using the emergency system for feedwater injection during power failures when common cause failures of the emergency diesel generators occur.

5.3 ACCIDENTS DUE TO EXTERNAL EVENTS

5.3.1 Overview

In Chapter 3 we presented the important external events that are taken as a basis for the design. Within the framework of the present risk study, we had to examine whether a contribution to total risk is to be expected, with regard to the frequency of external effects such as:

- failure of safety provisions
- weak points in the design
- potential exceeding of design load assumptions.

In addition, we had to check whether risk contributions could arise from other possible external events.

The procedure in analyzing external events differs from the procedure used with plant-internal accidents because we generally do not rely on extensive quantitative accident sequence analyses. For external events of low frequency (e.g., aircraft crash, explosion), the probability of such an accident terminating in a core melt was fixed by an upper limiting estimation. If the studies showed identical or comparable accident sequences as for plant-internal accidents (e.g., power failure), then further consideration could be terminated if the frequency of the initiating, plant-internal accident sequence was significantly greater than the frequency of the external events. The accident sequences due to external events of flood and lightning were treated qualitatively.

The important results of the studies on external events are presented in Sections 5.3.2 to 5.3.7.

The effects of harmful substances (aggressive, oxygen displacing, toxic substances, etc.) resulting from accidents in the environment or from fires were not studied separately. A detailed research project by the Federal Ministry of Research and Technology will deal with this topic and will also study the relevance to risk of such events (5). Final results from this research project will be available for phase B of the risk study. On the basis of previous work, significant risk contributions are not to be expected.

The analyses on external events necessarily relate to the reference plant or its site. Transfer of results to other plants or sites is not possible without modification. For instance, different external events like earthquake, flood, and explosion depend on the site with regard to both frequency and magnitude. In addition, in newer plants the safety provisions against external events (especially earthquake, aircraft crash, lightning strike) have been expanded significantly so that lower risk contributions are expected for them.

5.3.2 Earthquake

5.3.2.1 Introduction. In WASH-1400, reactor accidents due to earthquake were assigned a small contribution to the total risk: the frequency of a core melt accident due to earthquake lies between 10^{-6} and 10^{-8} per reactor-year. These results cannot be adopted without modification. Although we can ask whether detailed studies are indeed necessary with regard to the relatively low earthquake frequency in the FRG, for the sake of completion, a detailed analysis was performed.

5.3.2.2 Frequency and Intensity of Earthquakes. Seismic activities have been recorded by instruments for about 80 years. For a period of about 1000 years, descriptions of earthquake damage and effects have been available. In accordance with their damage and effects, earthquakes are categorized on intensity scales (Seismic categories). There is a multitude of observations of earthquakes. We will not discuss at this point the difficulties in estimating high-intensity earthquake frequencies by statistical methods from these observations; an appropriate discussion is found in the appendix.

The magnitude is the best statistical measure for an earthquake (this is the energy released by the earthquake in the form of waves) or the intensity (a measure for the effects of the earthquake observed on people, structures, and terrain). Through statistical evaluation of observations, earthquake intensities or magnitudes can be assigned average frequencies per year per region. If we plot the frequencies for intensities or magnitudes on a logarithmic scale, then we have an approximately linear relationship (Figure 5-9) (6).

The location of the reference plant lies at the northern end of the upper Rhine trough, a seismically active region where earthquakes are observed again and again. Therefore, extensive information is available about occurring earthquake intensities over a long period. For earthquakes that are more frequent than 10^{-3} per year, the relationship between frequency and intensity can be estimated. Basically, earthquakes of intensities greater than those previously observed are possible. The frequencies of such earthquakes are obtained by extrapolation of existing data (extrapolation to frequencies less than 10^{-3} per year).

Note here that for physical reasons earthquake magnitude cannot increase indefinitely since maximum energy release is limited. Extrapolation takes place by means of Gumbel extreme statistics; this provides the needed, extreme magnitudes or intensities for a prolonged time period as earthquake intensity assumptions, e.g., for the service life of a nuclear power plant of about 40 years. The intensities or magnitudes cannot be used directly to calculate earthquake loads. Seismic engineering parameters are needed to determine loads on structures and system components. Between macroseismic intensity and seismic engineering parameters like, for instance, the maximum ground acceleration, a functional relationship exists. Figure 5-10 gives probabilities of exceeding the maximum ground acceleration a_0 per year for a location in the upper Rhine trough (7). The value of 1.5 m/s^2 for a_0 corresponds to the acceleration value taken as a basis for the design of the reference plant (safe shutdown earthquake).

5.3.2.3 Procedure to Determine Accidentc Consequences To study possible accident sequences that could result from an earthquake, an extensive analysis of event sequences is necessary. The attendant event tree is similar to the event tree for power failure. However, after initial agreement, its further profile exhibits additional branchings. These depend primarily on whether an earthquake-induced

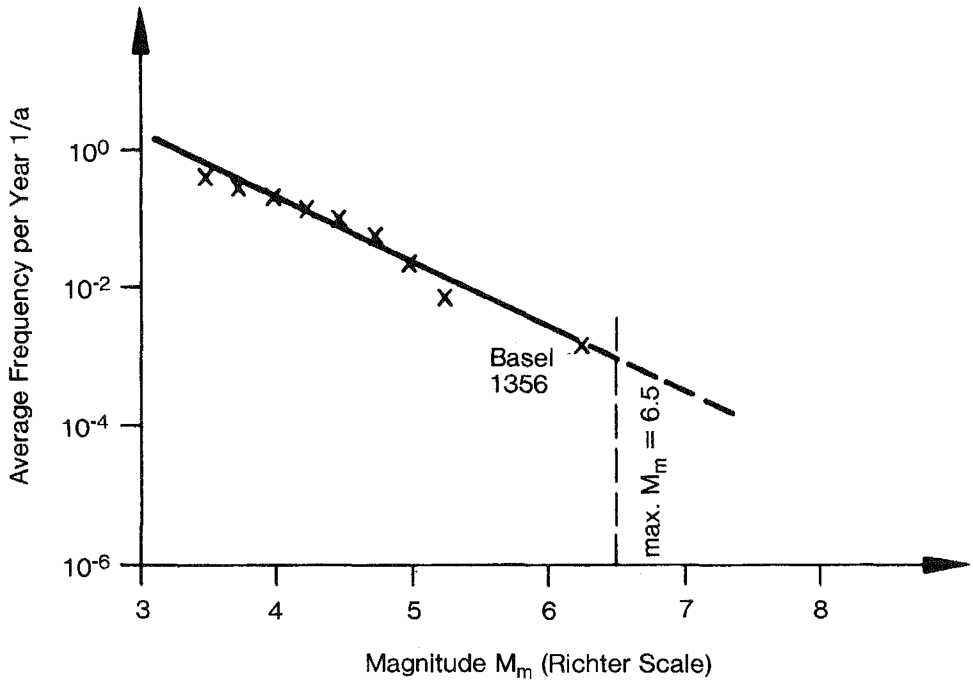


Figure 5-9. Average frequency N per year of earthquakes of a certain macro-seismic magnitude M_m on the Richter Scale for the upper Rhine Valley

LOCA takes place and whether the containment is isolated. Calculation of branching probabilities in the event tree can be implemented primarily by means of the same fault tree analyses that were also used in the corresponding plant-internal event sequence analyses. However, in addition to the reliability data used there, earthquake-induced failure probabilities must also be considered. These will include additional failure modes attributable to earthquake. For instance, failure of a structure can lead to failure of a number of components. But components may not fail directly as a result of earthquakes. Of component failures we distinguish between failure of structure (fracture, deformation) and failure of functionality (incorrect control, interruption, disruption in kinematics).

Within the framework of the present study, it was not possible to examine all reactor structures and components in detail. But going beyond WASH-1400, where only an overall estimation of system failure probability was made, in this study a detailed analysis of earthquake-induced failures was made, for a number of structures. Selection of structures proceeded on the basis of their sensitivity to earthquake effects and their importance to overall safety.

The results are shown in the next chapter. Individual components will be studied in phase B.

5.3.2.4 Failure of Assemblies Due to Earthquake

Calculation of Failure Probability

The calculation of probabilities for failure of structures and components is divided into two sections:

- From known or assumed probability distributions for strengths, sizes, and loads, the distribution of the safety interval Z between load capacity and load was calculated with regard to the safety requirements inherent in the design. This was performed for various selected earthquake regions having attendant maximum acceleration a_0. The conditional probability $p_f^*(a_0)$ for failure of the component for a given earthquake was found (Figure 5-11).
- From the conditional probabilities $p_f^*(a_0)$ and the frequency distribution of maximum acceleration a_0 for the site of the reference plant (Figure 5-10), the earthquake-induced failure probability p_f of the component was determined.

Failure Probabilities of Structures

Failure of the following structures caused by earthquake was studied and probabilities evaluated (for a better understanding, see Figure 3-20 in Chapter 3):

(a) Global failure of the interior cylinder in the reactor building in the lower region: a failure of the interior cylinder due to earthquake is impossible according to this study, even for exceptional earthquake regions.

(b) Fracture failure of the internal cylinder beneath the round overhead crane in the region of the fuel storage pool: the circular beam above the fuel storage pool is designed against horizontal shock from the crane. Its failure frequency thus remains at a value of about $p_f = 4 \times 10^{-5}$ per year.

(c) Fracture failure of the separation between fuel storage pool and storage ground for core internals: the pool wall is designed for plate stresses in accordance with the elasticity theory. If we consider probable failure mechanisms, a failure is practically impossible.

(d) Support of the RPV on the support shield: the structure stresses depend primarily on whether components of the reactor coolant system simultaneously fail in an earthquake. Initial studies indicate that the failure probability of support and mount for a superimposition of stresses due to earthquake and leak (reaction forces) would be greatly increased. Now the loads on the reactor coolant system due to the safety earthquake are clearly lower than the load assumptions on which its design was based. A direct failure of the reactor coolant system, and thus a superimposition of the loads due to earthquake and leak, is therefore improbable. Under this assumption, the failure probability of the foundation is hardly affected by earthquake.

(e) Upper support of the steam generator: the information given under (d) above for a fracture failure also applies to the upper support.

(f) Wall of valve room in the area of the main steam pipe restraints: the wall is designed to withstand reaction forces resulting from destruction of the main steam pipelines in the region of the turbine hall--for instance, due to an earthquake--on the basis of a theoretical elastic plate calculation. Because of the very conservative definition of reaction forces for main steam pipeline fractures, the failure frequency of the chamber wall remains below about $p_f = 4 \times 10^{-6}$ per year even for fracture of several pipelines.

(g) Crossbeams in the turbine hall: the steel-concrete frames used for lateral reinforcement of the turbine hall behave excellently for earthquake effects, i.e., they withstand overloads by local plastic deformation. This means that, on the average, they can withstand relatively high maximum accelerations (up to

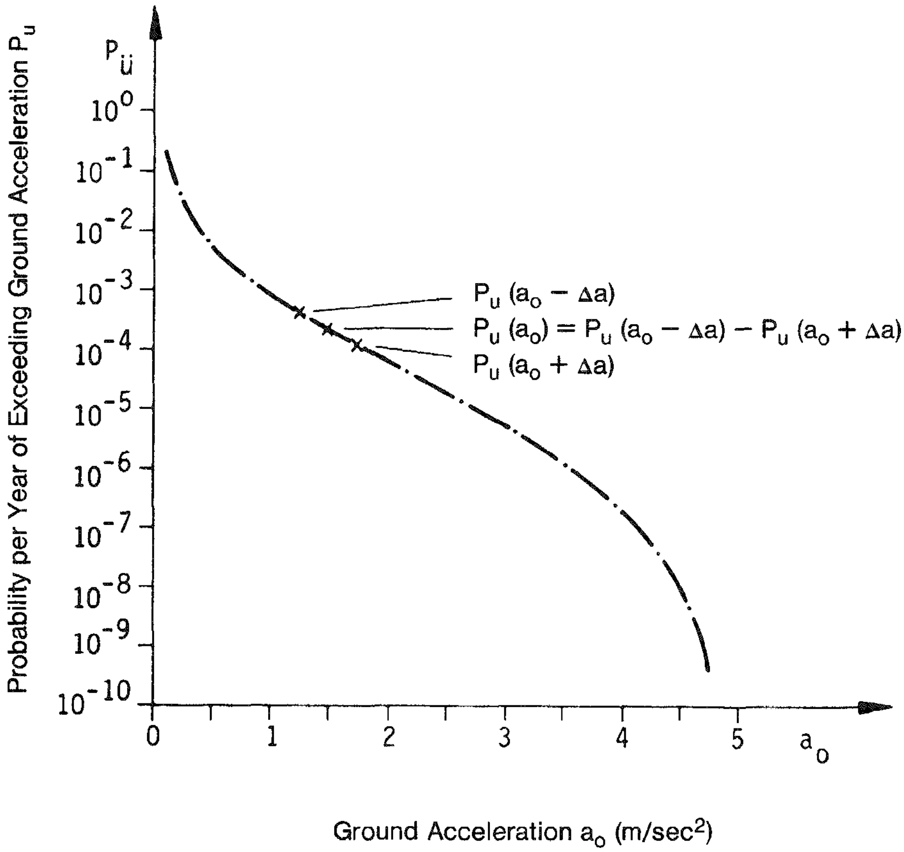


Figure 5-10. Annual probability P_u of exceeding the maximum ground acceleration a_o at the Biblis site

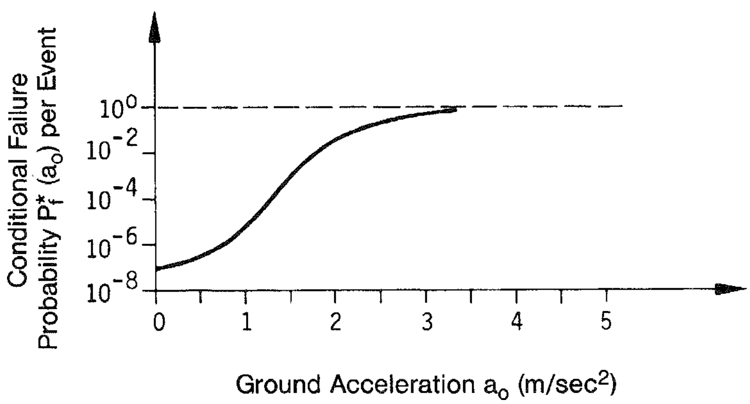


Figure 5-11. Example of the conditional failure probability P_f^* (a_o) of a component in a given earthquake with ground acceleration a_o

a safe shutdown earthquake). The failure frequency of the frames due to earthquake is thus less than about $p_f = 10^{-3}$ per year.

5.3.2.5 Preliminary Evaluation. In the licensing procedure, the failure of components of the reactor coolant and secondary coolant system (within the containment) during a safe shutdown earthquake (SSE) is not assumed, since the dynamic stresses in the safety earthquake remain significantly below the design loads of these system components. This means that an overload of stresses due to earthquake and leaks is not assumed. In phase A of the study, in the investigation of potential structural failure, we also proceeded from this assumption. The validity of the assumption--even for greater maximum accelerations than occur in the SSE--must be verified in phase B.

The structural evaluation shows that the following structures should be studied more carefully because of the determined failure frequencies:

- internal cylinder under the round overhead crane in the region of the fuel storage pool
- wall of the valve room in the area of the main steam pipe restraints
- crossbeams (frames) in the engine room.

A failure of these structures does not lead directly to core melt. Rather, additional events must occur before a core melt can take place. The frequency for a core melt accident, caused by earthquake, is thus significantly below the failure probabilities for the particular structures. The attendant event sequences are described in the appendix. The event sequence upon failure of the crossbeams in the turbine hall will be discussed briefly as an example.

Since the turbine hall is designed only conditionally against earthquakes, a greater failure frequency results for the crossbeams in comparison to the other structures under discussion. In the study it was assumed that a failure of crossbeams leads to destruction of walls and ceiling of the turbine hall. Thus, the auxiliary power supply fails. The further event sequence now resembles a power failure. Since the frequency of such earthquake-induced event sequences is 10^{-3}/year--thus about two orders of magnitude smaller than the frequency of power failure--this event sequence makes an insignificant contribution to risk. Section 5.3.6 will discuss other subsequent events that could arise from the collapse of the turbine hall.

To determine the probabilities for an earthquake-induced failure of components of safety systems, or of the safety systems themselves, a closer examination of the problem will be undertaken in phase B of the study. This is a highly complicated procedure that goes beyond the bounds of phase A. If we take results from (8) as a basis for our initial estimation of earthquake-induced failure probability of safety systems, the failure of these systems does not result in a dominant contribution to risk. However, this presumes that the frequency of a LOCA as a result of earthquake is small because of the reasons given above.

5.3.3 Flood

Nuclear power plants are designed to withstand floods in accordance with the particular siting conditions. The individual provisions are established in the framework of the licensing procedure. As in WASH-1400, phase A of the German study assumes that flood does not provide any dominant contribution to core melt frequency. Studies to verify this statement are planned for phase B of the study.

5.3.4 Rough Weather

5.3.4.1 Storms. The studies have shown that in the FRG, observed and anticipated wind conditions do not indicate any danger to the plant on the basis of the plant's ability to withstand wind loads and other external effects. As a maximum possible event, a power failure is possible, for instance, through a failure of turbine and auxiliary power transformers or by power mains failure outside of the buildings. However, the value for frequency of a power failure caused by "storm" is less than the value found in the plant system studies.

5.3.4.2 Lightning Strikes. The expected frequencies for lightning strikes at the reference plant were set at 3×10^{-1}/year for the reactor building and at 9×10^{-1}/year for the exhaust stack. By means of extensive protective measures, which are divided into external and internal lightning protection measures, effects on buildings and systems are counteracted. The effects on electrical and electronic systems due to induced voltages and the electric decoupling of redundant systems are important considerations here.

The application of probabilistic methods to a quantitative determination of potential impairment and failure of electrical components is basically possible with regard to various parameters; however, the data needed for this are not available. A quantitative evaluation of risk is, therefore, not possible. Because of rough estimations, it seems possible that an event chain that proceeds from a lightning strike and terminates in a core melt is of little significance when considering existing protective measures and safety systems; it would therefore be comparatively unimportant to the core melt frequency.

5.3.5 Aircraft Crashes

The airspace over the FRG is characterized by a close network of civilian transportation routes with high-density use. The resulting global flight density increases if we include military aircraft as well.

Within the framework of the study, the situation at the site of the referenced plant was studied first, and the frequency for the crash of civilian and military aircraft on the plant was determined. Relative to a safety-relevant nuclear power plant surface area of 10,000 m^2, we have:

- civilian aircraft on flight paths (with a takeoff weight greater than 200 kN: 2×10^{-11}/year
- civilian aircraft not on flight paths (with a takeoff weight less than 200 kN): 9×10^{-7}/year
- fast military aircraft: 1×10^{-6}/year.

These values were determined by the specific crash frequency for cruising or training flights. The influence of aircraft taking off and landing is negligible due to the distance to the closest airport.

If we consider the potential loads in addition to the crash frequencies, then crash of a fast military aircraft represents the dominant event for a determination of a potential risk.

Accordingly, with regard to hypothetical loads and existing protective measures, subsequent events down to a "core melt" were studied for this case. The event tree for "aircraft crash" is shown in Figure 5-12.

If we ignore subsequent phenomena like fuel fires and the effect of debris (wreckage, broken pieces of buildings), which were, however, included in the analysis, then aircraft crash represents a local effect. The event tree can thus be applied, in a somewhat modified manner, to all buildings at the site. Because of the low frequency of the initiating event, however, no detailed analysis was performed; rather, an upper limit for the frequency of one subsequent event "core melt" was estimated for the particular local effects (on a building for instance).

The list in Table 5-7 shows the determined frequency values whereby we distinguish between the "full power" operating mode (a) and "shutdown plant" (b).

In summary we can say that a core melt accident resulting from an aircraft crash onto a power plant is anticipated with a frequency of less than 2 X 10^{-7} per year. Compared with other accidents, the aircraft crash therefore makes no notable contribution to risk because of appropriate plant design.

5.3.6 Explosion Shock Waves

Experience shows that explosions should be anticipated in industrial plants and on transport routes (highway, rail, river, pipeline). Explosions can be caused by explosives per se or by other explosive substances when accidents occur in storage, transport, or handling.

The design of the plant is based on load assumptions that will accommodate shock waves from a deflagration of saturated hydrocarbons. It is assumed that this event (owing to the potential drift of an explosive gas mixture) can occur right next to the outer wall of a building. This points up the following questions:

(a) What is the frequency of loads taken as a basis for the design?

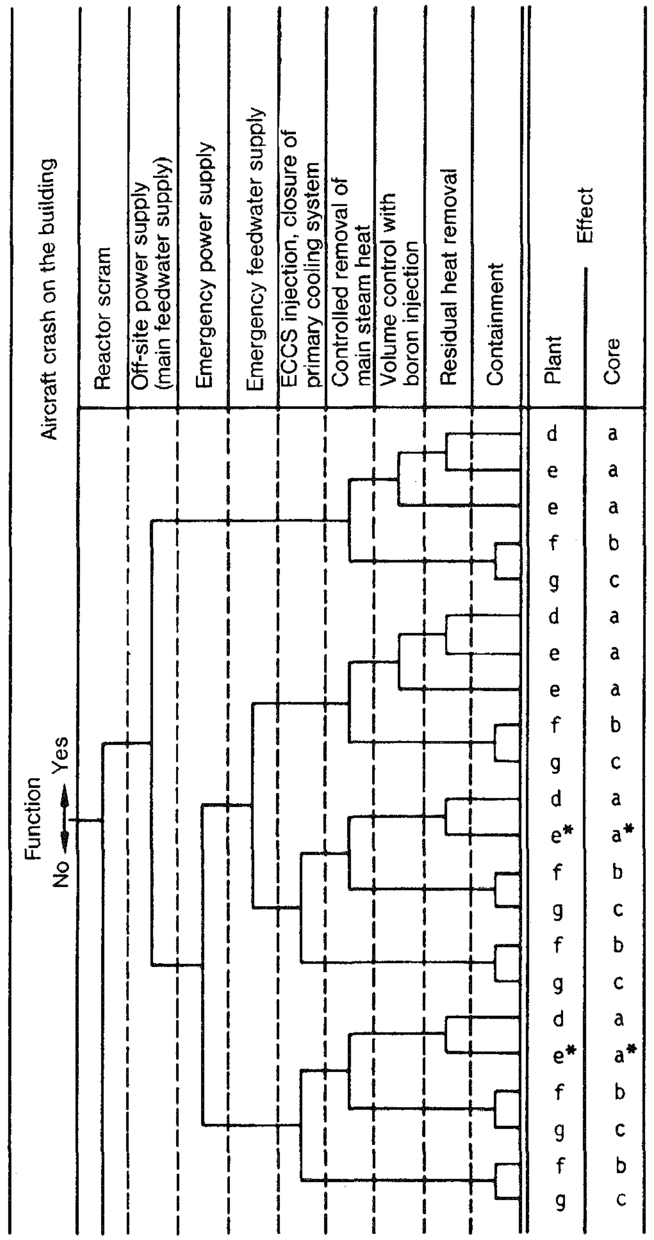
(b) How probable are possible subsequent events that can result in core melt if we assume that design loads have been exceeded?

(c) What is the frequency that the design load assumptions can be exceeded?

For the frequency of a shock wave that equals the design loads of the system, a value of 10^{-5}/year to 5 X 10^{-7}/year was estimated.

Potential subsequent events that can lead to core melt if we assume exposure to design loads were studied and their probability evaluated. It turns out that the probability of a core melt due to exposure to design loads is relatively small. Accordingly, this event sequence does not make any relevant contribution to risk.

The boundary conditions (point of deflagration, gas cloud model) used in determining the load function were conservatively set so that greater released quantities of gas upon deflagration basically do not lead to greater shock waves. Shock waves moving from a detonation (e.g., unsaturated hydrocarbons, explosives, dammed gas clouds) can result in considerably higher loads. As a result of the pre-



*Under consideration of special measures

With i = 1. Switchgear building
2. Reactor auxiliary building
3. Valve room
4. Valve equipment room
5. Region of exposed main steam pipeline
6. Covered walkway
7. Secondary cooling water pump chambers
8. Turbine hall and main power connection

Consequences: a) No core melt
b) Core melt with isolated containment
c) Core melt with non-isolated containment
d) System becomes cold sub-critical
e) System remains hot sub-critical
f) Over-pressure failure of primary coolant vessel
g) Damage to the primary coolant vessel through direct aircraft effects must be assumed

Figure 5-12. Event tree for "aircraft crash"

Table 5-7. Frequencies of core melt due to externally-induced effects on the buildings or plant parts

Effects on buildings or plant parts	Annual frequency of core melt under consideration of damage frequency on the particular building on plant part
Reactor building	$< 6 \times 10^{-8}$*
Switchgear building	$< 3 \times 10^{-8}$
Reactor auxiliary building	$\ll 10^{-8}$
Valve room	$< 2 \times 10^{-8}$
a) Region of exposed main steam pipeline	$\ll 10^{-8}$
Covered hallway	$\ll 10^{-8}$
Secondary cooling water pump chambers	$\ll 10^{-8}$
Turbine hall and main power connection	$< 1 \times 10^{-8}$
Reactor building	$< 2 \times 10^{-8}$*
b) Plant parts containing components of the RHR system or attendant power supply equipment	$< 6 \times 10^{-8}$

*Values assume that the containment is not isolated due to the effects on the reactor building.

vailing distance between sensitive areas of the plant and potential locations for accidents along this particular section of the River Rhine, protection is provided even when the detonation occurs at the point of an accident. Scenarios where explosive gas clouds drift to the power plant or a deflagration near the plant caused by unfavorable boundary conditions is converted into a detonation, have such low probabilities that there is no relevant risk contribution from such event sequences.

These results reflect the fact that nuclear power plants in the FRG are designed against explosion pressure waves.

5.3.7 Effects on the Nuclear Power Plant Area or on Plant Parts Relevant to Safety Due to Failure of Secondary Components

A rupture failure of components in the secondary sphere (turbine, generator, pressure vessels in the turbine hall) can result in transient accidents. If we ignore the system-specific effects of transients examined in Section 5.2.2 but study the mechanical effects on other systems or system components, then our results are comparable to the effects of explosion shock waves and aircraft crashes. Therefore it seemed useful to assign these under the heading "external effects."

5.3.7.1 Turbine Explosion. The event sequence is divided into:

1. rupture of turbine moving parts and destruction of the outer turbine housing
2. flight of fragments and impact on important plants relevant to safety
3. destruction of relevant plant parts and components by these fragments.

If W_1 is the frequency for a turbine explosion, then the frequency W_{ges} pertains to the probability of the destruction of important plant parts and components, obtained from the product of the frequency W_1 with the individual probabilities W_2 and W_3. The effects of fragments were studied separately for the reactor building and the switchgear building.

In the switchgear building, a separation into two subregions was necessary because one subregion (II) similar to the reactor building can only be affected by indirect fragment flight paths because of the building layout.

Proceeding from the event sequences that could develop from the potential effects of fragments, the frequency of a core melt accident due to turbine explosion was determined. The important results are summarized in Table 5-8.

Table 5-8. Results of studies on turbine deblading accident

Effect on	W_1	W_2	W_3	W_{ges}	$W_{Core\ melt}$
Reactor building	--	1.8×10^{-3}	3×10^{-1}	$<1.0 \times 10^{-8}$/a	$<<1 \times 10^{-8}$/a
Switchgear building, subarea I	10^{-5}/a*	5.0×10^{-1}	2×10^{-1}	1.0×10^{-6}/a	$<1 \times 10^{-8}$/a
Switchgear building, subarea II	--	6.5×10^{-2}	1	6.5×10^{-7}/a	$<5 \times 10^{-8}$/a

*Considering contributions due to earthquakes, the frequency of a turbine explosion may be greater. It is estimated that no significant contribution to total risk would result in this case, however.

5.3.7.2 Failure of Pressure Vessels in the Turbine Hall. The components under discussion have a great energy potential because of their large volume and the high pressures and temperatures of their steam/water content. In the appendix, the different failure potentials and resulting stresses are discussed in detail. The frequency for failure of these components is determined by flaws in materials and processing, as well as by effects like earthquakes and turbine explosion. Owing to the arrangement and location of components in the turbine hall and their distance to the reactor building, hazards to the reactor building are unlikely. Potential consequences for other areas of the plant that could arise from different loads (shock wave, fragments) also do not result in any notable contribution to risk.

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Section 6

RELEASE OF FISSION PRODUCTS

6.1 BACKGROUND INFORMATION

To guard against accidents, nuclear power plants are equipped with extensive safety features. In the design and licensing of these safety features, the principle is followed of early recognition of accidents and either preventing or--if this is not possible--mitigating their consequences so that no significant amount of radioactivity is released to the environment around the plant. Because of the extensive safety features, it is generally assumed that accidents that lead to a melt of the reactor core can be excluded as far as humanly possible. Nuclear power plants, therefore, usually do not have a special design to cope with core melt accidents.

Accidents with which the safety systems can cope contribute little to risk since they do not damage the environment of the plant. Therefore, a contribution to risk is basically anticipated only for event sequences that are beyond the capability of the safety features and thus release radioactive fission products from the plant. Since the majority of radioactive fission products are bound in the crystal lattice of the fuel, a considerable release of fission products can occur only when the core melts and the remaining fission product barriers fail. Consequently, an important task of the risk study is to discuss these extreme cases and to evaluate their frequency and consequences.

No conclusions regarding core melt accidents can be drawn on the basis of available operating experiences. Therefore, in the treatment of core melt accidents or, in general, in risk determinations, one must rely heavily on theoretical studies.

The studies of which event sequences can lead to core melt upon what failures of combinations of safety systems and the results of the appropriate probability calculations are explained in Chapter 5. Proceeding from this discussion, the present chapter is concerned with a model description of the further event sequence, beginning with a core melt, to potential release of fission products to the environment. In particular, the processes during a core melt, the release of fission products from the reactor core, the loading on the containment, potential failure modes of the containment, and resulting release of fission products from the plant are studied.

Section 6.2 contains the results of model studies on core melt processes. All phases of the core melt sequence are discussed. These are primarily:

- melt of the reactor core
- melting through the reactor pressure vessel (RPV)
- processes involved in penetration of the melt into the foundation of the building.

In particular, the energy and mass carry-over connected with a core melt accident from the reactor coolant system into the containment are important for the loading on the containment and thus also for its potential failure.

The containment represents the last fission product barrier in a core melt accident. Its leak tightness in this case is decisive for determining whether and to what extent fission products can escape from the plant. The various potential failures of the containment will therefore be discussed in Section 6.3. A failure or leak in the containment can have two principally different causes. First, as a possible consequence of a core melt accident, loads may occur that exceed the failure limits of the containment and thus result in structural damage. Second, failure to isolate penetrations through the containment at the beginning of an accident can cause loss-of-leak tightness of the containment. Initially, we compile and discuss all potential sequences that could, by one method or another, result in loss-of-containment integrity. Next, the failure modes of the containment that are pertinent for the reference plant and that must be pursued are delineated.

Section 6.4 is concerned with the problem of steam explosion. In WASH-1400, it was pessimistically assumed that for a core melt accident a certain probability exists of a steam explosion to occur in the RPV destroying the RPV and containment, as well as releasing large quantities of fission products to the environment. In the meantime, a series of more intensive studies on this problem has begun indicating that accident sequences proceeding from a steam explosion through failure of the RPV to a destruction of the containment will not occur. A final evaluation of the steam explosion is not yet possible because the continuing research projects have not yet been concluded. For this reason, the occurrence of a steam explosion causing failure of the RPV with subsequent destruction of the containment is evaluated the same as in WASH-1400.

Section 6.5 concerns model investigations of the release of radioactive fission products. Initially, the fission product inventory in the plant is determined. This is followed for different accident sequences by determining fission product release from the core fuel (magnitude and time history). Transport and retention of fission products in the containment are studied with regard to their thermodynamic conditions. Finally, for the different accident sequences and attendant failure modes of the containment, fission product release to the environment will also be determined.

The studies in Section 6.5 show that different accident sequences sometimes result in similar releases. The releases can therefore be categorized into a series of representative releases, called release categories. Thus, the complexity of calculating accident sequences can be reduced considerably. Section 6.6 explains the considerations that play a role in forming the release categories and the important properties for the individual release categories.

6.2 STUDY OF THE CORE MELT SEQUENCE

6.2.1 Introduction

Radioactive fission products of a nuclear power plant are contained by several structures located one behind the other. These are called fission product barriers. They are (see Section 3.3.1):

- the crystal lattice of the fuel itself
- the fuel cladding
- the RPV together with the reactor coolant system
- the containment.

A large release of radioactive fission products from the nuclear power plant can occur only when the fission products bound in the fuel partially or completely escape from the fuel and cladding and when, in addition, the other fission product barriers fail. The fission products escape to a notable extent from the fuel and cladding only when the fuel melts.

The level and time history of fission product release from the reactor core depend on the event profile of the core melt.

In addition to release of fission products from the reactor core, during a core melt accident a number of other important physical or chemical processes occur. For instance, if in the course of a core melt accident molten core material comes into contact with coolant, then heat is transferred to the coolant, causing it to vaporize. Depending on prevailing conditions, there may be merely boiling or in an extreme case, a spontaneous vaporization of the coolant resulting in a steam explosion. Furthermore, under unfavorable conditions, it must be anticipated that as the result of a core melt accident, the molten fuel will melt through the RPV and will come into contact with the concrete structures. These structures, especially the building foundation, are destroyed by the melting process. Furthermore, upon penetration of the melt into the concrete, water vapor and even hydrogen (generated by chemical processes) are released to the containment. All these processes cause loads on the containment.

From the above reasons a careful study of core melt accidents is necessary within the framework of a risk study. To date, there has been no core melt accident in any nuclear power plant anywhere. Core melt accidents can be described only by using theoretical models. Recently, some aspects were validated by research work, but at present there are no models for accurately predicting the entire sequence of a core melt accident. As in the American Reactor Safety Study (WASH-1400), core melt accidents are treated here by using pessimistic assumptions. The methods and models developed for WASH-1400 are also used here.

The results of core melt accidents are determined largely by the design features of a particular plant-- here, the Biblis B pressurized water reactor. For instance, the design of the biological shield and the spatial arrangement of the building sump are of decisive importance to processes occurring upon penetration of the core melt into the foundation. The result of core melt studies on the reference plant cannot therefore be applied directly to other nuclear power plants. Especially in plants with different building designs--for instance, the Surry I nuclear power plant examined in WASH-1400--other effects can become more important than they are in our reference plant.

6.2.2 Assumptions and Boundary Conditions of the Core Melt Investigations

If during an accident residual heat removal is entirely or partially interrupted, then a core melt may occur under some circumstances. The processes taking place during a partial or complete termination of residual heat removal depend greatly on the particular case and are in general highly complex. Therefore it is difficult to establish realistic criteria for when an event sequence will result in a core melt and when the core will remain intact. Analogous to the method in WASH-1400, the present study proceeds from pessimistic assumptions that an event sequence terminates in a core melt when the safety systems needed to cope with the accident do not fulfill the minimum requirements established in the licensing procedure. All cases where the safety systems operate after a certain delay or at reduced capacity are considered to be core melt accidents in this model, even if the core in reality may actually stay cool. At present, several research centers are working on loss-of-coolant accidents to determine more accurately the size and scope of impairment of residual heat removal from the core that can be expected before a core melt will occur. Primarily, the temperature behavior of the core is being studied by computer for different quantities of water injected by the emergency coolant systems. Preliminary results indicate that the number of potential core melt accidents is clearly over-estimated in the present risk study.

In Chapter 5, the event sequences that can occur in the plant due to technical systems were studied. That is, for all initiating events we checked to see which safety systems come into play and how an assumed failure of these safety systems will effect the event sequence. Depending on whether the minimum requirements established in the licensing procedure are met or not, a decision is made on which event sequences will cause core melt. In order to have definite criteria for the core melt studies, it is always assumed that a system completely fails if it does not meet the minimum limits established in the licensing procedure. As a failure time point we use that time at which the particular system should come into play. For example, let us assume that after a double-ended break of a primary coolant loop and successful reflooding of the reactor core with sump recirculation, the loops of the emergency and RHR system specified in the licensing procedure are not available because of a component failure. Switching from reflood operation to sump recirculation takes place after about 20 minutes for a double-ended rupture of a primary coolant pipeline. In this case it is assumed in the core melt studies that 20 minutes after the beginning of the accident, injection of emergency cooling water into the reactor pressure vessel is completely terminated.

Analogous to the procedure in WASH-1400, the phenomenological sequences of all potential core melt accidents are not detailed. Only those core melt accidents resulting from a large break of a primary coolant pipeline and failure of the emergency and RHR systems are thoroughly examined. It is assumed that all other core melt sequences will be covered by these cases. In general, this assumption can be justified because the different core melt accidents proceed in a similar manner. Only the time history can differ, since the existing residual heat generation is decisive for the time history of a core melt accident. The existing residual heat generation decreases with time. If the decay heat removal from the core fails shortly after the beginning of the accident, then the decay heat generation is still relatively great. If the residual heat removal from the core fails long after the beginning of the accident, then the prevailing residual heat generation is much less. This means that in an early start of core melt, the entire core melt process proceeds more quickly than in the case of a later onset of core melt. An earlier and faster sequence of the core melt process leads to a corresponding early release of fission products from the core and to greater loads on the containment. Both factors are unfavorable for fission product release to the environment.

If we assume, after a large leak in a primary coolant pipeline, the failure of the emergency and RHR systems, then a core melt occurs comparatively more quickly than in other core melt accidents such as those due to small leaks in primary coolant pipelines or from transients. If no injection from the emergency and RHR system occurs, for instance, the reactor coolant system and RPV quickly drain because of the large break cross-sectional area. The exposed reactor core very soon begins to melt. Therefore, other core melt sequences can be covered by core melt sequences arising from large leaks in primary coolant pipelines. For transients, accident sequences are possible which lead to core melt under full pressure in the reactor coolant system. A more accurate study of such accident sequences is not yet available. As in WASH-1400, it is assumed that these core melt accidents can also be covered by core melt sequences from large leaks.

6.2.3 Results of the Core Melt Investigations

All event sequences that could result from a large break in a primary coolant line, if we also assume failure of safety systems, are compiled in Chapter 5 (Figure 5-2). From these event sequences we selected two representative core melt accidents, which shall be studied more accurately and described on the basis of existing models.

The accident sequence given below as "core melt accident 1" is based on the following assumptions:

- large leak in a primary coolant line
- performance of the accumulators and low-pressure injections from the storage tanks in accordance with the minimum requirements of the licensing procedure
- failure of emergency and RHR systems when switching to sump recirculation after about 20 minutes.

This core melt accident covers all accident sequences caused by failure of the long-term residual heat removal. This is also true for accident sequences due to medium or small leaks in a primary coolant line, since switching to sump recirculation occurs later than for the sequence of large leak in a primary coolant pipeline.

The accident sequence given below as "core melt accident 2" is based on the following assumptions:

- large leak in a primary coolant line
- performance of the passive safety features (accumulators) in accordance with the minimum requirements of the licensing procedure
- complete failure of all active safety systems (low-pressure injection systems).

This core melt accident pessimistically describes all accident sequences where failure of active safety systems is assumed per se (for accident sequences arising from medium and small leaks in the primary coolant pipeline, this case would mean failure of the high-pressure and low-pressure injection systems). The core is reflooded by the accumulators, and the pressure vessel is filled up to the lower nozzle edge. A little while later the accumulator injection stops. Because of the cessation of cooling water injection, vaporization of water in the RPV and subsequent melt of the core begins. For a double-end rupture of a primary coolant line a pessimistic determination estimates about 100 seconds until the beginning of vaporization. For smaller fracture sizes and for transients, this value represents a conservative estimation. Core melt accident 2 results in a core melt sooner than core melt accident 1.

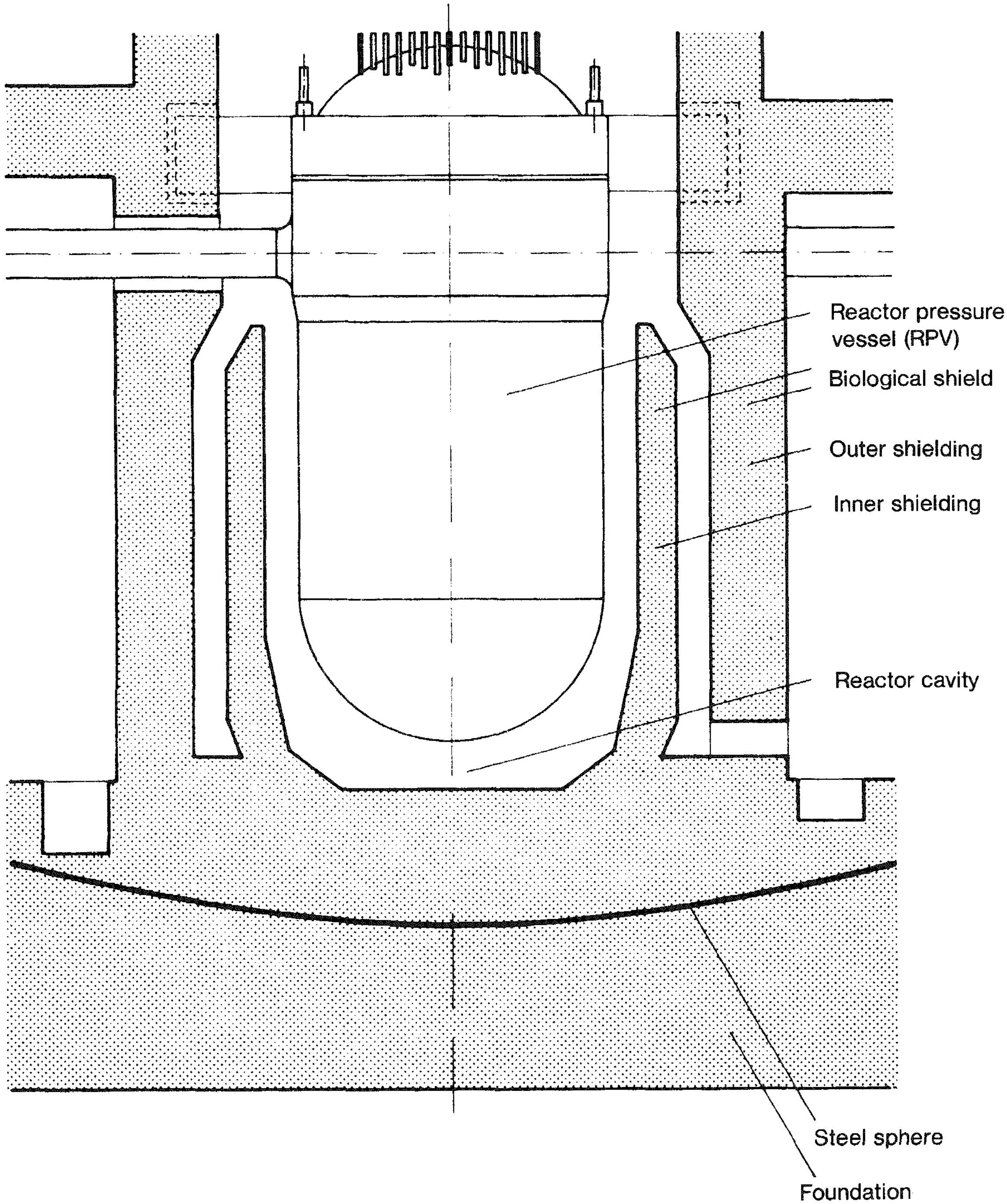


Figure 6-1. Reactor pressure vessel (RPV) and biological shield

The detailed results of model studies or estimations on core melt accidents are documented, together with relevant boundary conditions and parameters in the appendix. Below we explain the important study results using the example of the first of the two core melt accidents. The spatial arrangement of the RPV and the building structures important to a core melt sequence are seen in Figure 6-1. In core melt accident 1, it is assumed that upon switching to sump recirculation about 20 minutes after the beginning of the accident, all four subsystems of the emergency and RHR system fail entirely, and thus injection of emergency cooling water to the RPV stops. At this time the RPV is still filled with water at least up to the lower edge of the broken primary coolant pipeline. The model calculations then provide the following conceptions for the further progress of the accident.

The residual heat generation in the core first heats and then vaporizes the water in the RPV which thus reduces the water level. When the water level drops to the core zone, the upper zones of the core are exposed and begin to overheat. Above a temperature of about 950°C, an exothermic chemical reaction takes place between steam and the zirconium in the cladding. In the course of this zirconium-water reaction, large quantities of heat are generated. The core heating is thus accelerated, finally melting individual core zones. The zirconium-water reaction oxidizes the cladding material, forming hydrogen that is transported into the containment through the rupture.

The onset of core melt alters the original core geometry. When a core is at least partially covered with water, the molten core material will generally not fall directly into the lower plenum of the RPV, but will first solidify on colder core structures (1). The resulting crusts then collect larger quantities of molten material under some circumstances. Only when the core support structure has reached its failure temperature can we expect the core to collapse into the lower plenum of the RPV, which is likely to be still filled with water.

This section does not discuss the potential of a steam explosion. Questions with regard to the problem of steam explosion will be discussed in detail in Section 6.4.

After the molten core material collapses into the lower plenum of the RPV, the core debris and molten core material release their stored and residual heat to the residual water in the lower plenum and vaporize it. Next, the core material again heats up, melts through the reactor pressure vessel, and drops onto the concrete reactor foundation.

The concrete wetted by the melt heats up and then melts, the melt gradually penetrating into the surrounding concrete and liberating the physically and chemically bound water in the concrete. This water can be transported into the containment as superheated steam. Experimental studies on this problem (2) indicate, however, that almost the entire volume of steam oxidizes the metal parts of the melt and, as a consequence, liberates hydrogen. Generation of CO_2 is not expected--in contrast to WASH-1400--because a different type of concrete has been used in the German plant.

The melt process moves laterally toward the sump and vertically into the foundation. As it continues, it is assumed that the melt comes into contact with sump water. Vaporization of the sump water leads to a continuous increase in containment pressure.

What fraction of residual heat generated in the melt will contribute to vaporization of sump water is not presently known. Intensive studies are underway on this question. Vaporization rate for the sump water is therefore determined on the assumption that the entire residual heat is used exclusively for vaporization. This limiting estimation is confirmed by initial results of on-going research projects. As long as the containment remains leaktight, it is anticipated that a considerable fraction of the vaporized sump water will condense on the cooler internals and structures of the containment and flow back into the sump. The sump water vaporization continues. During this period, the melting front penetrates only a little into the foundation.

Only after a failure of the containment can the vaporized sump water escape from the containment so that the melt is no longer cooled and can penetrate further into the foundation.

The timing at which the melt completely penetrates the concrete foundation and moves into the earth has been estimated by using various models. Accordingly, we can cite a value of four to five days.

Core melt accident 2 proceeds in a similar manner to core melt accident 1. The significant results on the time history of the two accidents are compiled in Table 6-1. The times given there were determined partly by the BOIL computer program used in WASH-1400 and partly by means of hand calculations. These dealt primarily with pessimistic assumptions so that the indicated times might be considerably shorter than would be expected in an actual core melt accident.

Table 6-1. Results of model studies on the time behavior of core melt accidents

Process	Elapsed time after initiating event (hours)	
	Case 1	Case 2
Failure of ECSS	0.3	0
Beginning of core melt	1.1	0.6
End of core melt, collapse of core into the lower plenum	1.4	0.9
End of residual water vaporization in the lower plenum	1.9	1.3
Melt-through of RPV, collapse of the melt into the reactor cavity	2.2	1.6
Melt-through of the inner shielding, contact of melt with sump water	4.4	3.7
Overpressure failure of the containment	26	21
Failure of the building foundation	ca. 100	ca. 100

Case 1:
- --Large leak in primary coolant pipeline
- --Functioning of accumulators and of the low-pressure injection from the storage tanks corresonding to the minimum requirements of the licensing procedure
- --Failure of RHR systems upon switchover to sump recirculation after about 20 minutes

Case 2:
- --Large leak in a primary coolant pipeline
- --Functioning of the passive safety features (accumulators) corresponding to the minimum requirements of the licensing procedure
- --Complete failure of all active safety systems (low-pressure injection system)

6.3 STUDIES ON FAILURE OF THE CONTAINMENT

6.3.1 Background Information

The present chapter concerns studies on potential failure of the containment. Analogous to WASH-1400, detailed studies have been performed only for those event sequences that could result from a large rupture in a primary coolant pipeline. It is assumed that these event sequences include all other event sequences with regard to loads on the containment. A discussion of this assumption is found in the appendix. During a malfunction or accident, various processes occur depending on the sequence of events; these processes can lead directly or indirectly to a loading on the containment. If failure limits of the containment are exceeded, then structural damage occurs. As an initial step, all processes affecting the status of the containment are summarized and discussed individually, regardless of their importance. Cases can then be determined in which failure of the containment can be anticipated based on resulting loads on the containment.

During a failure of the containment, fission product release to the environment is affected not only by the timing and mode of failure, but also by thermodynamic conditions in the containment.

6.3.2 Discussion of Containment Failure Modes

A failure or loss-of-containment leaktightness during a malfunction or an accident can have two basically different causes. First, it is possible to render the containment leaky because isolation failures occur upon demand. Second, it is also possible that in the course of a malfunction or an accident the containment may be subjected to loads for which it is not designed. If such stresses exceed failure limits, the containment will be damaged.

We will briefly discuss the first cause. Subsequently, we present a detailed discussion of the second cause.

6.3.2.1 Failure of the Containment Isolating Valves. The containment consists of a gas-tight, welded, spherical steel envelope of 56 m diameter. Three pressure-resistant and gas-tight airlocks lead into the interior of the steel sphere. In addition, a number of pipelines and cable penetrations into the steel sphere are needed primarily to operate the systems located within the containment. Each pipeline penetrating the steel sphere is equipped with at least two isolation valves, located one behind the other. Once an accident begins, the isolating devices on all pipelines not needed to cope with the accident are automatically closed. If both isolating valves of a pipeline do not close for any reason, then in particular pipelines the containment can no longer be tightly closed. A leak corresponding to the size of the pipeline then occurs. Potential leaks of the containment during malfunctions or accidents have been studied in detail by means of systems and reliability analyses, and their probabilities evaluated. It is useful to divide the spectrum of potential leaks of the containment into the following three areas:

- large leak of the containment, represented by a 300-mm diameter leak
- medium leak of the containment, represented by a 80-mm diameter leak
- small leak of the containment, represented by a 25-mm diameter leak.

Additional details on the studies and results achieved are provided in the appendix and will not be explained further at this point.

6.3.2.2 Failure of the Containment as a Result of Exceeding Permissible Loads. During a malfunction or an accident, various physical or chemical processes can occur, depending on the sequence of events, which more or less significantly affect the integrity of the containment.

Compilation of Potential Loads

The model investigations on the event sequences of core melt accidents in Section 6.2.3 provide the following occurrences that can affect the integrity of the containment:

- liberation of steam into the containment
- liberation of water into the containment
- release of fission products and residual heat connected with them into the containment
- mechanical load of the containment resulting from a steam explosion in the RPV (see Section 6.4).

Hydrogen that may be released into the containment can behave in different ways:

- As a gas, it can increase pressure in the containment.

- It can burn, adding heat and pressure to the containment.
- It can explode under certain conditions.

Model Description of Processes in the Containment

The effects outlined above will now be discussed in greater detail with respect to their effects on the containment.

Adding mass and energy from the reactor coolant system and reactor core cause the pressure and temperature in the containment to rise. But this pressure and temperature rise is counteracted by the containment internals made of steel and concrete. At normal power operation, the internals have a temperature of 30 to 40°C. If the temperature in the containment atmosphere rises, then the internals form heat sinks. That is, they absorb a part of the energy added to the containment, and they heat up. In general, the internals are quite important with regard to the pressure and temperature history in the containment during malfunctions or accidents.

The transient pressure and temperature in the containment is calculated by the CONDRU computer program. This program, used in the licensing procedure, has now been expanded by adding models that allows the investigation of the effects of core melt accidents on the containment.

Pressure and Temperature Profile in the Containment

The results of studies on the transient pressure and temperature in the containment are all documented in the appendix. This document also contains the results of parameter studies to delineate the influence of the significant parameters. The following examples help explain the study results.

Figure 6-2 shows the pressure profile in the containment for a successfully controlled large LOCA. Here, the double-ended break of a hot primary coolant pipeline is assumed, since this is the worst case with respect to loads occurring on the containment.

First, the primary coolant empties into the containment with large-scale formation of steam. The pressure and temperature in the containment increase continuously. At the end of the primary coolant outflow (blowdown end), the

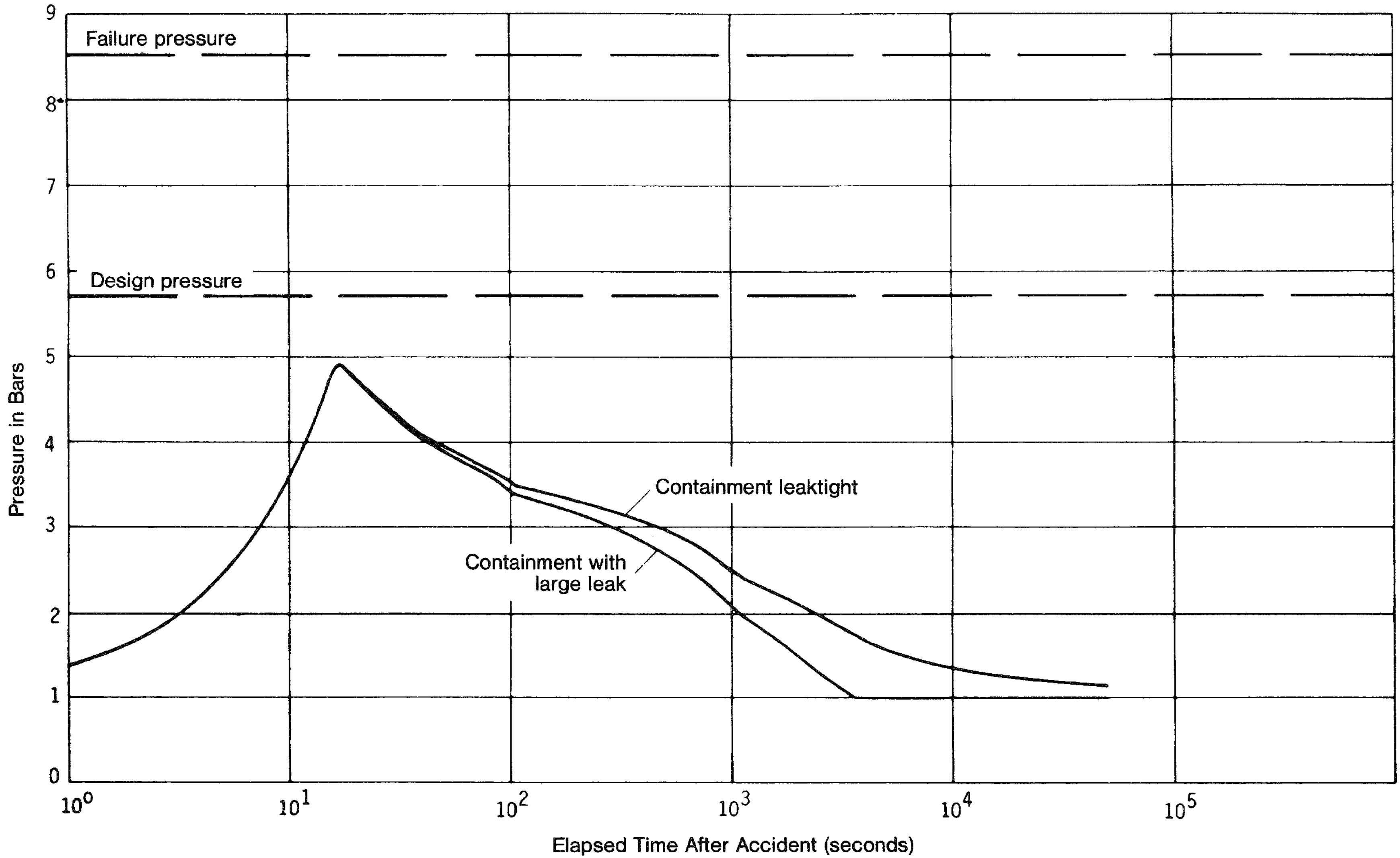


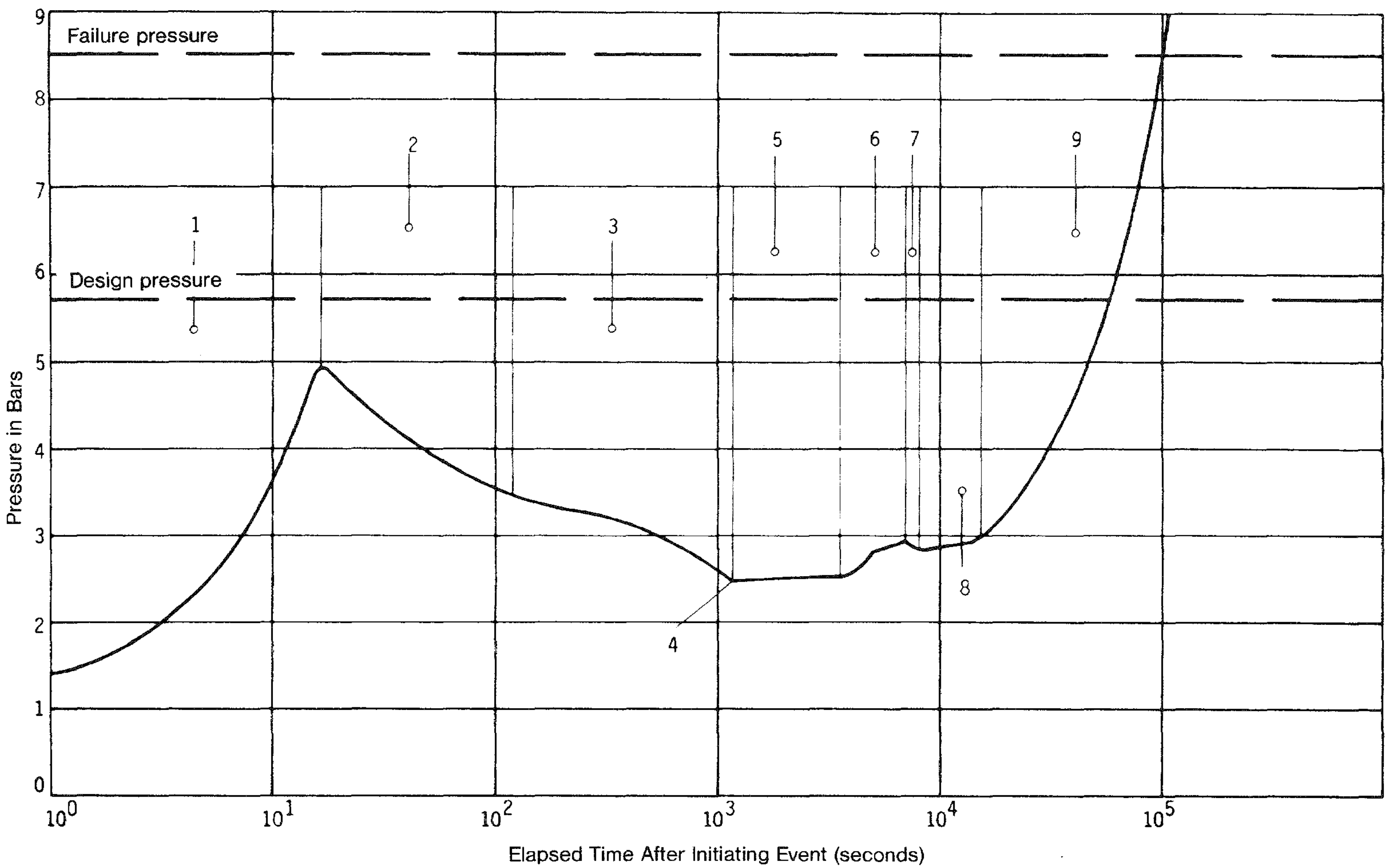
Figure 6-2. Pressure history in the containment for correct plant response to ended break in a primary coolant pipeline

pressure reaches a maximum of about 4.9 bar. Just prior to the blowdown end, the accumulators begin to inject cold emergency coolant into the reactor coolant system. Shortly thereafter, the RHR pumps start up and pump cold water into the reactor coolant system. The RPV is refilled with water and forms additional steam. After the RPV has been entirely refilled, steam formation stops. The injected, cold emergency coolant now absorbs the residual and stored heat of the reactor, causing it to heat up and flow through the rupture into the containment sump. The steam that flows into the containment partially condenses on the internals and then flows down to the sump. The heat removal by the internals is greater than the energy fed to the containment atmosphere in the form of steam at the end of blowdown. Pressure in the containment continues to drop. After about 20 minutes the system is switched over to sump recirculation. The RHR pumps now draw hot water from the containment sump--instead of cold emergency coolant from the flood tanks--and pump it through a residual heat exchanger back to the reactor coolant system. The residual heat exchangers withdraw the residual and stored heat of the reactor, as well as heat absorbed over the long term by the containment, and send them through the nuclear intermediate cooling circuit and the secondary cooling water system to the river. The pressure and temperature in the containment thus drop continuously and finally approach the initial conditions before the beginning of the accident.

During the course of the described accident, a maximum pressure of about 4.9 bar occurs in the containment. This is distinctly below the designed pressure of the containment of 5.7 bar.

The second curve in Figure 6-2 shows the pressure history in the containment for the same accident, but under the assumption that the containment has a large leak (diameter: 300 mm). The pressure history behaves similary to the previous case. As a result of the additional energy and mass flow through the leak, the pressure drops in the containment sooner. After about an hour, the containment pressure reaches atmospheric pressure.

Below, the pressure history in the containment will be discussed for core melt accident 1, which was presented in Section 6.2.3. Until switching to sump recirculation operation, this accident proceeds exactly as a controlled, large LOCA. The pressure history in the containment (Figure 6-3) therefore agrees with the pressure transient in Figure 6-2 until the switch to sump recirculation.



1. Blowdown phase of the primary coolant
2. Refilling and reflooding the RPV
3. Low-pressure injections from the flood tanks
4. Switching to sump recirculation (at this time failure of ECCS is assumed in core melt accident 1)
5. Dry boil-out of the RPV
6. Melt-down of reactor core, hydrogen formation from the Zr-H_2O-reaction and hydrogen combustion in the containment, residual water vaporization
7. Melt through the RPV
8. Beginning of degradation of foundation and inner shielding
9. Vaporization of sump water

Figure 6-3. Pressure history in the containment for a core melt accident

We assume failure of the emergency and RHR system upon switching to sump recirculation, i.e., no emergency coolant is injected into the RPV. The RPV, which is initially full, begins to dry out. Steam moves through the leak into the containment. During this phase of the accident, the energy input connected with the movement of this steam into the containment atmosphere and the heat removal by the containment internals are approximately in balance. The pressure in the containment stays nearly constant.

The onset of the zirconium-water reaction releases hydrogen to the containment. Because of the low-ignition temperature and the high hydrogen release temperature, phase A of the study assumes that the hydrogen burns immediately by spontaneous combustion. Combustion of the hydrogen releases large quantities of heat. The pressure in the containment thus briefly increases.

During core melt, large quantities of fission products escape from the degraded fuel elements into the containment. The residual heat of the released fission products is transmitted directly to the containment atmosphere. This effect was considered in all calculations.

When the molten core collapses into the residual water in the lower plenum of the RPV, the generation of hydrogen and release of fission products is initially stopped. Vaporization of the residual water, however, further increases pressure in the containment. While the melt penetrates through the RPV, almost no mass or energy flows from the RPV into the containment. The pressure in the containment again drops off. After the melt has dropped into the reactor cavity, it gradually penetrates into the concrete, causing the water contained in the concrete to vaporize. If we assume that water vapor is liberated into the containment, then the resulting energy transfer to the containment would be relatively low, and the pressure would drop. However, recent studies indicate that under the prevailing conditions, the water vapor is completely reduced by the metal parts of the melt, and the resulting hydrogen generated flows to the containment and burns there. The pressure in the containment then increases slightly.

In the subsequent course of the accident, it is assumed that the melt comes into contact with the sump water. As a result of sump water vaporization, greater quantities of steam flow into the containment, and the pressure increases greatly over the long term. Studies in the appendix estimate a

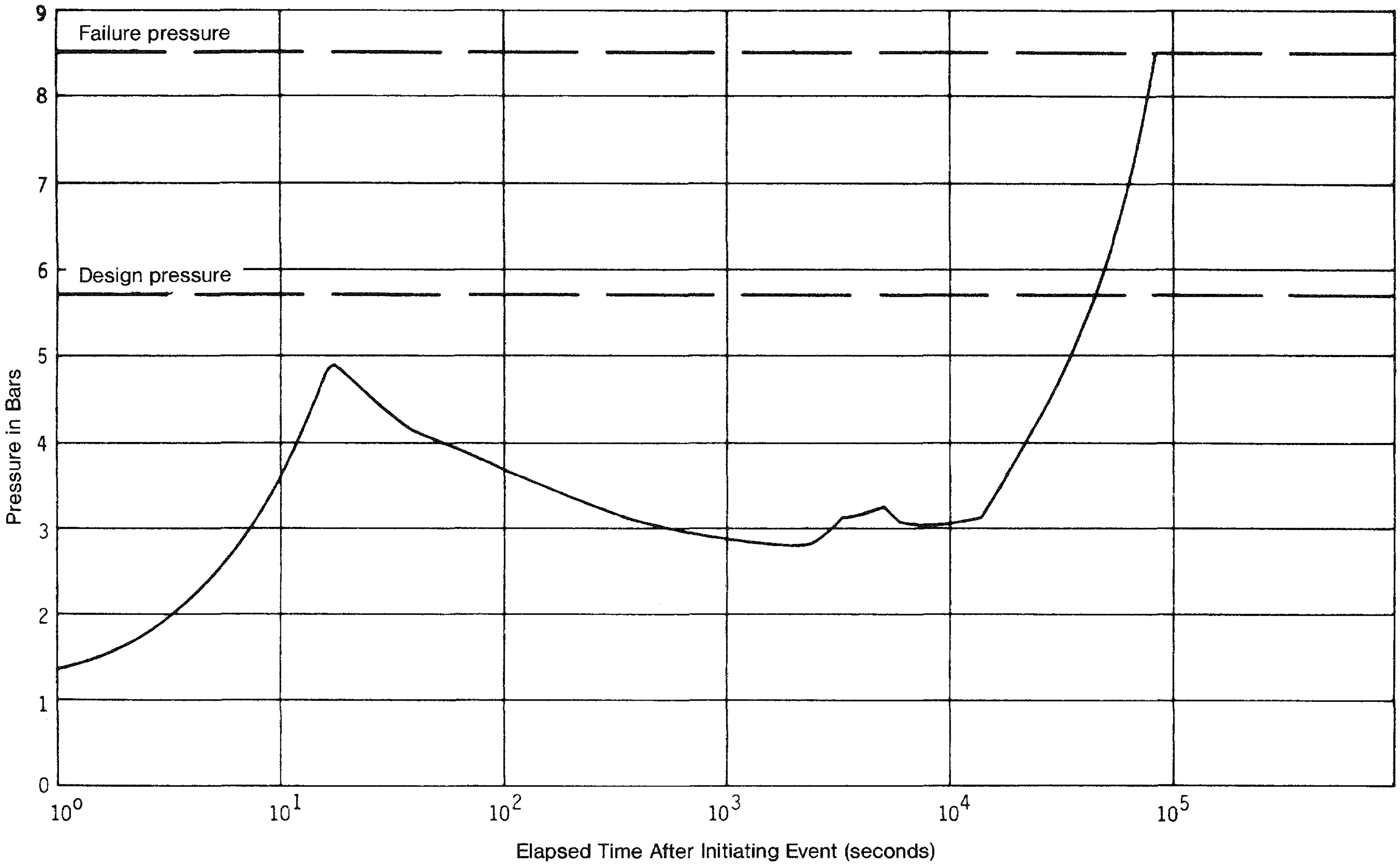


Figure 6-4. Pressure in the containment for core melt accident 2

failure pressure of about 8.5 bar for the containment. If we use pessimistic assumptions, this pressure is reached in the present case somewhat more than one day after the beginning of the initiating event; thus, at this time an over-pressure failure of the containment must be assumed.

Figure 6-4 shows the pressure history in the containment for core melt accident 2 (Section 6.2.3). The pressure in the containment behaves in a similar manner to core melt accident 1. However, the pressure level overall is somewhat higher, and the assumed failure pressure of the containment is reached earlier. This is due to the fact that in core melt accident 2, an earlier failure of the emergency and RHR system is assumed than in core melt accident 1. All processes therefore take place more quickly, and lead to the higher pressure behavior.

Overall, the calculations provide the following result: In the initial phase of core melt accidents, the containment pressure remains below the design pressure. An over-pressure failure of the containment during this period is not to be anticipated. The release of fission products and hydrogen to the containment plays only a minor role in the pressure build up, if we assume an immediate, continuous combustion of hydrogen. In all core melt accidents, however, vaporization of sump water occurs after the internal concrete shield has melted. This causes a long-term, severe pressure increase that finally results in over-pressure failure of the containment.

In core melt accident 1 if we assume a leak in the containment from the beginning of the initiating event, then during the entire accident sequence mass and energy flow from the containment to the environment because of the pressure gradient. This reduces pressure in the containment. For the leak sizes presented in Section 6.3.2.1, the pressure history in the containment exhibits the following behavior (Figure 6-5):

- For a small leak in the containment, a long-term over-pressure failure of the containment results. Since the pressure buildup is slower than for a tight containment, the over-pressure failure takes place later.
- For a medium leak in the containment, the pressure also increases slowly. For core melt accident 1, there is still an over-pressure failure of the containment.
- For a large leak in the containment, the pressure stabilizes over the long term to a value of between 1 and 2 bar. An over-pressure failure of the containment is impossible.

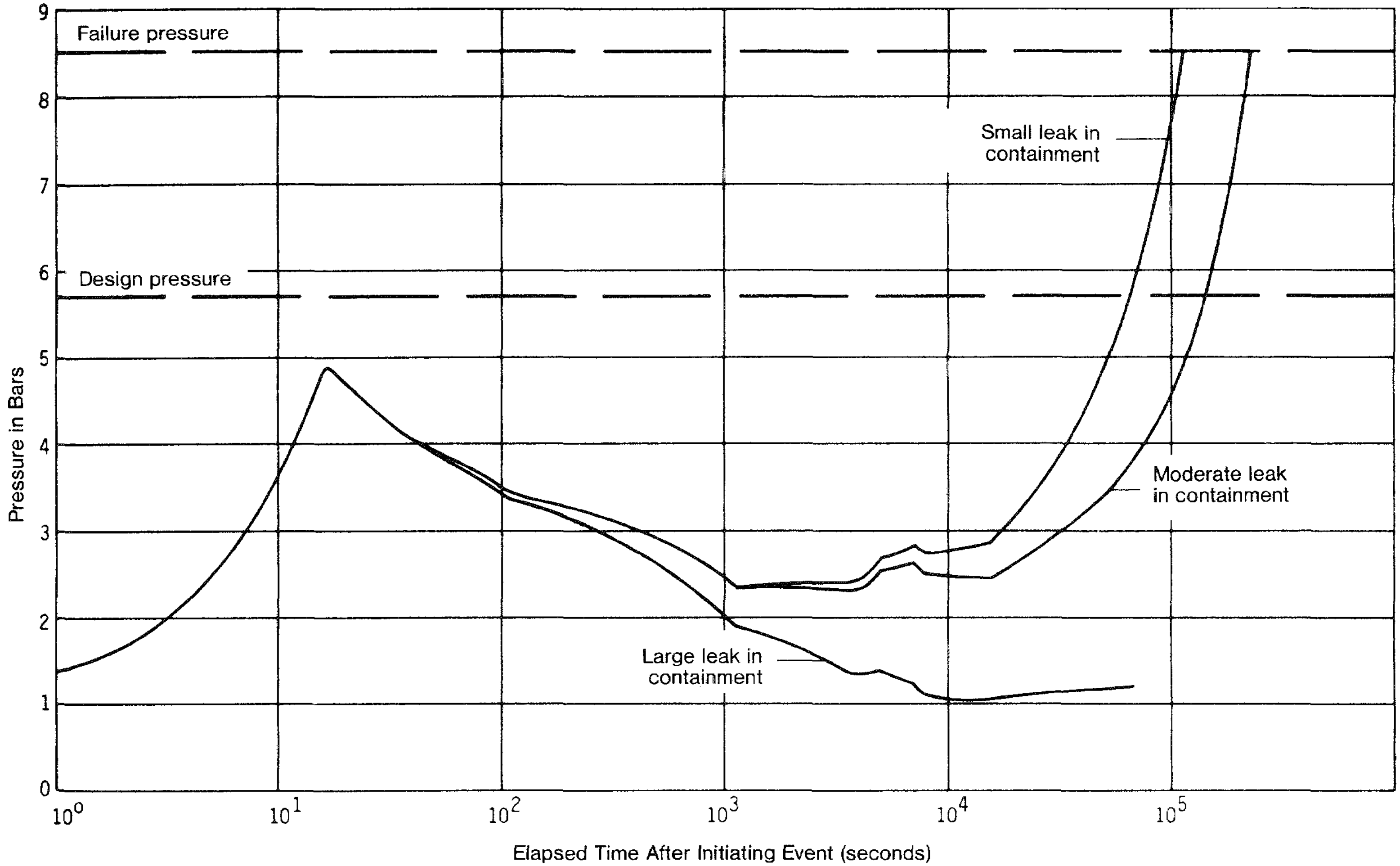


Figure 6-5. Pressure history in the containment for core melt accident 1 assuming various size leaks in the containment

In two phases of the core melt accident, hydrogen forms as a result of chemical processes and is liberated to the containment. If this hydrogen does not burn, it may possibly concentrate in the containment and cause a sudden combustion and explosion. With reliance on analysis of the H_2 explosion in WASH-1400, phase A of this study assumes no failure of the containment occurring in this case. The validity of this assumption must still be checked in phase B of the study.

6.3.3 Summary

Section 6.3.2 examined the various potentials for failure of the containment during the course of a malfunction or an accident. The results of these investigations will now be summarized.

During accidents coped with by the safety systems, leakage (failure to isolate) is the only possible type of containment failure.

The following failure modes of the containment must be considered for core melt accidents:

- α failure of the containment as a result of steam explosion in the RPV (see Section 6.4)
- β_1 large leak in the containment (no later over-pressure failure)
- β_2 medium leak in the containment (plus possible over-pressure failure)
- β_3 small leak in the containment (plus over-pressure failure)
- δ over-pressure failure.

The Greek letters are abbreviations for the containment failure mode. Selection of the letters was adopted from WASH-1400.

WASH-1400 assumed that a failure of the containment with a release into the atmosphere has more severe effects than melt-through of the foundation and the associated release of fission products to the soil. If containment failure occurs before melt-through of the foundation so that a release to the atmosphere occurs, the release to the soil is ignored thereafter in WASH-1400. This same method from WASH-1400 was adopted here.

In the case of the German reference plant, as the studies in Section 6.3.2 show, a core melt accident that occurs before destruction of the foundation always leads to containment failure by a different route and thus to a release of radioactive substances to the atmosphere. The release of fission products into the soil after melt-through of the foundation is therefore not further pursued.

6.4 STUDIES ON STEAM EXPLOSION

6.4.1 Introduction

During an accident connected with core melt, if molten core materials come into contact with coolant, then heat is transferred to the coolant, causing the coolant to vaporize. The volume increase caused by the vaporization leads to a pressure increase. Depending on the prevailing conditions, the coolant may boil or, in an extreme case, spontaneously vaporize, causing a steam explosion. For a steam explosion to occur, several conditions are important:

- The surface area for transfer of heat from the molten material to the coolant must be extremely large; for this it is necessary that the molten core disintegrate to extremely small fragments on the order of 100 to 4,000 micrometers (10^{-6} m).
- The fragmented core melt must be in close contact with the coolant; i.e., the core melt fragments must disperse uniformly throughout the coolant in the reaction zone. This condition must exist in a very short time in order to achieve a coherent, i.e., uniform, reaction of the participating reactants.
- Very good heat transfer conditions between the two liquids (molten fuel and coolant) must last long enough to allow sufficient energy transfer to the coolant for a subsequent, spontaneous vaporization.

This type of interaction between molten core materials and coolant can occur if a larger quantity of molten core material suddenly drops into the lower plenum of the water-filled RPV. The model conception of how such a contact between melt and water can occur is discussed in detail in the appendix.

If, in spite of the low probability, we assume a powerful reaction between molten core material and coolant, then the surrounding structures of the RPV are loaded. Below, we will discuss the results of initial, orienting calculations on the loads of the RPV. The objective of these studies is to check whether and to what extent the RPV can fail.

6.4.2 Interaction Between Core Melt and Coolant

The mechanical load on the RPV or containment during a steam explosion depends first on the mass of molten material that can react with the residual water. The total mass of the core is 149 tons (116.3 tons fuel; 30.3 tons zirconium; 2.4 tons steel). According to WASH-1400, a reaction of a molten mass 20% greater than the core endangers the containment. In (3,4) the conclusion is drawn that in a reaction that involves more than 1% of the core, the loads on the RPV resulting from a steam explosion may not be accommodated under some circumstances.

The discussion of fuel masses which can simultaneously react with coolant in the lower plenum of the RPV is connected with uncertainties in the description of the behavior of the melt process (see also Section 6.2).

When considering the melt process, two boundary cases are possible: either the melt drops continuously over a long period of time into the residual water of the lower plenum; or the core support structure suddenly fails in whole or in part due to the excess temperature, and a large molten mass falls into the water. In the former case--that is, droplets falling into the water--the water would constantly vaporize over a long period of time without causing significant pressure increase through vapor explosion.

In the latter case, it is physically unrealistic to imagine that the entire falling mass could be liquid throughout. Because of the experimentally proven low viscosity of the melt, droplet formation would occur even before the collapse. Nevertheless, in the following cases we assume that most of the melt would suddenly come into contact with water.

Clarification of the phenomena occurring during a steam explosion is the object of numerous experimental and theoretical investigations. By rough calculations it can be shown that the energy carried into the water from the melt is greatest when the mixture is composed of one part melt by volume and 1 to 2 parts water by volume. Therefore, as a precondition for a powerful steam explosion, the initially cohesive molten mass would have to be fragmented over a total volume 2 to 3 times as large. This would have to occur in an extremely short time, that is, within a few milliseconds. If we assume a melt mass of 10 tons to be subjected to this fragmentation, then the forces needed would be so great that it is hardly possible they could be made available in the course of a steam explosion.

Simple computer models that calculate heat transport from the core melt to the coolant and its conversion to mechanical energy provide conversion factors (under pessimistic assumptions) that can be up to 10%. However, it must be remembered that the necessary fine and coherent fragmentation is naturally less likely as the quantities of core melt become greater. A fragmentation of larger quantities of core melt--on the order of tons--into particle diameters of several hundred to several thousand micrometers necessary for the occurrence of a steam explosion is extremely unlikely.

Even though the occurrence of a steam explosion does not seem realistic--due to the existing difficulties of a comprehensive analytical treatment--in the following section we shall examine loads on surrounding structures by postulating a steam explosion of larger quantities of molten materials with a conversion of thermal to mechanical energy as indicated by theoretical models.

6.4.3 Load on the Surrounding Structures

To determine the effects of a postulated steam explosion in the RPV, WASH-1400 uses a model that treats the loads in an axial direction. In order to be more complete than in WASH-1400, we used the SEURBNUK (5) computer program in this study for initial, orienting calculations on pressure vessel load.

The energy release in a postulated steam explosion is simulated by a reaction bubble. This is identical to the reaction zone between molten core materials and coolant in the lower plenum of the RPV. The location and size of the reaction zone were changed for parameter studies, and the time-dependent pressure in the reaction bubble was predetermined in accordance with the characteristic profile of a steam explosion.

The model calculations considered the following phenomena. Under the assumption of a pessimistic bubble size at time zero, which can be assumed for a steam explosion with respect to an upper estimation, expansion of the reaction bubble in the water results in a lateral, but primarily upward-directed compression of the fluid. Since relatively little water is available, the reaction bubble soon penetrates the surface of the water and mixes with the water vapor located above it.

The water continues to press against the lid in an annular flow after penetrating the water surface. There it is diverted by the spherical lid and meets in the middle. Upon rediversion downward, a local stagnation pressure forms while the water continues to flow downward.

The lid is loaded twice as a result of the steam explosion. First, a shockwave proceeding from the reaction bubble travels through the residual water to the water surface. Because of the density difference between water and water vapor, the shockwave--provided the reaction zone does not encompass the entire volume of residual water--is reflected at the free surface and runs largely back into the water as a rarefaction wave, whereas the energetically smaller fraction proceeds as a shockwave through the water vapor in the direction of the lid. On the way to the lid, the shockwave loses energy and peak pressure and is then reflected at the lid or even beforehand, by existing internals. As a result of this load, the lid is less endangered than the base of the RPV.

The second, chronologically shifted load is due to the previously described surge of the water in an annular flow, after the reaction bubble has penetrated through the surface of the water. Because no water hammer strikes it, the lid is not expected to fail even as a result of this pressure load.

In addition to the load on the lid, direct load on the bottom of the RPV was examined with SEURBNUK. Parameter variations were implemented for this purpose. The bottom of the RPV is loaded primarily only in the first, acoustic phase. Only the high, extremely short-term pressures acting during this phase can lead to a plastic deformation of the walls of the RPV. The pressures occurring in the second phase of the steam explosion as a result of boiling of the coolant and loads caused in the RPV wall are below the yield strength of the vessel material in this estimation, and therefore provide no significant contribution to the total load.

Even though elongations of up to 1% are expected according to the present model investigations, for unfavorable parameter selection a failure of the RPV cannot be absolutely precluded. More accurate studies on this question were not possible in phase A of the study. With respect to an upper estimation of risk, the assumed probability of a steam explosion indicated that we should adopt the procedure used in WASH-1400. In phase B this question will be treated in detail in accordance with the results available at that time.

6.5 RESULTS OF CALCULATIONS ON FISSION PRODUCT RELEASE

6.5.1 Introduction

The present section concerns the inventory of fission products in the core, the potential release of these fission products from the core during malfunctions or accidents, the transport and deposition processes in the containment, and releases from the plant to the environment.

Should fission products move from the reactor core into the containment during a malfunction or accident, then their concentration in the containment atmosphere at the time of any release from the plant is considerably reduced by active and passive removal processes (e.g., spray systems or natural deposition) and by radioactive decay. Therefore, the containment with its surrounding buildings plays a special role as the last barrier for retention of the fission products. The retention effect of the containment is greater the longer the fission products remain in the containment. Therefore, even the consequences of a core melt accident can be significantly reduced by the protective function of the containment.

The processes from the formation of fission products down to release from the plant are treated by the models used in WASH-1400. In the model, only the parameters and safety systems specifically pertaining to the German reference plant Biblis B were used.

6.5.2 Model Description of the Transport and Deposition Processes

Possible radiological effects of malfunctions or accidents depend primarily on the magnitude of the existing radioactive inventory. To calculate the nuclear inventory broken down by nuclides, we used the ORIGEN (6) program, as did WASH-1400; this program can also treat complex activation and decay chains, and it exhibits good agreement with existing experiments.

ORIGEN calculates a broad spectrum of nuclides, a few of which are stable and thus of no importance to radiation exposure. As in WASH-1400, we considered in the calculations below (accident consequences model) only the 54 nuclides of particular importance because of their half-life values and radiological properties. The selection of these 54 nuclides assures that the main contributions to the radiation exposure are included in the determination.

In a core melt accident the release of fission products from the core generally takes place over a longer time period. During this time, the release rates can fluctuate over a broad range because of physical, chemical, and thermodynamic conditions. According to WASH-1400, these can be divided into the following four release phases:

1. gap release of gaseous and volatile fission products that accumulated primarily during normal operation in the fission gas plenum

2. meltdown release as a result of heating fuel rods to melt temperature

3. vaporization release during the interaction between melt and concrete foundation

4. release due to a steam explosion.

In accordance with the physical and chemical properties of the individual fission products, different fractions of the nuclear inventory are released in the various release phases. The individual elements can be divided into seven groups in accordance with their release behavior:

- noble gas (Kr-Xe)
- halogens (I-Br)
- alkali metals (Cs-Rb)
- tellurium group (Te)
- earth alkali metals (Ba-Sr)
- noble metals (Ru)
- non-volatile metal oxides (La)

The release rates for the individual groups of elements in the four release phases were taken from WASH-1400. Presently available results of new German experiments provide comparable or lower release factors. The fission products released from the core to the containment atmosphere exist there as gases or aerosols. The majority of released fission products are subject to different natural processes that reduce fission product concentration in the containment atmosphere. The deposition of noble gases or methyl iodide is negligible and is therefore not considered.

Elementary iodine is transported by natural convection and diffusion to the walls and surfaces as a result of the temperature gradient between the air and the structures of the containment. It is then deposited on the water film which has precipitated there. It was observed by experiment (7) that the iodine concentration in the air initially decreases greatly until it reaches about 1% of the initial concentration; thereafter the decrease is quite small. This is because after a certain time equilibrium between the iodine concentration in the water film and in the containment atmosphere sets in. The natural deposition of aerosols is caused by gravity and turbulent diffusion. Experiments have shown that a reduction in aerosol concentration in the post-accident atmosphere occurs primarily by gravity deposition on horizontal surfaces (7).

In accordance with their deposition behavior, we can therefore divide the fission products into the following groups:

- noble gases
- methyl iodide
- elementary iodine
- aerosols.

Because of their analogous deposition behavior, the following release groups are included in the aerosol group:

- alkali metals (Cs-Rb)
- tellurium group (Te)
- earth alkali metals (Ba-Sr)
- noble metals (Ru)
- non-volatile metal oxides (La).

The fission products released during a malfunction or core melt accident from the core and reactor coolant system first move into the surrounding space and spread out through diffusion, or by carry-over by flowing steam into other areas of the containment. A reduction in concentration of airborne fission products can take place in the individual areas at different deposition rates. Therefore, a multiple region model was used to divide the containment into several regions. This permits a realistic description of the deposition processes in the containment. The magnitude of fission product deposition in the individual regions of the containment depends on the physical boundary conditions and on a series of geometric parameters (compartment height for sedimentation of aerosols or the ratio of surface area to volume in the deposition of elementary iodine). Other important parameters for the change in airborne concentration of fission products in the individual regions are the overflow rate from one region to another and the outflow rate from the containment, because the dwell time of fission products in the individual regions decisively affects them. The large overflow rates from

WASH-1400 were used for the calculations. This procedure leads to pessimistic results.

Calculation of the transport and deposition processes described here, down to release of fission products from the plant, was performed by the CORRAL computer program used in WASH-1400 (8).

6.5.3 Results of Studies on Fission Product Release

The examination of fission product transport and release using the CORRAL computer program yields the relative concentrations of fission products in the individual regions and the accumulated, relative concentrations outside the containment as a function of time after the initiating event. As a reference point of concentration, we selected the total core inventory of the considered nuclide group: i.e., we obtained as a result the fraction of core inventory found in the atmospheres of the individual regions or outside the containment. The reduction in radioactivity by radioactive decay is not included in the CORRAL calculations but is considered later in accident consequence calculations.

Detailed calculations were performed for a broad spectrum of event sequences. In particular, fission product release to the atmosphere was determined for each of the two core melt accidents analyzed in Section 6.2 in combination with all assumed failure modes of the containment.

During a medium or large break in a primary coolant line coped with by the emergency cooling system, cladding can leak because of the temporary temperature increase in the core. The gaseous and volatile fission products collected in the fission gas plenum exit into the containment (gap release). Even in this case, release of fission products to the environment was studied under the assumption of various large leaks in the containment.

The detailed boundary conditions and results of all CORRAL calculations are presented in the appendix. Section 6.6.2 summarizes the results. Below, the studies are explained on the basis of several representative examples.

The first example is based on the following core melt accident (core melt accident 1): after a double-end break in a primary coolant pipeline, failure of emergency cooling is assumed upon switchover to sump recirculation. It is further assumed that during the course of the accident, an overpressure of the containment occurs at a later stage (see Section 6.3).

Immediately after an accident begins, the gaseous and volatile fission products are released from the fission gas plenums into the containment. The main contribution of fission product release from the core begins with the onset of core melt. After the molten core collapses into the residual water in the RPV, release from the melt is initially stopped. Once the residual water in the RPV has vaporized, fission products from the melt (vaporization release) are again released. This essentially stops when the melt comes into contact with the sump water.

Figures 6-6 and 6-7 illustrate the time history of fission product concentration in the containment atmosphere for the first 1000 minutes. The noble gases are almost entirely in the atmosphere of the containment after 1000 minutes, whereas the airborne concentration of elementary iodine has dropped to about 1% because of the natural deposition process on the interior surfaces of the containment wetted by water; this occurs after about 400 minutes. Thereafter, the airborne iodine concentration remains constant between the gaseous and liquid phase due to exchange effects. The fraction remaining in the air is released after overpressure failure, except for minor leakage quantities which exit beforehand.

A state of equilibrium among the aerosols does not exist. Sedimentation of aerosols due to gravity plays an important role here and causes a constant decrease in aerosol concentration. Since the aerosols are considered to have the same deposition behavior in the CORRAL program, the influence of release time can be seen by comparing nuclide groups Cs-Rb and Te. Both nuclide groups are completely released from the core. Whereas the majority of the Cs-Rb group is released during the melt phase into the containment, primary release of the Te-group occurs later (during the vaporization phase).

This different time behavior in the release from the core has no notable influence on the total quantity released to the outside during a core melt accident which results in over-pressure failure.

Fission product release from the containment to the environment for delayed overpressure failure is illustrated in Figure 6-8. The total released fraction of the core inventory for the various nuclide groups is plotted as a function of time. Note that the total released fraction of the core inventory is plotted logarithmically. Even before overpressure failure of the containment, a release is noted. This is attributable to the fact that in the calculations--analogous to

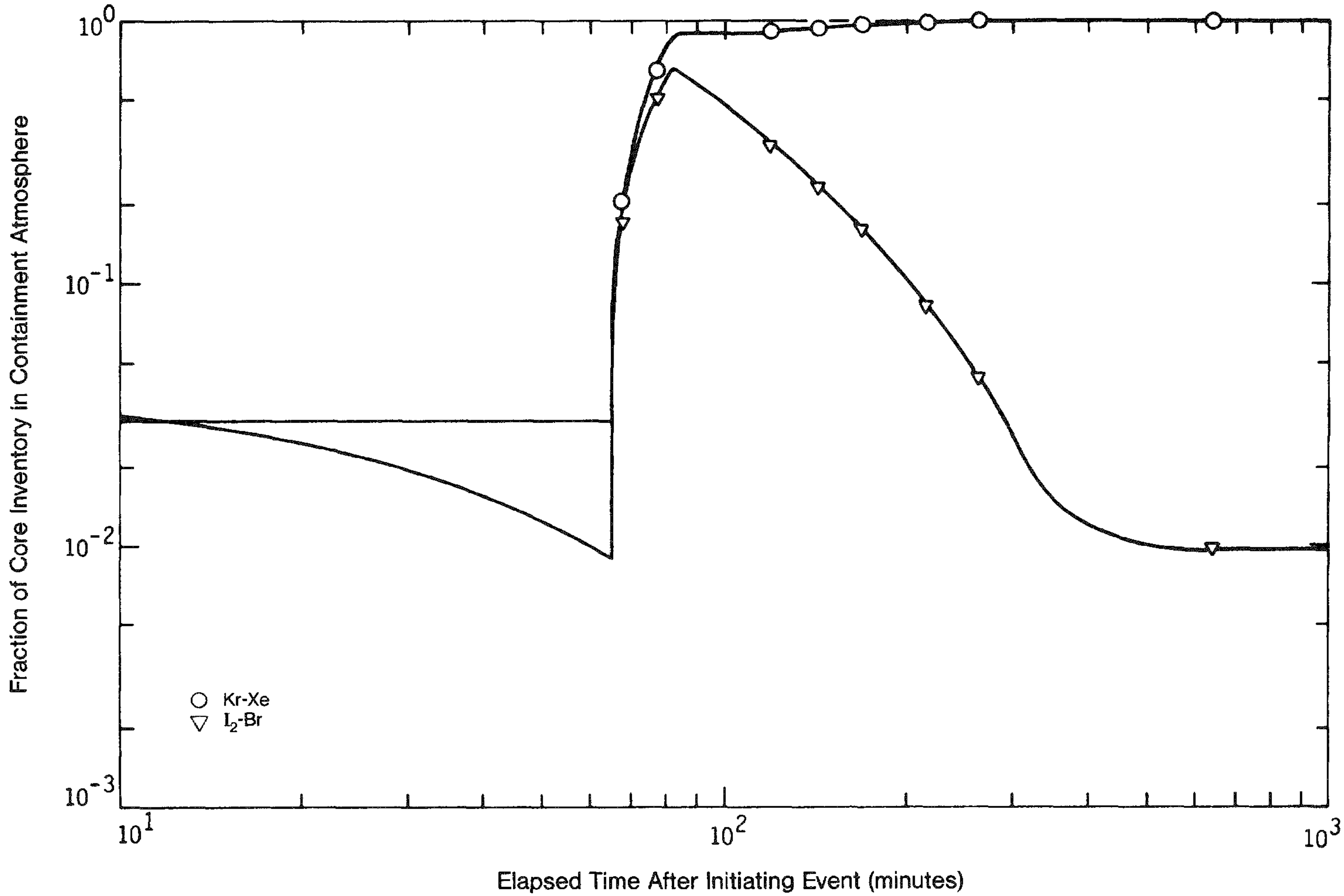


Figure 6-6. Fission product concentrations in the containment atmosphere for core melt accident 1 (Kr-Xe, I_2-Br)

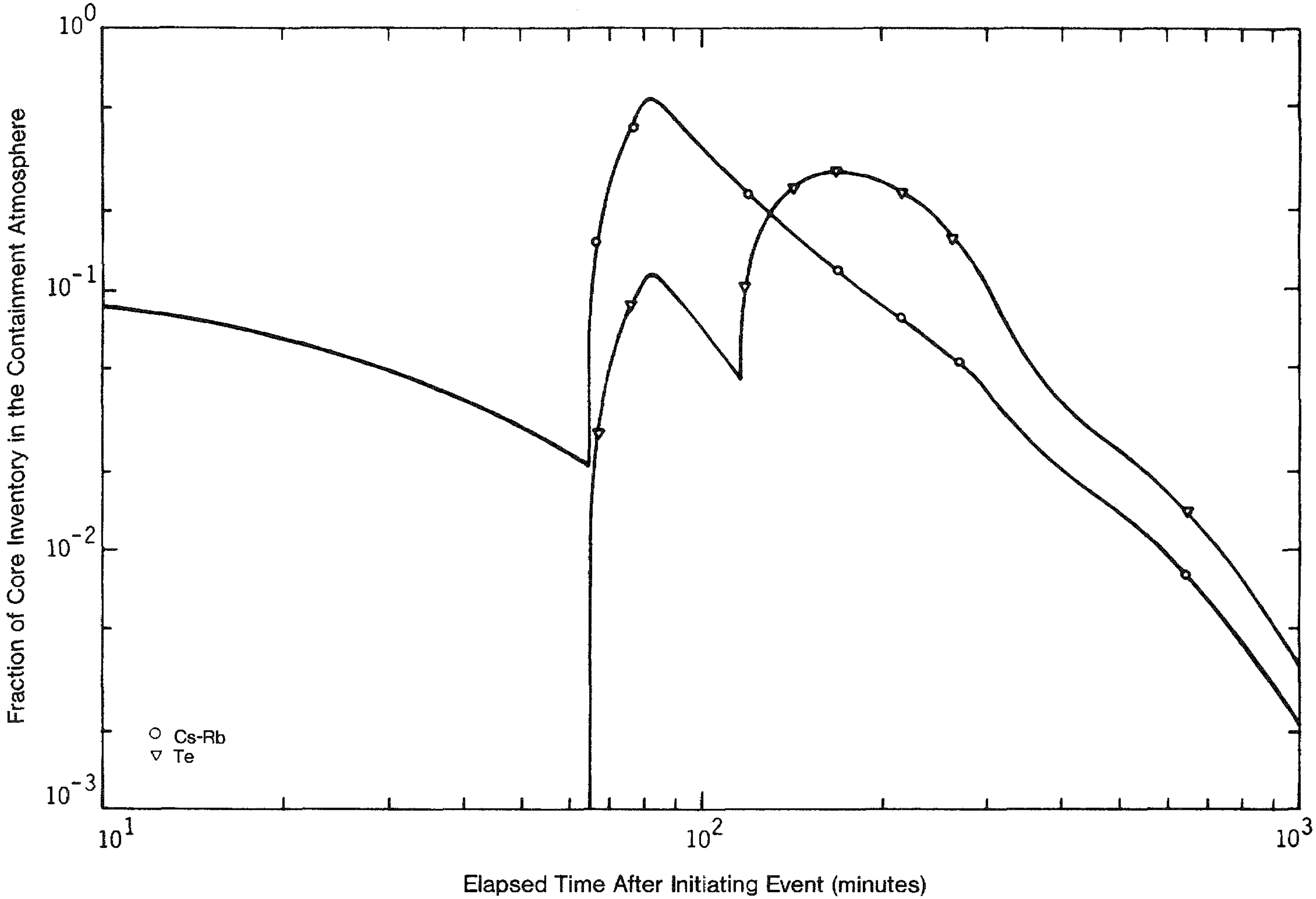


Figure 6-7. Fission product concentrations in the containment atmosphere for core melt accident 1 (Cs-Rb, Te)

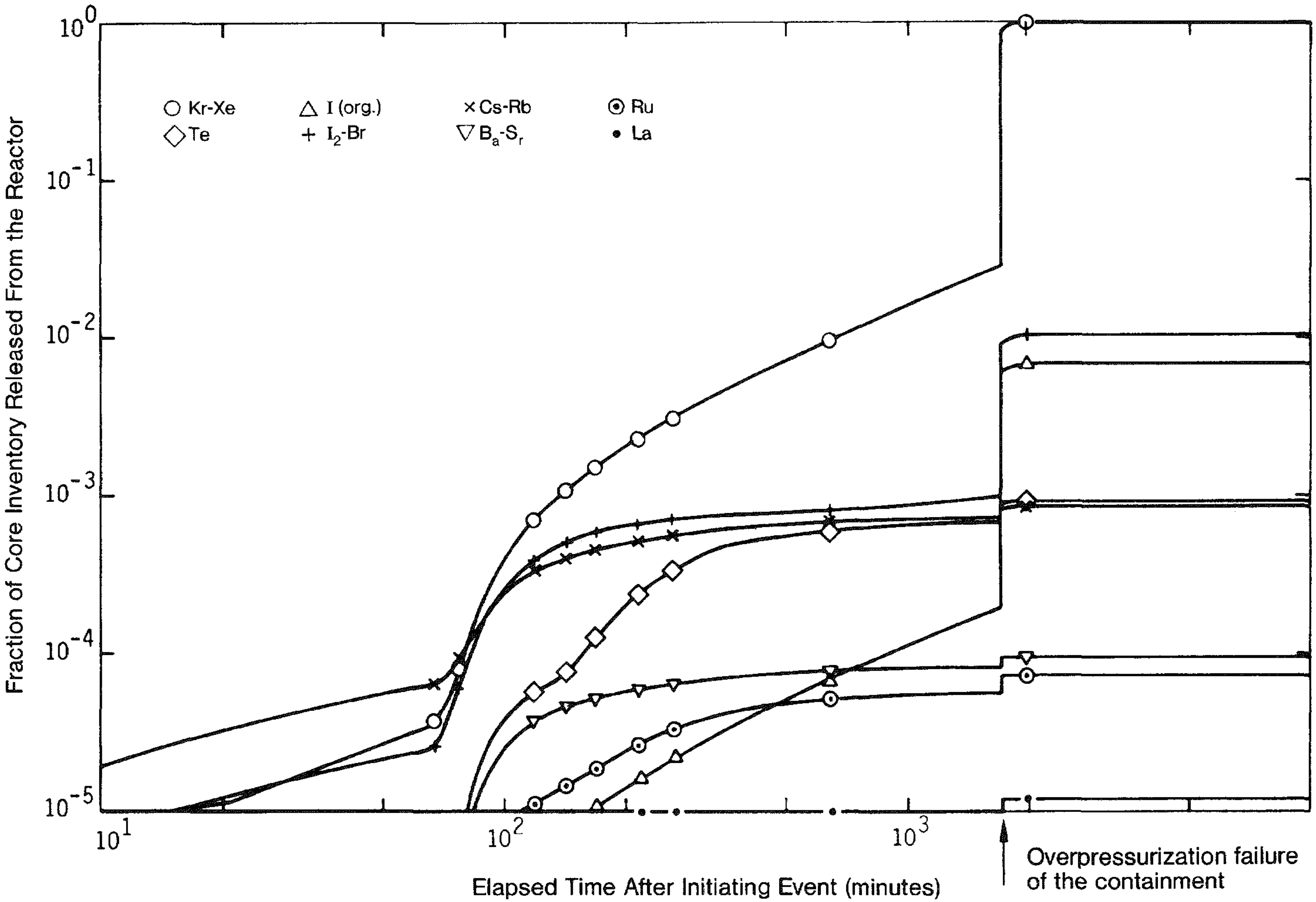


Figure 6-8. Fission product release from the containment for core melt accident 1 and overpressurization of the containment

WASH-1400--a pessimistic leak rate is assumed in the containment during this period that is ten-fold higher than the designed leak rate. Since no deposition processes are assumed in the containment for the noble gases and organic iodine, during and immediately after overpressure failure, practically complete release takes place. Elementary iodine is deposited until a state of equilibrium is attained between concentration in the fluid phase and concentration in the gaseous phase at the surfaces wetted by water. The fraction remaining in the air is released except for leakage losses, primarily by overpressure failure. In the case of aerosols, the concentration in the containment air is reduced by gravitational settling until no notable release occurs.

The results discussed above can be applied analogously to other containment failure modes. Figure 6-9 shows fission product release to the environment for the same core melt accident, except that we assume a large leak in the containment from the beginning of the initiating event. A severe increase of the total quantity of fission products released to the outside is observed during the melt period until the core drops into the residual water in the RPV. This observation is noted for all nuclide groups. Thereafter, release of the individual nuclide groups differs. Among the group of noble gases and for organic iodine, the main release from the core has already ended by the time the core drops into the residual water; the fission products held in the containment atmosphere are released to the outside. This is also true for the nuclide groups I_2-Br, Cs-Rb, and Ba-Sr. Because of deposition processes in the containment, release to the outside ends earlier for these nuclide groups. The remaining groups of nuclides (Te, Ru, and La) behave differently. The majority of these fission products are still present in the melt when the core collapses into the residual water. The main release from the core takes place in the vaporization phase. The rapid increase in total quantities of fission products released by these three groups to the outside during the vaporization phase is clearly seen.

In comparison to core melt accident with overpressure failure of the containment, a clearly higher release is seen for all nuclide groups, with the exception of noble gases and organic iodine. This effect is due to the fact that the average dwell time of fission products in the containment, and thus the deposition, is less for these groups. This effect is even more pronounced if we assume a steam explosion with subsequent direct failure of the containment for this particular core melt accident (Figure 6-10).

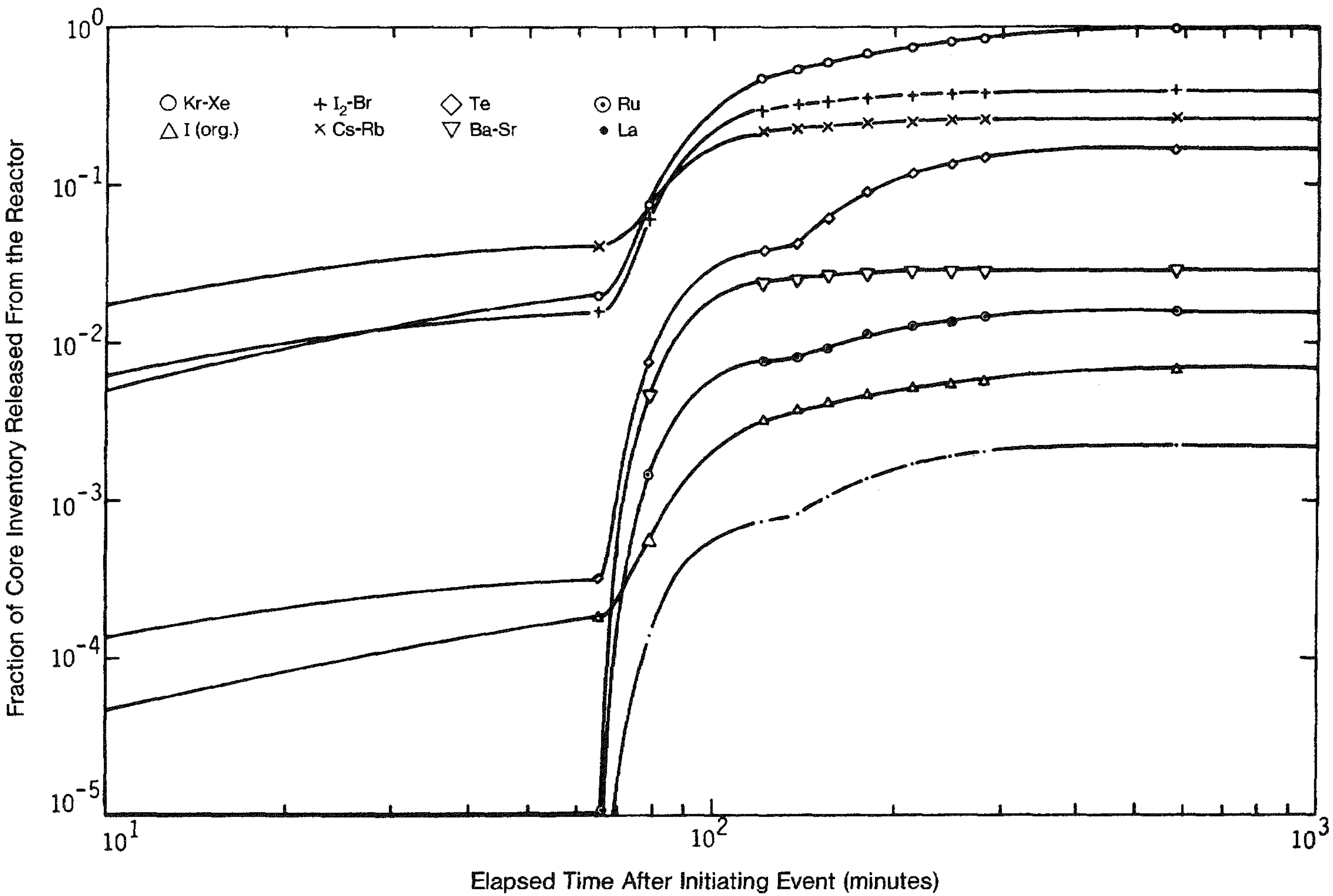


Figure 6-9. Fission product release from the containment for core melt accident 1 and a large leak in the containment

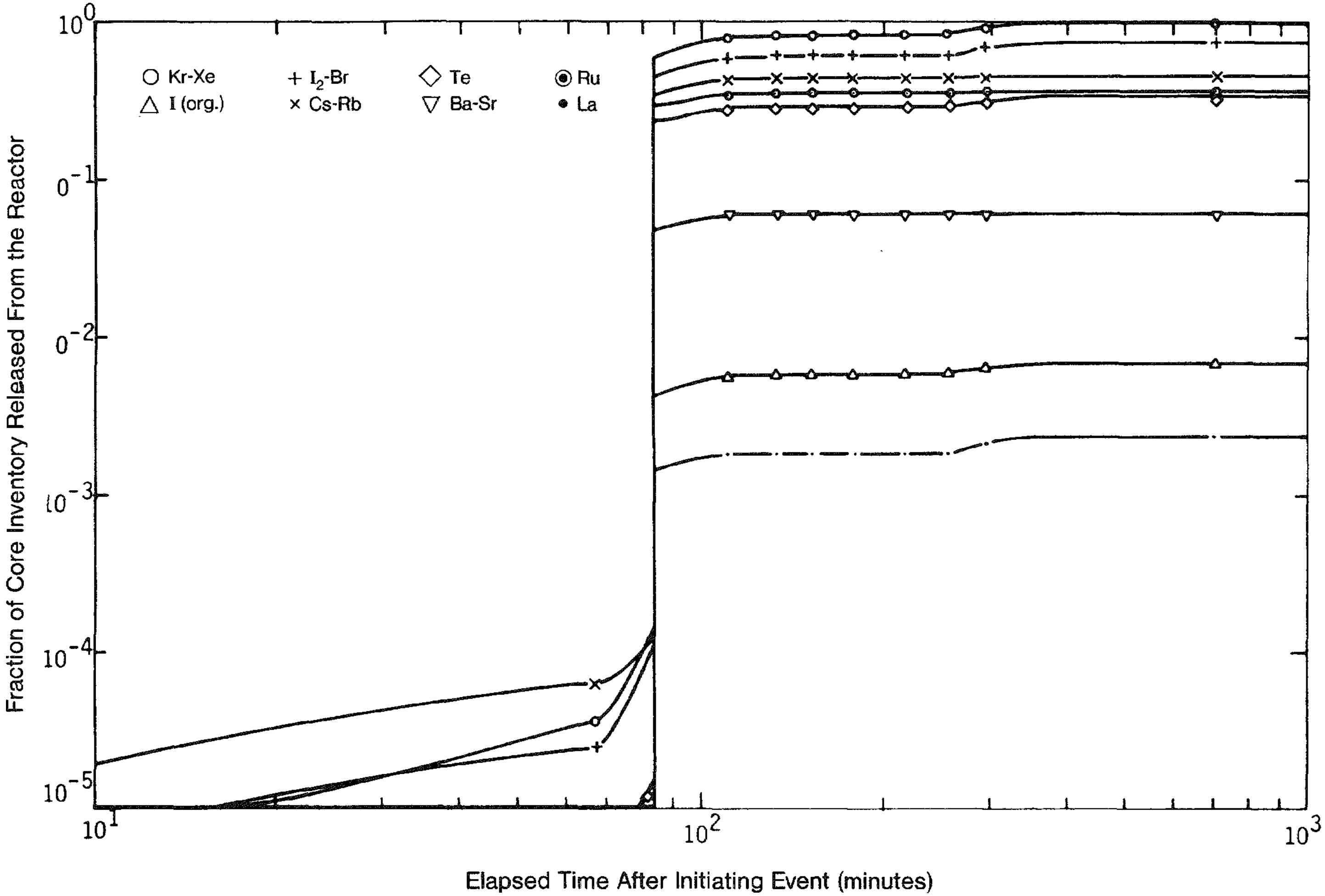


Figure 6-10. Fission product release from the containment core melt accident 1 assuming a steam explosion resulting in failure of the containment

It must be emphasized again that radioactive decay is considered only later in the accident consequence calculation, when fission products are in the environment. A simultaneous consideration of radioactive decay would result in a greater effective retention value by the containment--depending on half-life--and thus in a lesser release to the outside.

6.6 DETERMINATION OF RELEASE CATEGORIES

6.6.1 Background Information

The investigations in Chapter 5 have shown that many potential event sequences could lead to a core melt in accordance with the findings in Section 6.2.2. Except for time history, the physical processes in the core melt itself and in fission product release from the core are quite similar for the various core melt accidents. The discussion in Section 6.2 led to the result that in the framework of this study, the number of core melt accidents to be studied can be reduced to two representative core melt sequences.

These two accident sequences can cover all other core melt accidents, both with respect to chronology, as well as to fission product release into the containment.

For each of these two core melt accidents, fission product release from the plant was determined in combination with all assumed failure modes of the containment (see Section 6.5). The calculated releases can be categorized into representative releases, so-called release categories. The formation of these release categories is explained more carefully below.

6.6.2 Formation of Release Categories

If we analyze the results of the calculations of fission product release from the plant, we see:

- The release magnitude depends greatly on whether an accident leads to core melt and what failure mode is assumed for the containment.
- For core melt accidents the amount of release depends very little, by comparison, on which of the two representative core melt sequences is used as a basis for the calculations.

Therefore, it is possible to summarize the releases into groups. The discrimination takes place according to whether core melt occurs and which failure mode of the containment is assumed.

As discussed in Sections 6.3 and 6.4, the following containment failure modes are considered for core melt accidents:

- leakage of the containment
- over-pressure failure of the containment
- steam explosion in the RPV which leads to destruction of the containment.

Plant system investigations to determine potential leaks from the containment as well as the results of calculations on fission product release have shown the advantages of dividing the failure mode "leak from the containment" into three sections, according to the size of the leak.

Among the accident sequences that do not result in core melt, only the failure mode "leak from the containment" is possible.

By means of the criteria explained above, we can form a total of eight release groups. Each event sequence can be assigned to one of these eight groups and each group described by a representative release (release category). The determination of representative release is performed analogously to the method in WASH-1400. For each group of nuclides, the released fractions arising from an event sequence in a particular category are compared. The representative release is then formed from the worst values of each group of nuclides. For example, Table 6-2 shows the cumulative fractions of the core inventory released from the plant for the event sequences of a release category. As we can see from the table, core melt accident 2 leads to the greater releases. Therefore, we use the released fractions from core melt accident 2 for the representative releases of the category. Since most release occurs one to four hours after an accident has begun, the beginning of release at one hour after initiation of the accident and the end of release at four hours after the accident are assumed for the release category.

It was assumed that the total fission product release from the plant takes place during this time.

The release categories formed by this method are compiled in Table 6-3 and are explained below. Release category 1 includes all core melt accidents with an assumed steam explosion in the RPV of such magnitude that failure of the containment occurs. The fission product release from the plant is greatest for this

Table 6-2. Cumulative fraction of core inventory released from the plant for the event sequences of a release category (release category 2)

Time	Kr-Xe	I(org)	I_2-Br	Cs-Rb	Te	Ba-Sr	Ru	La
Core melt accident 1, large leak in the containment								
0- 1.1 h	2.0-2*	1.8-4	1.6-2	4.1-2	3.1-4	4.7-6	--	--
0- 1.4 h	1.2-1	8.6-4	9.3-2	1.0-1	1.3-2	8.3-3	2.5-3	2.5-4
0- 1.9 h	4.3-1	3.0-3	2.8-1	2.1-1	3.7-2	2.4-2	7.3-3	7.3-3
0- 4.9 h	8.5-1	5.9-3	3.8-1	2.6-1	1.5-1	2.9-2	1.5-2	2.1-3
0- 10 h	9.9-1	6.9-3	3.9-1	2.6-1	1.6-1	3.0-2	1.6-2	2.3-3
0-100 h	1.0	7.0-3	3.9-1	2.6-1	1.6-1	3.0-2	1.6-2	2.3-3
Core melt accident 2, large leak in the containment								
0- .62 h	1.4-2	1.3-4	1.3-2	3.6-2	2.7-4	4.1-6	--	--
0- .87 h	9.6-2	7.1-4	8.1-2	9.0-2	1.1-2	7.2-3	2.1-3	2.1-4
0- 1.3 h	4.1-1	2.9-3	2.7-1	2.2-1	3.8-2	2.5-2	7.5-3	7.5-4
0- 4.3 h	8.7-1	6.1-3	3.9-1	2.8-1	1.7-1	3.2-2	1.7-2	2.4-3
0- 10 h	1.0	7.0-3	4.0-1	2.9-1	1.9-1	3.2-2	1.7-2	2.6-3
0-100 h	1.0	7.0-3	4.0-1	2.9-1	1.9-1	3.2-2	1.7-2	2.6-3

* In the table, an abbreviated method is used to illustrate the powers of 10, i.e., 2.0-2 means 2.0 x 10^{-2}.

category for two reasons. First, the majority of release takes place immediately after core melt. Because of the extremely short dwell time of fission products in the containment atmosphere, deposition effects are small. Second, as assumed in WASH-1400, the processes connected with a steam explosion lead to an additional release of fission products in comparison to core melt accidents without steam explosion (see Section 6.5.2).

Release category 2 contains core melt accidents that assume a large leak in the containment. For this leak size, no pressure buildup in the containment takes place over the long term. The fission products released from the fuel move through the leak into the containment and to the atmosphere after a relatively short dwell time. Even though deposition processes do not play a significant role here, fission product release from the plant is smaller overall than for release category 1.

Release categories 3 and 4 contain core melt accidents where a medium or small leak in the containment is assumed. The outflow from the containment occurs much more slowly in both these cases than in a large containment leak. This means that the dwell time of fission products in the containment is relatively long, depending on the size of the leak, and that the deposition effects lead to a definite reduction in release from the plant.

Release categories 5 and 6 contain core melt accidents where the containment is initially intact. But over the long term, due to the results described in Section 6.3, over-pressure failure of the containment has to be anticipated. Analogous to WASH-1400, a pessimistic assumption is made of a leak ten-fold higher than designed before over-pressure failure occurs. This leakage moves into the annulus between containment and concrete shield and is then diverted to the environment through the annulus exhaust air handling system via the filter and stack. In contrast to category 6, category 5 assumes a failure of the exhaust air handling system or of the filter. Calculations on the event sequences of release categories 5 and 6 show that fission product release from the plant takes place over a long time period and that the release for several groups of nuclides before over-pressure failure is at the same level as for over-pressure itself. For this reason, the released fractions are given for three different time frames.

Release categories 7 and 8 contain LOCA coped with by the emergency cooling system; these LOCAs were caused by a medium or large break in a primary coolant pipeline. Under these accident sequences the core remains intact, with the exception of possible damage to cladding. Therefore, only the gaseous and volatile fission products collected in the fission gas plenums can be released from the core. In comparison with core melt accidents, this release is relatively small. For release category 7, we assume a large leak in the containment. For release category 8, the containment is intact. However, a leakage ten-fold higher than design was assumed, analogous to the conservative assumption in WASH-1400.

The release categories described here contain the entire spectrum of radioactive releases, beginning from a controlled LOCA down to the worst core melt accident. Calculations of accident consequences (Chapter 7) are based on these release categories.

6.6.3 Frequencies of Release Categories

In Section 6.6.2 we described the formation of the various release categories. Their important parameters are compiled in Table 6-3. The frequencies of release categories are discussed in greater detail in the present section.

All event sequences that lead to radioactive release to the environment can be assigned to one of the eight release categories. Individual release categories were determined to pessimistically cover the event sequences--with respect to fission product release--contained in the category. The frequencies of the individual release categories result from the sum of frequencies of the particular event sequences assigned thereto.

Table 6-4 contains the event sequences for each category that contribute significantly to the frequency of that category. An expected frequency is given for each of these dominant event sequences.

The abbreviations introduced in Chapter 5 denote event sequences. These are again explained in the legend of Table 6-4, together with the abbreviations for the containment failure modes.

Table 6-4 also contains expected frequencies and parameters of the complementary cumulative distribution functions for the frequencies of the individual categories. The distribution functions are characterized by the median and by the upper and lower bounds of the 90% confidence interval. The distribution functions are determined by summation of distribution functions for the frequency of the event sequences contained in the particular categories.

Table 6-3. Release categories

Release category	Description	Time of release (h)	Duration of release (h)	Release height (m)	Released energy (10^6 KJ/h)	Frequency of release (per year)	Released fraction of core inventory							
							Xe-Kr	I(org)	I_2-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru**	La#
1	Core melt with steam explosion	1	1	30	540	2×10^{-6}	1.0	7.0×10^{-3}	7.9×10^{-1}	5.0×10^{-1}	3.5×10^{-1}	6.7×10^{-2}	3.8×10^{-1}	2.6×10^{-3}
2	Core melt, large leak in containment (300mm diameter)	1	3	10	15	6×10^{-7}	1.0	7.0×10^{-3}	4.0×10^{-1}	2.9×10^{-1}	1.9×10^{-1}	3.2×10^{-2}	1.7×10^{-2}	2.6×10^{-3}
3	Core melt, moderate leak in containment (80mm diameter)	2	3	10	1	6×10^{-7}	1.0	7.0×10^{-3}	6.3×10^{-2}	4.4×10^{-2}	4.0×10^{-2}	4.9×10^{-3}	3.3×10^{-3}	5.2×10^{-4}
4	Core melt, small leak in containment (25mm diameter)	2	3	10	--	3×10^{-6}	1.0	7.0×10^{-3}	1.5×10^{-2}	5.1×10^{-3}	5.0×10^{-3}	5.7×10^{-4}	4.0×10^{-4}	6.5×10^{-5}
5*	Core melt, overpressurization failure, failure of the stack filters	0 1 25	1 1 1	10 10 10	-- -- 200	2×10^{-5}	2.0×10^{-5} 2.3×10^{-2} 9.8×10^{-1}	1.8×10^{-7} 1.6×10^{-4} 6.8×10^{-3}	1.8×10^{-5} 9.6×10^{-4} 9.6×10^{-3}	4.7×10^{-5} 6.7×10^{-4} 4.5×10^{-4}	3.6×10^{-7} 6.7×10^{-4} 7.7×10^{-4}	5.5×10^{-9} 8.0×10^{-5} 4.7×10^{-5}	-- 5.5×10^{-5} 5.3×10^{-5}	-- 8.8×10^{-6} 9.5×10^{-6}
6**	Core melt, overpressurization failure	0 1 25	1 1 1	100 100 10	-- -- 200	7×10^{-5}	2.0×10^{-5} 2.3×10^{-2} 9.8×10^{-1}	1.8×10^{-9} 1.6×10^{-6} 6.8×10^{-3}	1.8×10^{-8} 9.6×10^{-7} 9.6×10^{-3}	4.7×10^{-8} 6.7×10^{-7} 4.5×10^{-4}	3.6×10^{-10} 6.7×10^{-7} 7.7×10^{-4}	5.5×10^{-12} 8.0×10^{-8} 4.7×10^{-5}	-- 5.5×10^{-8} 5.3×10^{-5}	-- 8.8×10^{-9} 9.5×10^{-6}
7	LOCA, large leak in containment	0	1	10	9	1×10^{-4}	1.7×10^{-2}	3.7×10^{-5}	5.3×10^{-3}	1.3×10^{-2}	2.5×10^{-5}	2.5×10^{-7}	0.	0.
8	Controlled LOCA	0	6	100	--	1×10^{-3}	4.6×10^{-4}	1.0×10^{-8}	1.2×10^{-8}	2.1×10^{-8}	4.1×10^{-11}	4.1×10^{-13}	0.	0.

* Since the release takes place over a longer time period, the release fractions are given separately for three time intervals.

** Includes Ru, Rh, Co, Mo, Tc.

\# Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

Table 6-4. Dominant accident sequences in the individual release categories

	Release Categories							
	1	*2*	*3*	*4*	*5*	*6*	*7*	*8*
Major leak A	AH-α 2×10^{-9}	AG-β_1 1×10^{-8}	AF-β_2 3×10^{-10}	AF-β_3 4×10^{-9}	AF-η 1×10^{-9}	AH-δ 5×10^{-8}	A-β_1 6×10^{-8}	A 3×10^{-4}
	AF-α 4×10^{-9}	AB-β_1 1×10^{-8}	AE-β_2 1×10^{-9}	AE-β_3 7×10^{-9}	AE-η 6×10^{-9}	AF-δ 1×10^{-7}		
	AE-α 2×10^{-9}					AE-δ 4×10^{-8}		
	AD-α 5×10^{-9}					AD-δ 2×10^{-7}		
Sum A	1×10^{-8}	2×10^{-8}	1×10^{-9}	1×10^{-8}	7×10^{-9}	4×10^{-7}	6×10^{-8}	3×10^{-4}
Intermediate leak S_1	S_1H-α 4×10^{-9}	S_1G-β_1 4×10^{-8}	S_1F-β_2 1×10^{-9}	S_1F-β_3 8×10^{-9}	S_1F-η 3×10^{-9}	S_1H-δ 2×10^{-7}	S_1-β_1 2×10^{-7}	S_1 8×10^{-4}
	S_1F-α 1×10^{-8}	S_1B-β_1 2×10^{-8}	S_1E-β_2 5×10^{-10}	S_1E-β_3 4×10^{-9}	S_1E-η 2×10^{-9}	S_1F-δ 4×10^{-7}		
	S_1E-α 2×10^{-9}		S_1C-β_2 3×10^{-9}	S_1C-β_3 2×10^{-8}	S_1C-η 2×10^{-8}	S_1E-δ 8×10^{-8}		
	S_1D-α 4×10^{-9}		S_1CE-β_2 3×10^{-9}	S_1CE-β_3 1×10^{-8}	S_1CE-η 2×10^{-8}	S_1D-δ 2×10^{-7}		
	S_1C-α 2×10^{-8}					S_1C-δ 9×10^{-7}		
	S_1CE-α 2×10^{-9}					S_1CE-δ 8×10^{-8}		
Sum S_1	5×10^{-8}	6×10^{-8}	8×10^{-9}	5×10^{-8}	4×10^{-8}	2×10^{-6}	2×10^{-7}	8×10^{-4}
Minor leak in the main coolant line S_2	S_2IF-α 4×10^{-8}	S_2IG-β_1 1×10^{-7}	S_2IF-β_2 1×10^{-9}	S_2IF-β_3 1×10^{-8}	S_2IJ-η 5×10^{-8}	S_2IF-δ 1×10^{-6}		
	S_2IC-α 7×10^{-8}	S_2IB-β_1 8×10^{-8}	S_2IE-β_2 2×10^{-9}	S_2IE-β_3 1×10^{-8}	S_2IJCE-η 3×10^{-8}	S_2IC-δ 3×10^{-6}		
	S_2IJ-α 1.4×10^{-6}	S_2IJG-β_1 3×10^{-8}	S_2IC-β_2 4×10^{-9}	S_2IC-β_3 2×10^{-8}		S_2IJ-δ 5×10^{-5}		
			S_2ICE-β_2 4×10^{-9}	S_2ICE-β_3 2×10^{-8}				
			S_2IJ-β_2 2×10^{-8}	S_2IJ-β_3 1×10^{-7}				
			S_2IJF-β_2 2×10^{-9}	S_2IJF-β_3 1×10^{-8}				
			S_2IJCE-β_2 4×10^{-9}	S_2IJCE-β_3 1×10^{-8}				
Sum S_2	1.5×10^{-6}	2×10^{-7}	3×10^{-8}	2×10^{-7}	1×10^{-7}	5.5×10^{-5}		
Transients T	T_1IR-α 1×10^{-7}	T_1IR-β_1 7×10^{-9}	T_1IR-β_2 1×10^{-8}	T_1IJQ-β_3 3×10^{-7}	T_1IJQ-η 6×10^{-6}	T_1IR-δ 4×10^{-6}		
	T_1IJQ-α 2×10^{-7}	T_1IJQ-β_1 1×10^{-8}	T_1IJQ-β_2 1×10^{-7}		T_1IJMQ-η 4×10^{-7}	T_1IJQ-δ 1×10^{-6}		
	T_2R-α 2×10^{-8}					T_2R-δ 6×10^{-7}		
	T_2IJQ-α 5×10^{-8}					T_2IJQ-δ 2×10^{-6}		
	TKL-α 1×10^{-8}					TKL-δ 5×10^{-7}		
	TKM-α 2×10^{-8}					TKM-δ 7×10^{-7}		
Sum T	4×10^{-7}	2×10^{-8}	1×10^{-7}	3×10^{-7}	7×10^{-6}	9×10^{-6}		
Minor leak in pressurizer during transients TS_2	T_1S_2'IC-α 2×10^{-8}	T_1S_2'IG-β_1 1.5×10^{-8}	T_1S_2'IC-β_2 1×10^{-8}	T_1S_2'ICE-β_3 8×10^{-8}	T_1S_2'ICE-η 1.5×10^{-6}	T_1S_2'IC-δ 4×10^{-7}		
	T_1S_2'ICE-α 7×10^{-8}	T_1S_2'ICE-β_1 1×10^{-9}	T_1S_2'ICE-β_2 1×10^{-7}	T_1S_2'IJCE-β_3 1×10^{-8}	T_1S_2'IJCE-η 4×10^{-7}	T_1S_2'ICE-δ 1×10^{-6}		
	T_1S_2'IJ-α 5×10^{-9}	T_1S_2'IB-β_1 8×10^{-9}	T_1S_2'IJCE-β_2 1×10^{-8}			T_1S_2'IJ-δ 2×10^{-7}		
	T_1S_2'IJCE-α 2×10^{-8}							
	T_1S_2''IC-α 2×10^{-9}	T_1S_2''ICE-β_1 5×10^{-10}	T_1S_2''ICE-β_2 4×10^{-8}	T_1S_2''ICE-β_3 7×10^{-8}	T_1S_2''ICE-η 1.5×10^{-6}	T_1S_2''ICE-δ 1×10^{-7}		
	T_1S_2''ICE-α 6×10^{-8}		T_1S_2''IJCE-β_2 1×10^{-8}	T_1S_2''IJCE-β_3 1×10^{-8}	T_1S_2''IJCE-η 5×10^{-7}	T_1S_2''IJCE-δ 2×10^{-8}		
	T_1S_2''IJ-α 2×10^{-9}							
	T_1S_2''IJCE-α 2×10^{-8}							
Sum TS_2	3×10^{-7}	2×10^{-8}	2×10^{-7}	2×10^{-7}	4×10^{-6}	4×10^{-6}		
	Sum of All Frequencies in the Individual Release Categories							
Expected value	2×10^{-6}	6×10^{-7}	6×10^{-7}	3×10^{-6}	2×10^{-5}	7×10^{-5}	1×10^{-4}	1×10^{-3}
Median (50% percentile)	4×10^{-7}	3×10^{-7}	3×10^{-7}	1×10^{-6}	9×10^{-6}	2×10^{-5}	6×10^{-5}	6×10^{-4}
Lower boundary (5% percentile)	4×10^{-8}	7×10^{-8}	8×10^{-8}	3×10^{-7}	2×10^{-6}	5×10^{-6}	9×10^{-6}	9×10^{-5}
Upper boundary (95% percentile)	7×10^{-6}	2×10^{-6}	2×10^{-6}	9×10^{-6}	7×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

The values listed are frequencies per operating year. Unless otherwise noted, we are dealing with expected or mean values.

In determining the total of all release frequencies, an amount of 10% was considered from the neighboring release categories.

Abbreviations

A Large leak in a primary coolant pipeline
B Measurement of sensing variable for startup of ECCS
C High-pressure injection
D Accumulator injection
E Low-pressure flooding injection
F Low-pressure injections with sump recirculation
G Containment integrity for emergency cooling
H Long-term decay heat removal
I Main feedwater supply and main steam heat removal
J Emergency feedwater supply and main steam heat removal
K Reactor scram
L Opening the pressure relief valve of the primary cooling system
M Closing the pressure relief valve of the primary cooling system
N Delayed feedwater supply and main steam heat removal
O Long-term feedwater supply and main steam heat removal

S_1 Moderate leak in a primary coolant pipeline
S_2 Small leak in a primary coolant pipeline
T Total of all anticipated transients requiring intervention of safety systems
T_1 Loss of failure
T_1S_2 Small leak at the pressurizer during power failure, composed of the two accidents T_1S_2' and T_1S_2''
T_2 Failure of the main feedwater supply
α Failure of the containment resulting from a steam explosion in the RPV
β_1 Large leak in the containment (no later overpressure failure)
β_2 Moderate leak in the containment (possible overpressure failure)
β_3 Small leak in the containment (overpressure failure)
δ Overpressure failure of the containment
η Failure of the equipment room annulus evacuation or of the stack filter

The expected frequency for the individual categories generally deviates from the sum of event sequence frequencies of the category given in Table 6-4. These deviations are due to the following:

- The results given have generally been rounded off to one place so that rounding errors can occur after addition.
- Only the most important event sequences are presented in the table.
- For the inclusion of uncertainties in the release calculations, in WASH-1400 a 10% overlap was taken for each release category from the neighboring release categories. This method definitely increases risk and cannot be justified. In the present study, this 10% addition was also used. However, we should point out that because of this method, several release categories have significantly greater frequencies than would result on the basis of plant system investigations. For instance, the expected value for category 7 increases by a factor of more than 100, and the expected value of category 4 increases by a factor of four.

Event sequences presented in Table 6-4 are discussed below. Corresponding to the structure of the table, the different initiating events are discussed separately. The dominant event sequences are of primary interest.

An event sequence is generally characterized by three different elements. These are:

- the initiating event
- the specific combination of system functions whose failure is assumed
- the containment failure mode.

The frequency of the event sequences results from the frequency of initiating events, the probability of system failure, and the probability of containment failure.

Chapter 5 presents a detailed discussion of the frequency of core melt. Therefore, evaluation of the entire event sequence including containment failure is of primary interest below.

The conditional probabilities for the various containment failure modes were determined in different ways:

- For the probability that a steam explosion occurs for a core melt accident which then results in failure of the RPV and early failure of the containment (failure mode α), the value presented in WASH-1400 was used (Median = 10^{-2} with an uncertainty factor of ten; from this follows the expected value of $w\delta = 2.7 \times 10^{-2}$).

- The probabilities for the occurrence of a leak in the containment (failure modes β_1, β_2, and β_3) or for failure of annulus exhaust air handling or of the filter (failure mode η), were determined by means of fault tree analyses. It was considered here that these probabilities generally depend both on the initiating event as well as the failure of system functions.

- The results of Section 6.3 show that for core melt accidents, overpressure failure of the containment (failure mode δ) occurs provided it has not already failed in some other manner. The probability w_δ therefore results from the difference between unity and the sum of probabilities of the failure modes of the containment for an otherwise corresponding event sequence.

Large and Medium Leak

In Table 6-4, column "summation A," all frequencies are added for event sequences with the initiating event "large leak" that contribute to the individual release categories 1-8. The same has been done for contributions for event sequences with initiating event "medium leak," which are summarized in column "summation S_1." If we compare the named values with the attendant total frequencies in the individual release categories, then we obtain amounts less than or equal to 3%, and often far below 1%. The sole exception is category 2 for a medium leak with a frequency of 6×10^{-8}. This fraction makes up 10% of the total result of category 2. The contributions to this value come from event sequences $S_1G\ \beta_1$ and $S_1B\ \beta_1$. In the former case, we are dealing with event sequences where the emergency cooling fails due to leaks from the containment into the annulus (failure of the "containment integrity for emergency cooling" function, (see Section 5.2.1.2). Naturally the containment leak tightness has been lost. In the second case, with the failure of system function B (instrumentation for emergency cooling startup signal), the control for the building isolation is also lost. The descriptions of these two cases are analogous to the event sequence with Large Leak AG β_1 and AB β_1; however the corresponding frequencies are smaller.

The LOCAs controlled by the emergency cooling system are contained in release categories 7 and 8. Accordingly, there can at most be cladding damage but the core remains intact.

Small Leak

As we can see from column "summation S_2" in Table 6-4, the accident "small leak in a primary coolant line" has a frequency of 1.5 X 10^{-6} in category 1 and 5.5 X 10^{-5} in category 6. These are dominant in the overall results (frequencies) of the attendant categories. Event sequences $S_2IJ-\alpha$ and $S_2IJ-\delta$, i.e., potential error when shutting down the plant manually (see Section 5.2.1.3), are important for these release categories. Even in categories 3 and 4, event sequences $S_2IJ-\beta_2$ and $S_2IJ-\beta_3$ contribute significantly to the summed frequencies determined for a small leak in these categories. Failure of all reactor protection signals for two of the four redundant subsystems plays a significant role here.

The event sequences of the "small leak" initiating event make up one-third of the total result in category 2 because of their frequency 2 X 10^{-7}. The important participating events $S_2IG- \beta_1$ and $S_2IB- \beta_1$ are described like the corresponding cases for large and medium leak. The former deals with loss-of-containment integrity accident; the latter deals with common-cause failure of the instrumentation needed for emergency cooling and containment isolation.

Transients

The influence of transients on the total result in the individual release categories is greatest in category 5 (30%). In category 1, we find a 20% contribution, followed by categories 3, 4, and 6, with values between 10 and 15%. The primary contributors to this event are the event sequences T_1IR... and T_1IJQ... arising from a power failure. In the first case, we are dealing with the failure of "long-term feedwater supply and main steam relief" (function R). In the second case, the function "emergency feedwater supply and main steam release" fails, and timely feedwater supply from unit A also fails. The overwhelming influence of common cause failures of the emergency diesel generators is important here. For example, in category 5 a common cause failure of the emergency diesel generators leads to failure of the annulus exhaust air handling system. In release categories 3 and 4, the common cause failure of the diesel system, together with two independent failures, lead to core melt and failure of the building leak tightness. Details on these accident sequences are found in Section 5.2.2.2.

Small Leak in the Pressurizer

The greatest contribution from this accident is found in category 3 (about 30%). In categories 1, 4 and 5 contributions are between 10 and 20%. The event sequences T_1S_2ICE... and T_1S_2ICE... are decisive for all these results. Since the accident under discussion develops from a power failure, the overwhelming dependence on operationality of the emergency diesel generator is evident. As in the power failure case, common cause failures of the emergency diesel generators in release categories 3 and 4, together with two other independent failures, play an important role (see Section 5.2.4). Common cause failure of the emergency diesel generators contributes strongly in almost all cases in release category 5.

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Section 7

ACCIDENT CONSEQUENCE MODEL

7.1 OVERVIEW

Section 7 describes the mathematic models and input data used in the study for the determination of radiological consequences and potential health effects to persons after a nuclear power plant accident in the Federal Republic of Germany. The detailed presentation and documentation are found in the appendix.

Figure 7-1 shows a schematic illustration of the entire accident consequence model. The model was divided into the following parts:

- atmospheric dispersion and deposition
- calculation of dose
- protective actions and countermeasures
- health effects.

Results or interim results are presented in the rectangular boxes; input data and parameters appear in boxes with rounded sides; and frequency or probability numbers are placed in diamond-shaped rhombuses. This schematic illustration is followed by explanations.

The starting points of all calculations are the quantity and nature of radioactive material that can be released in the course of a reactor accident from the containment to the atmosphere. The release spectrum for the radioactive substances was divided into eight so-called release categories (see Chapter 6). In contrast to WASH-1400, we considered pressurized water reactors only, based on the representative type of modern German PWR with 1300 MWe power output. In the study it was assumed that all nuclear power plants with a power output of at least 600 MWe, which were in operation or under construction on July 1, 1977, or for other reactors for which licensing applications had been received by the reference date, were equipped with this particular pressurized water reactor design (see Chapter 1).

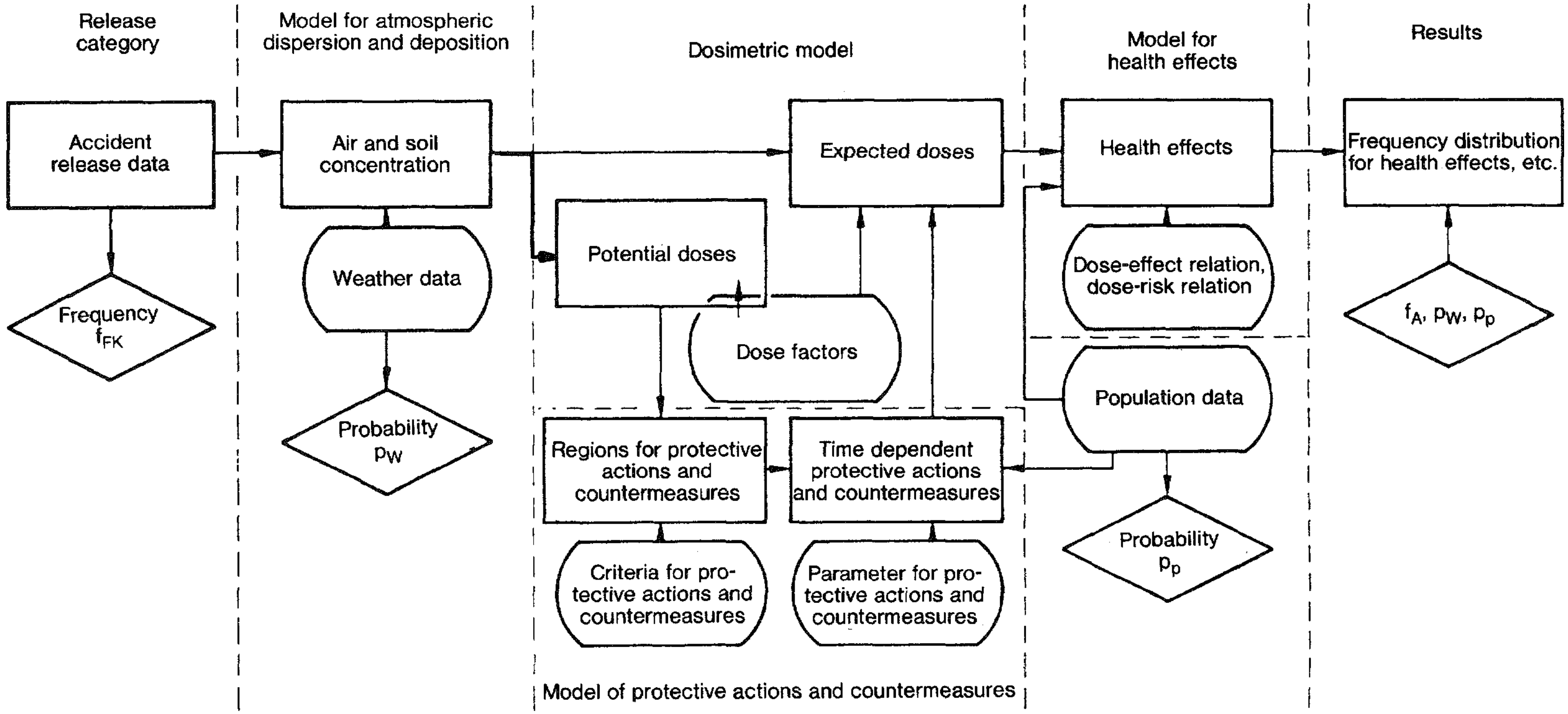


Figure 7-1. Schematic diagram of consequence model

The release categories are characterized by time history, release fraction of important radionuclides, and frequency of these releases. These characteristic data were obtained from investigations described in the appendices.

The model for atmospheric dispersion and deposition provides the chronological and spatial distribution of the radioactivity concentrations in the air and ground contamination in the environment of the plant.

The model considers:

1. radioactive plume rise due to thermal energy (sensible and latent heat)

2. the influence of the building wake on dilution and rise of the radioactive plume

3. the time-dependent turbulence (stability category) of the atmosphere, wind direction, and precipitation (the model does not yet include changes in wind direction in the course of dispersion; this is intended for phase B of the study)

4. the decay of radioactivity as a function of time after the accident

5. depletion of the radioactive plume as a result of dry and wet deposition.

From the radioactive concentrations in the air and from the ground contamination, the dosimetric model initially calculates potential doses, for which the various protective actions and countermeasures are geared. Thereafter, the doses resulting after implementation of these measures are determined. The following exposure pathways are considered:

1. external exposure from the passing radioactive plume

2. external exposure from the radioactive material deposited on the ground

3. internal exposure from the radioactive material inhaled from the air, divided into:

   --inhalation of airborne radionuclides from the radioactive plume, and

   --inhalation of resuspended radionuclides which had previously been deposited

4. internal exposure from the radioactive material incorporated from foodstuffs.

Doses were calculated for the whole body and for the following organs: bone marrow, bone surfaces, lungs, thyroid, breast, gonads.

In calculating the expected doses, the following types of protective actions and countermeasures were considered:

- sheltering in buildings
- evacuation and relocation
- decontamination
- restrictions on consumption of agricultural products.

The criteria for the onset of these measures and their time history are described in Section 7.4.

The consequences to the affected population are determined from the calculated doses, using the model for calculation of health effects. These health effects are:

- somatic early injury (death by acute radiation syndrome)
- somatic latent injury (death by leukemia or cancer)
- genetic effects (genetically significant dose).

Risk from 25 commercial reactor units at 19 sites was calculated as follows: 19 sites were assigned to four meteorologically representative regions. Their characteristic parameters were used to calculate atmospheric dispersion in the assigned regions. Selection of representative data for the four regions is described in detail in the appendix. Radioactivity concentrations in the air and on the ground are calculated with respect to location up to 540 km (a) from the point of accident. The local doses thereby calculated were applied to the population distributions after considering countermeasures that reduce dose as a function of time and location. Up to 80 km (50 miles in WASH-1400), site-specific population data were used. Beyond 80 km, average population densities were used. The 540 km range corresponds to the average radius of Central Europe. At distances greater than this, topographical features (ocean, mountains, etc.) may profoundly affect population densities. The effect of radioactive material carried beyond these boundaries is taken into account by the deposition of radioactive aerosols and radioactive iodine over an area which in our study--analogous to WASH-1400--corresponds to the total remaining area of Europe.

Repetition of this calculation for all accident categories and 115 representative weather sequences at each of the 19 sites gave the matrix of individual results. The somatic early and latent health effects and genetically significant doses calculated from these results are illustrated in complementary cumulative distribution functions. In addition, the expected values for early and latent health effects and for the genetically significant doses are given. Values for latent somatic health effects were compared with expected incidence for leukemia and cancer caused by natural and civilization-induced factors.

The results are presented in Chapter 8, together with preliminary sensitivity analyses and estimations of uncertainty.

7.2 ATMOSPHERIC DISPERSION AND DEPOSITION

7.2.1 Atmospheric Dispersion Model

As soon as radioactive substances are released from the outer containment structure or from the exhaust stack to the atmosphere, they are subject to atmospheric dispersion. The activity plume moves from the site at the same speed as the wind. The radioactivity concentration decreases continuously, primarily dependent on atmospheric turbulence and the topography of the terrain beneath the plume.

The various models that describe atmospheric dispersion range from the mathematically complex to simple "box models." For practical purposes, a special solution of the diffusion equation has proven useful. The radioactivity distribution in the plume cross-section transverse to the wind direction is approximated in this solution by a Gaussian distribution. This so-called Gaussian dispersion model is used in this study.

The model has been confirmed by experiments as reliable at ranges up to 20 km. Within this range, early health effects are possible for high-release rates. In order to calculate the magnitude of such effects, reliable information on the dose distribution is necessary because of the nonlinear dose-effect correlation (see Section 7.5). Even at ranges beyond 20 km the same model is used. On one hand, some experimental results exist that do not contradict the applicability of the model even at longer ranges. On the other hand, accurate knowledge of dose distribution is not crucial in order for the result--the number of latent fatalities--to be determined (see Section 7.2.2).

The standard deviation of the Gaussian bell-shaped curve is given by the horizontal and vertical dispersion parameter $\sigma_y(x)$ or $\sigma_z(x)$. The dispersion parameters used were determined by dispersion experiments at the Karlsruhe Nuclear Research Center (KfK) (1). They represent terrain with a rough surface (forest, population centers), also demonstrated by other experiments on similar topographic regions (2,3). These parameters are modified accordingly for flat terrain (North German Plain).

The range of vertical turbulence exchange in the atmosphere is usually bounded on the top by an inversion layer. The height of the mixing layer is closely connected to the top of the lower planetary boundary layer. Simply speaking, this is the lowest layer of the atmosphere in which the air masses are thoroughly mixed vertically. Thus, the vertical dispersion parameter is kept constant once the top of the mixing layer is reached. After that point, a reduction in the radioactivity concentration takes place only through the horizontal dispersion.

The rise of the radioactive plume due to released thermal energy is calculated primarily by the equations of Briggs (4,5). Nester (6) improved on these formulae by including the effect of the building's wake. The influence of decay energy on the rise is neglected because of its minor contribution (7). The plume rise is limited by the maximum value of $\sigma_z(x)$. The original radioactivity content of the plume is reduced by dry and wet deposition and by radioactive decay.

The meteorological data used to calculate radioactivity concentrations in air and on the ground--namely wind speed, stability category, and information on precipitation--are adapted to real weather sequences measured hourly. Thereby it was assumed that meteorological parameters measured at the site assume the same values at all distances. To be sure, this is normally not the case. This type of assumption seems justified, however, because more than 100 multi-hour lasting weather sequences were tracked (this figure is needed to adequately characterize all possible weather sequences), and because the uncertainties in the calculations tend to be self-compensating in the summarized result when using individual weather sequences.

Each of the 115 weather sequences gives a radioactivity concentration and contamination field. Each field extends from the point of emission in the assumed dispersion direction. In accordance with WASH-1400, it was assumed that any wind direction is equally probable. The assumption of equal wind direction probability is taken into account by dividing the wind rose into 36 ten-degree sectors and by assigning the same probability to the centerline of each of these sectors.

7.2.2 Limitations of the Model

Linear transport of the radioactive plume is assumed. Larger fluctuations in wind direction are not considered, leading to a conservative estimation of dose. This assumption corresponds to the model in WASH-1400. In this manner, the dose is usually overestimated for a release lasting several hours. This includes a frequent excess of the threshold dose for early health effects and, accordingly, an overestimation of early injuries. This effect is only slightly compensated by the fact that at several points where the probability of early health effects is nearly unity due to the high dose value, a further increase in dose does not increase the early injuries.

At long range (more than 20 km) only latent health effects occur, according to the calculated results. The magnitude of such health effects is, at first approximation, independent of whether a given quantity of radioactivity is distributed over a narrow or broad sector under the same boundary conditions because of the linear dose-effect relationship. Beyond an 80 km distance, this is strictly true because the population is assumed to be uniformly distributed in an azimuthal direction.

This model of straight line dispersion is applied at ranges up to 540 km, as was done in WASH-1400. The area of this circle approximately corresponds to the area of Central Europe. Beyond this range dispersion conditions usually differ significantly (oceans, mountains).

The activity inventory in the plume has not decreased to the point that--even after dispersing out to 540 km--it needs not be considered any further. For an average of all weather sequences, the radioactive plume passes this "boundary" with about 40% of its original inventory of long-lived radionuclides--depending on the selected deposition coefficients. Radioactive gases and aerosols contribute in various ways to the total radiation exposure along the path of the passing radioactive plume. At the first pass, the radioactive noble gases contribute an insignificant (less than 0.1%) amount to latent health effects through direct external exposure. Later on, this radioactivity is globally dispersed. The dose to the population can be neglected by comparison to the other radionuclides. Radioactive iodine and aerosol as compared to noble gases, however, lead to a significantly greater radiation exposure during the first pass of the radioactive plume as a result of deposition on the ground. Up to a distance of 2500 km, on the average 99% is deposited. Therefore, the effect of radioactive iodine and aerosol activity remaining after 540 km is taken into consideration by assuming they are deposited on an annular ring of 2500 km outer radius--this area corresponds to the remainder of Europe, including water areas. This method corresponds to that in WASH-1400.

7.2.3 Meteorological Regions

For a realistic modeling of changing dispersion characteristics in the dispersion model described above, the hourly values of wind speed, dispersion category, and, if possible, precipitation intensity must be available for at least one year. Such detailed measurements are available only from a limited number of stations--in contrast to widely available dispersion statistics. The nineteen nuclear power plant sites (see Chapter 1) were assigned to four meteorological regions, within each of which one set of meteorological data was used. The definition of a region resulted from the requirements of similar dispersion conditions and geographic similarity. The four regions are:

- North German Plain
- Upper Rhine Plain
- South German Plateau
- Valleys.

These regions and meteorological stations belonging to them are discussed in the appendix.

7.3 DOSIMETRIC MODEL

The radiation dose to the affected population is calculated on the basis of spatial-dependent and time-dependent radioactive concentration in the air and on the ground determined by the model of atmospheric dispersion and deposition processes. The energy dose D for special organs and the whole body is specified as the characteristic quantities for radiation exposure. These quantities express the radiation energy absorbed per organ mass. The unit of measurement is the "rad" (b).

The ways by which the radiation of the released radioactive material reaches man is called the exposure pathway. From the release of radionuclides into the atmosphere, there are the following primary exposure pathways:

- external irradiation due to a passing radioactive plume

- external irradiation due to radioactivity deposited on the ground

- internal irradiation due to radioactivity inhaled by normal respiration, divided into:

  --inhalation of airborne radionuclides from the radioactive plume

  --inhalation of resuspended radionuclides that have been deposited on the ground

  --internal irradiation due to radioactivity ingested with foodstuffs.

With regard to external irradiation, the spatial distribution of radioactivity in the air and on the ground primarily determines the energy absorbed by the body, and this is almost independent of the organ. With respect to internal irradiation, the metabolic processes primarily determine the energy exposure in the individual organs, in addition to the spatial distribution of radioactivity in the air and on the ground and in addition to the incorporation (type and quantity of absorption in the body).

Calculation of radiation exposure is performed for the following organs (c) which are important for determining health effects:

- bone marrow
- bone surface
- lung
- thyroid
- breast.

The determination of injury to the other organs is based on the radiation exposure of the whole body.

To determine genetically significant dose, the radiation exposure of the testes and ovaries is used.

Since the criteria for countermeasures of relocation, decontamination, and temporary restriction on consumption of local agricultural products (see Section 7.4) are geared to predicted doses, doses are calculated in the first step for persons who would continuously remain outdoors and thus be susceptible to the relevant exposure pathways. These are called "potential doses" below. In the second step, doses to persons taking protective action and countermeasures are calculated (d). These are called "expected doses" below. They are used in the consequence model (see Section 7.5) in which the magnitude of early somatic fatalities (deaths due to acute radiation syndrome) is determined from the dose-effect relationships. The magnitude of somatic latent fatalities (premature death due to leukemia or cancer) is determined with regard to the dose-risk coefficient relationship. The genetic effects are expressed as the genetically significant doses.

The following doses are calculated in detail:

Potential doses:

To establish proper countermeasures to minimize early health effects, we calculate:

- the bone marrow dose due to external radiation from the radioactivity deposited on the ground during the first 7 days

To establish countermeasures to reduce latent somatic health effects and genetic effects, we calculate:

- the whole body dose due to external irradiation from the radioactivity deposited on the ground during the first thirty years

- the bone marrow, thyroid, and whole body doses due to internal irradiation from radioactivity incorporated with foodstuffs during the first fifty years.

Expected Doses:

To determine early health effects we calculate:

- short-term bone marrow dose, composed of:

  --external irradiation due to activity in the passing plume

  --external irradiation due to radioactivity deposited on the ground within the first seven days

  --internal irradiation due to inhaled radioactivity from the plume in the first thirty days.

To determine latent somatic health effects we calculate doses for the following organs:

- bone marrow
- bone surface
- lung
- thyroid
- breast
- remainder of the body.

The doses are summed for all exposure pathways resulting from the release of radionuclides to the atmosphere. The radiation exposure of the directly affected population, as well as of persons born after the accident, is taken into account.

To determine the genetic effects we calculate:

- The genetically significant doses summed over all exposure pathways resulting from release of radionuclides to the atmosphere. This is calculated from the doses to the testes and ovaries. The exposure of both the immediately affected population and of persons born after the accident is considered.

By integration of the spatial-dependent individual doses multiplied by the number of affected persons in the particular region, we obtain the resulting expected population dose (e), i.e., the sum of all expected individual doses.

7.4 MODEL FOR PROTECTIVE ACTIONS AND COUNTERMEASURES

7.4.1 Principles of the Model

For high-technology facilities with large hazard potential like industrial plants, nuclear power plants, storage and transport containers for poisonous or explosive substances, dams, transportation facilities, etc., the principle applies that safety measures must first be taken in the plant itself. This is done, for instance, by technical safety features; by strict supervision in planning, construction, and operation of such facilities; and by safeguards against the acts of third parties.

As additional measures, the individual States and their subordinate authorities in the FRG--within the framework of their general obligations to protect the public against natural catastrophies--prepare emergency response plans for the environment around plants of the above types, that is, also for nuclear power plants. Official emergency response plans for nuclear power plants are derived from the "basic recommendations for emergency protection in the environment of nuclear power plants" (8), which were jointly published by the State Committee on Nuclear Energy and the Interior Departments of the FRG States. The following sections are discussed in the basic recommendations:

- types of protective actions and countermeasures
- classification of the environment around the nuclear plant
- notification of authorities and population
- timing of measures at the official level
- radiological criteria for planning protective actions and countermeasures.

Some of the planned measures are:

- advice to the affected population not to stay outdoors but to seek shelter (preferably in cellars)
- distribution of iodine tablets
- evacuation
- prohibition of heavily contaminated areas and water sources
- distribution of noncontaminated foodstuffs and drinking water to the public
- decontamination.

A brief and generally understandable summary of the basic recommendations, coupled with procedural guidelines and information on protection measures for the public, are contained in the "Guidelines for Instructing the Public on Emergency Response Planning in the Environment of Nuclear Plants" (9), which were issued by the Standing Committee of the Interior Ministers and Interior Senators of the States. A series of brochures (10,11,12) are in turn based on these guidelines; the brochures are distributed by the authorities to the public.

In the model of protective actions and countermeasures, the basic recommendations (8) are considered. The type and urgency of protective actions and countermeasures differ from place to place. This leads to a matrix of regions, measures, and time sequences explained in the next section. The time sequences incorporate implementation and measurement times, as well as time requirements for the arrival of personnel, detection equipment, and transport means. Transportation facilities important for evacuation are considered crudely, with a distinction being made between urban, suburban, or rural sites. Several auxiliary models, e.g., for ingestion, were adopted without change from WASH-1400.

7.4.2 Structure of the Model

In the basic recommendation (8) and guidelines (9), we note that the population is requested to stay indoors and listen to radio or TV during nuclear accidents. As additional measures, if necessary, evacuation, decontamination, or temporary restriction on consumption of locally produced agricultural products are provided for in certain regions. If the radiation dose to the population so requires, and if the effects of the accident can thereby be mitigated, the emergency response authority may require additional measures like relocation within a few days or weeks (rapid relocation) or relocation in the course of several months (resettlement). Thus, for the present study, the list of protective actions and countermeasures is as follows:

1. going indoors
2. evacuation
3. fast relocation
4. resettlement
5. decontamination
6. temporary restriction on the consumption of local agricultural products.

Table 7-1 provides an overview of the application of protective actions and countermeasures with respect to region and time. The selected criteria for implementing measures (2), (3), and (4) are given in Sections 7.4.3 and 7.3.4. Consistent with WASH-1400, the consumption of iodine tablets to mitigate thyroid dose is not taken into consideration.

A detailed description of the model of protective actions and countermeasures is found in the appendix.

Table 7-1. Correlation of protective actions and countermeasures with regions and times

Measure \ Region, Time, Purpose	A	B_1	B_2	C	D_1	D_2	Time history	Main purpose of the measure is the avoidance of
Information and preparation (initiation phase)							0 - 2 hours	
Residence time in sheltered rooms	○	○					After two hours	Early injury due to plume or ground radiation
Evacuation	○						2 - 14 hours	Early injury due to ground radiation
Fast relocation		○	○				After fourteen hours	
Resettlement				○			30 days to one year	Latent health effects due to ground radiation
Decontamination					○		After 30 days	
Restrictions on consumption of agricultural products	○	○	○	○	○	○	In accordance with criteria	Latent health effects due to internal irradiation after ingestion
Delayed decontamination	○	○	○	○	○		Before return of the population	Latent health effects due to ground radiation

7.4.3 Regions

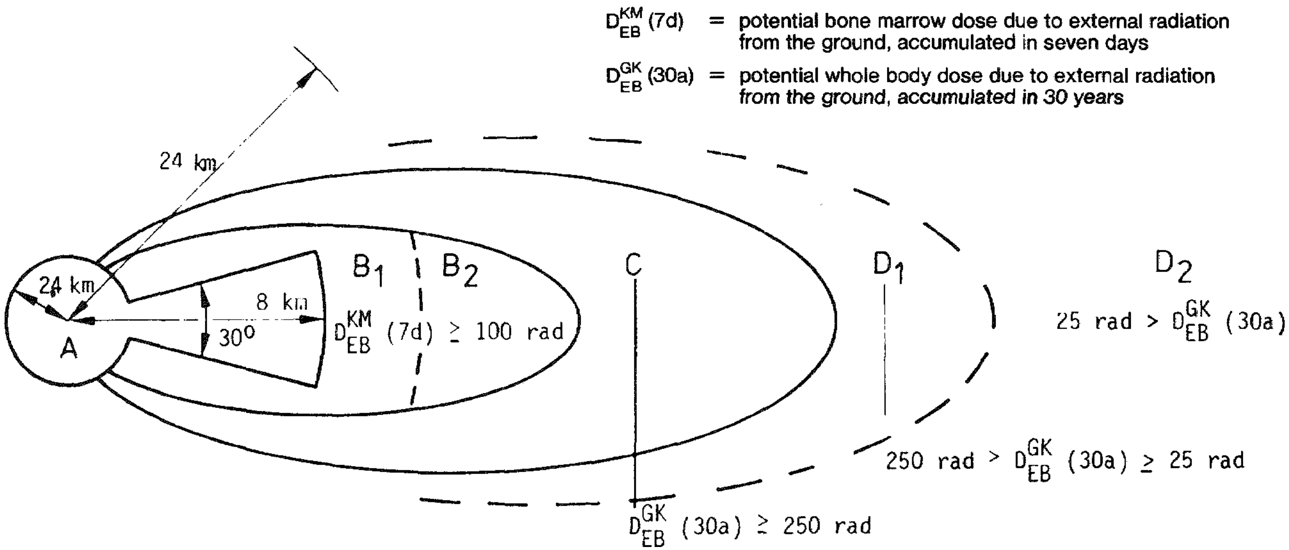
Nuclear power plant operators and the responsible authorities in the FRG have taken proper precautions so that dose distributions in the environment can be measured or estimated during and after a release of radioactivity. Depending on the results, the emergency response authority initiates the appropriate protective actions and countermeasures by applying the criteria defined in the basic recommendations.

The present study proceeds in an analogous manner. First, the spatial distribution of dose in the environment (called potential dose below) is calculated. Thereafter, if the doses exceed prescribed reference values, selected lines of equal dose (isopleths) are used to divide the affected area into five regions (B_1, B_2, C, D_1, and D_2) requiring the implementation of different measures (see Figure 7-2).

In addition, another region (A) is added. This region encompasses the immediate vicinity of the nuclear power plant where high doses can occur and where there may be no time available for radioactivity and dose measurements and their effects for large radioactivity releases and unfavorable dispersion conditions. The region, defined by a prescribed angle and ranges, consists of a sector in the propagation direction ($r = 8$ km, $\gamma = 30$ degrees) together with a complete circle of 2.4 km radius (f). Inclusion of a full circle is necessary because eddy currents and the diffusion processes in the immediate vicinity of the nuclear power plant can cause radioactive transport over short distances in all directions. Also the radiation emission from the radioactive plume has a limited range in all directions. In conformance with the basic recommendations (8), it can be assumed that preparatory emergency and evacuation plans exist for region A.

After the regions are delineated, the doses are recalculated under consideration of protective actions and countermeasures, and from these personnel health effects are determined.

Whereas the occurrence of early health effects depends on the magnitude of bone marrow dose, latent health effects are approximately proportional to the whole body dose. Therefore, in the delineated region boundaries we find lines of equal potential bone marrow dose D_{EB}^{KM}, as well as lines of equal potential whole body dose D_{EB}^{GK}. The region boundaries are selected so that early health effects can occur only in regions A, B_1 and B_2.



Region A, defined by angle and range, is the same for all release categories. For this region, the existence of prepared evacuation plans is assumed.

Regions B_1, B_2, C, and D_1 are defined by isodoses lines. Their consideration thus depends on the type of release and the prevailing weather situation.

In the majority of cases, the doses outside of region A remained below the defined values for regions B_1 and B_2. In such cases, these regions and the attendant countermeasures were omitted. In addition, regions C and D_1were also omitted in many cases.

Figure 7-2. Regions wherein emergency countermeasures are taken (schematic). For explanation, see Table 7-2.

A characterization of all regions is summarized in Table 7-2. The attendant measures are discussed in the next section.

The region boundaries described above depend on the type of release and the particular weather history. Eight release categories and 115 weather histories (per site region) are considered. Thus, we have 920 cases per site region with different region boundaries. In the next section we shall show that the density of the affected population, which differs for each site (19) and for each wind direction (36), affects the sequence of countermeasures. Therefore, calculation of consequences performed under consideration of protective actions and countermeasures encompasses 629,280 different combinations of reactor site, release category, weather history, and wind direction (= population distribution).

7.4.4 Measures

Initial Phase

After recognition of the imminent release, an implementation phase of two hours is assumed during which local and regional decision makers are informed, staffs are formed, and the standard signal for catastrophic accidents is given (one minute siren wail). As a result of the siren signal (and loud speaker trucks), the population in region A and possibly B_1 is requested to go indoors and to listen to radio or TV. If during these two hours, individual subregions are affected by radioactivity, an average shielding factor is used that corresponds to mixed dwell times in large and small buildings and in the open. It is assumed that 3% of the population will remain outdoors. Protective actions and countermeasures taken at a time later than two hours are specific to the region and will be illustrated below separately for each region (see also Table 7-2).

Protective Actions and Countermeasures in Region A

The main purpose of measures in region A is to reduce or entirely avoid the number of acute health effects. As a result of previous licensing practices, this region is generally agricultural. The study assumes that after two hours, about two-thirds (65%) of the population will have found larger buildings or cellars in small buildings and that they will remain at protected places (i.e., away from windows and doors), whereas about 1/3 (32%) have

Table 7-2. Model of protective actions and countermeasures

Regions	Region definition		Preventive measures (regardless of accident type and weather situation, except wind direction)	Dose-dependent measures*†
	Delineation through angle and range	Delineation through lines of equal potential dose		
A	$r \leq 2{,}4$ km $\sphericalangle$ given such as $2{,}4<r<8$ km $\sphericalangle = 30°$	--	Go indoors after 2 hours. Remain indoors up to 8 hours. Thereafter, evacuation, preparation, and driving time 1.5 hours.	--
B_1	$r \leq 24$ km	D_{EB}^{KM} (7d) ≥ 100 rad (If not belonging to A)	Go indoors after 2 hours. Remain indoors at least 14 hours.	Remain indoors. Rapid relocation after 14 hours.
B_2	$r > 24$ km		--	Indoors and outdoors. Then, rapid relocation after 14 hours.
C	--	D_{EB}^{GK} (30a) ≥ 250 rad	--	Indoors and outdoors. Thereafter, relocation beginning after 30 days.
D_1	--	250 rad $> D_{EB}^{GK}$ (30a) ≥ 25 rad	--	Normal activity at all times. Decontamination such that D_{EB}^{GK} (30a) = 25 rad in the entire region.
D_2	--	25 rad $> D_{EB}^{GK}$ (30a)	--	--

*In addition, the dose-dependent restriction on consumption of local agricultural produce.

†Return of the population occurs when D_{EB}^{GK} (30a) $\leq$ 25 rad.

retreated to small buildings, but not into cellars. It is assumed that 3% percent of the population remains outdoors in spite of the warning. This is the same population group assumed to remain outdoors during the implementation phase.

When staying in houses or cellars, the radiation dose from the air or ground is less than outdoors because of the greater distance to the radioactive substances and the shielding effect of the walls and floor (cellar).

Since the largest doses occur in region A for nearly all weather situations, and because no time is available for performing and evaluating the measurements, the study assumes that the emergency response authority will initiate evacuation for any Class 9 event. In addition, it is conservatively assumed that 12 hours are needed to evacuate this relatively small region of 33 square kilometers; i.e., between the second and fourteenth hour the inhabitants use cars or other transportation to flee the potential danger zone for acute health effects within a driving time of 1.5 hours. Driving time is treated as unshielded outdoor exposure. During the transport time, the local dose at the residential site, which generally corresponds to the maximum dose to which a person is exposed, is assumed.

Return of the population is anticipated when ground contamination has dropped due to radioactive decay, weather effects, and decontamination to such a level that the potential total body dosage D_{EB}^{GK} (30 years) does not exceed 25 rad. Latent health effects caused by residual ground contamination are taken into account in the study to the extent that it affects persons born after the accident. The type and scope of decontamination measures are described in the appendix.

Protective Actions and Countermeasures in Regions B_1 and B_2

(D_{EB}^{KM} (7d) $\geqq$ 100 rad)

In the majority of cases there is no region B_1 and B_2. But if the radioactive release and deposition during an accident are so great that a region B_1 can be defined (see Table 7-2), early health effects are also possible there. The study thus provides, as in region A, for protective actions and countermeasures, including relocation of population, whereby the measure "staying indoors" is the same in both regions (A and B_1).

In order to keep the radiation dose to the population in region B_1 as small as possible, the emergency response authority seeks to swiftly implement the evacuation in the form of a fast relocation.

The present study conservatively assumes, however, that emergency plans for this region do not exist and that fast relocation can be organized only after the accident occurs.

It is therefore assumed that fast relocation of citizens in region B_1 begins at the earliest 14 hours after the accident, i.e., after region A has already been evacuated. The long preparation time permits authorities to inform the population by radio and TV and allows time for task personnel and transportation means to arrive.

Driving times during fast relocation are needed to calculate total dose. For this purpose, the study distinguishes between urban areas, suburban areas, and rural areas for region B_1 (see appendix). By using a computer program to simulate population movements, a driving time spectrum is made for each of the three types of areas. Accordingly, these spectra are approximated by three driving times so that each driving time will include 1/3 of the particular population.

The driving times are treated in the dose calculation as outdoor exposure at the place of residence. A uniform preparation time of 0.25 h is added to the driving times for the total exposure from the ground. Spatial and time sequencing of fast relocation are not taken into consideration. After the preparation and driving time elapses, the inhabitants of region B_1 left the danger zone and receive no additional dose which can contribute to early health effects. Return of the population and calculation of latent health effects are handled as in region A.

Release category 1, occurring in connection with about 10% of the weather situations, and release category 2, occurring in connection with about 4% of the weather situations, lead to a 100 rad isopleth (potential bone marrow dosage due to external irradiation from the ground, accumulated over seven days), which encompasses areas whose distance to the reactor is more than

24 km. Since, in general, no emergency response planning is required for these regions, conclusions on the effectiveness of protective actions and countermeasures are uncertain. Therefore, the study pessimistically assumes that inhabitants of subregion B_2 continue their normal activities until a fast relocation is ordered. Fast relocation in region B_2 is the same as in region B_1.

Countermeasures in Regions C, D_1, and D_2

In regions C, D_1, and D_2, no dose is incurred that leads to early health effects. The objective of countermeasures in these regions is to keep latent health effects as low as possible. This is achieved by relocation, decontamination, and temporary restriction on consumption of local agricultural products produced in a specific subregion. This study assumes that potential doses due to radiation from the ground for the whole body (D_{EB}^{GK}) are below 25 rad in 30 years; beyond the restrictions on consumption of agricultural products, no other countermeasures are applied (region D_2). In regions with higher radiation levels, decontamination is needed (region D_1). However, if contamination is so high that the value (D_{EB}^{GK}) (30 years) = 25 rad can be attained only by means of a decontamination factor (_g_) greater than 10 (region C), then in the model, the decontamination is delayed, and the population is temporarily relocated.

In the present study, relocation in region C begins after 30 days. Proceeding from the areas next to the reactor, relocation broadens to greater ranges. In the entire period from release to relocation, a shielding factor representative of mixed residences is used. As soon as the potential dose from the ground due to radioactive decay and weather effects in areas of region C falls below a value of 250 rad in 30 years, decontamination (factor DF = 10) and return of the population are assumed. The total dose of the returning fraction of the population in region C--as in the other regions--is determined from the amounts received before relocation and after return.

Contamination in region D_1 is so low that potential whole body dose, summed over 30 years, can be reduced by decontamination to a value below 25 rad. Therefore, the present study assumes that population movements do not occur and that the inhabitants of this region go about their normal activities at all times. It is also assumed that decontamination in the entire region D_1 does not take effect until after 30 days.

In the entire region D_2, potential local doses D_{EB}^{GK} (30 years) are below 25 rad in accordance with the definition. The only countermeasures considered in the present study are restrictions on consumption of agricultural products. Therefore, a preliminary model was adopted without change from WASH-1400.

Table 7-2 gives an overview of all countermeasures (in addition to restrictions on consumption of agricultural products) in regions A, B_1, B_2, C, D_1, and D_2. Additional details can be found in the appendix.

7.5 MODEL TO DETERMINE HEALTH EFFECTS DUE TO RADIATION

The biological mechanism of radiation acting on an organism and the resulting potential types of radiation health effects will be discussed below. Next, models for determining the risk of radiation health effects are described as a function of dose used in this study to analyze the consequences of reactor accidents.

7.5.1 Biological Effects of Radiation and Types of Radiation Health Effects

Absorption of radiation energy by a living cell or tissue initiates a chain of physical, chemical, and biological reactions that can injure the health of the affected individual; or, in the case of radiation to the gonads (testes, ovaries) injure the person's offspring. Figure 7-3 illustrates the reaction chain of biological effects in an organism and the potential types of radiation health effects. Since both cell and organism have highly effective mechanisms to repair or eliminate primary biological effects, the absorption of radiation by an organ or tissue of the body does not necessarily result in later manifestation of radiation injury. Therefore, only the possibility for radiation health effects exists, and this will be denoted as "radiation health effect risk." Radiation risks for an irradiated tissue thus denote the probability for physical ill-health.

According to Figure 7-3, we distinguish between four types of radiation health effects:

1. acute or early health effects that appear shortly after radiation (e.g., acute radiation sickness)
2. nonmalignant delayed injury recognizable years after radiation (e.g., fibrotic tissue changes, clouding of the retina, reduction in fertility)

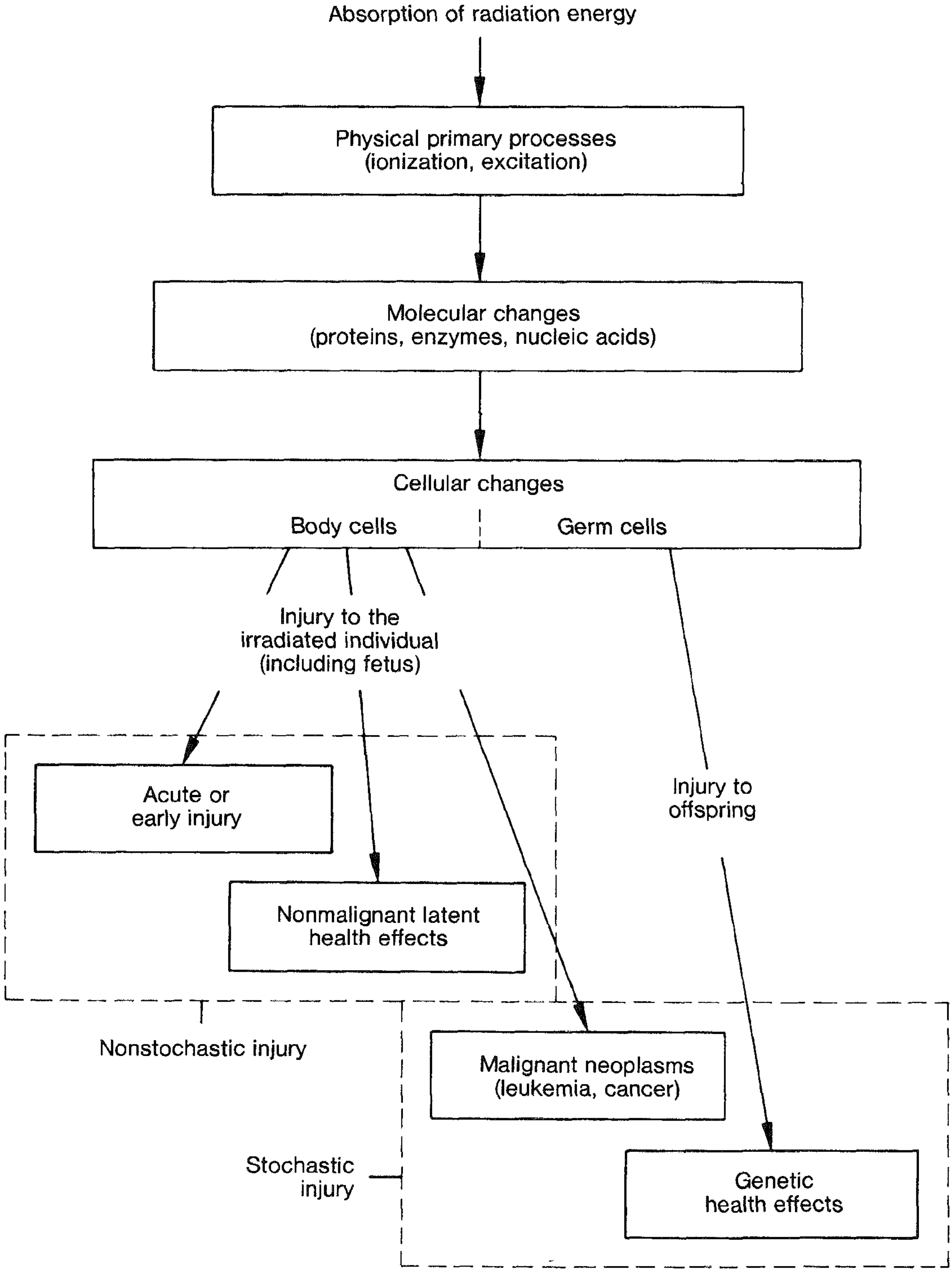


Figure 7-3. Reaction chain of biological radiation effects and types of radiation injury

3. delayed malignant health effects manifested only after a latent period lasting from years to decades (e.g., leukemia due to radiation of red cell bone marrow and tumors in other body tissues)

4. genetic effects resulting from specific mutations of reproductive cells after irradiation of the gonads (testes, ovaries) causing changes in offspring (e.g., skeletal anomalies, mental retardation, changes in eye color, hereditary illnesses).

The first three types of injury are somatic health effects that can appear in the affected individual. The fourth refers to injuries that can occur in later generations as a result of radiation to the womb.

The carcinogenic and genetic effects are designated as stochastic, i.e., chance radiation effects, since with these types of injury the probability of radiation-induced incidence, and not the level of injury, depends on radiation dose. Conversely, for acute health effects and noncarcinogenic delayed health effects, the level of injury depends on the dose: such health effects are grouped under the heading "nonstochastic effects."

The physical ill-health to an individual or to the general population due to radiation effects therefore should not be evaluated from the lumped incidence of radiation injury; rather, the frequency and severity of the individual types of injury must be considered. For example, the appearance of a temporary, acute change in the blood count is assigned a lesser degree of injury than a severe, acute radiation illness with fatal consequences. Accordingly, formation of a papillary or follicular cancer nodule in the thyroid--which in most cases is neither noticed nor causes health injury--is assigned a different weight than leukemia which, as a rule, will result in early death. Therefore, the risks of incurring type of health effects that can result in radiation-induced fatality (mortality risk) are important for evaluating overall health effects caused by ionizing radiation. Among such mortality risks are severe, acute radiation sickness caused by high irradiation of blood-forming organs, and malignancies like lung cancer and leukemia.

Analysis of potential radiation consequences requires information on the relation between dose incurred by a tissue and probability for the occurrence of damage to this tissue. With respect to the type of this dose-effect or dose-risk relation, there is a basic difference between delayed stochastic health effects (cancer, genetic injury) and nonstochastic health effects (acute injury, noncancerous delayed injury). This is illustrated schematically in Figure 7-4.

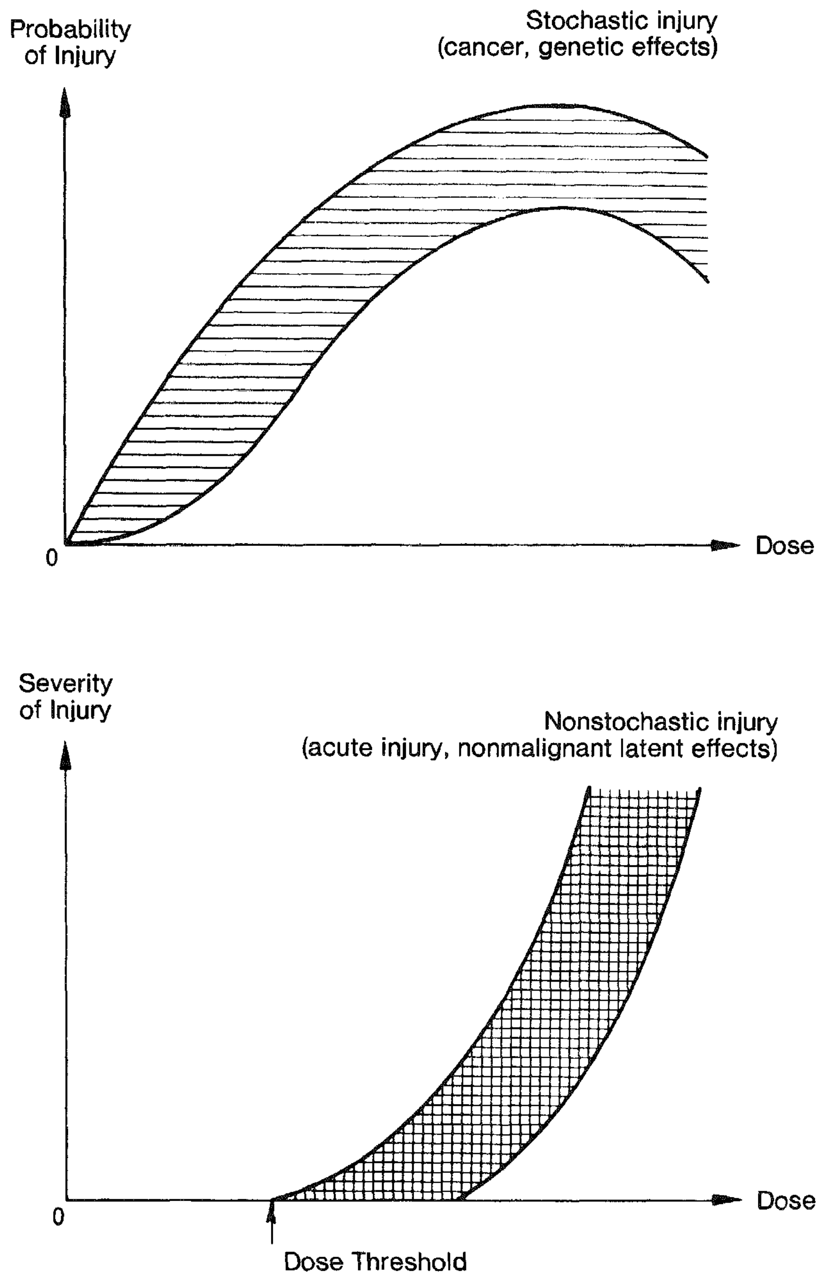


Figure 7-4. Shape f the dose-risk relationship for stochastic and nonstochastic radiation health effects

For nonstochastic health effects a threshold dose exists for the appearance of a health effect. Conversely, experience with carcinogenic and genetic effects of radiation indicate that there is no such threshold dose: thus these types of injury can occur even for low doses, but the probability for the occurrence of such injury does decrease with decreasing dosage. At low doses, i.e., below about 100 rad in the case of a short-term whole body dose, the health effect is determined practically by the carcinogenic and genetic risk of radiation injury. For very high doses, the probability of radiation cancer decreases again since nonstochastic effects are decisive in reducing longevity.

The shape of the dose-risk relation depends not only on the absorbed radiation energy, i.e., on the energy dose to the particular tissue, but also on the chronological distribution of the received dose and the type or quality of radiation. The latter is characterized by local density distribution of ionization in the track of the charged particles. Ionizing particles of high-ionization density along the particle path, i.e., high linear energy transfer (abbreviated LET), generally have a higher biological effectiveness at the same energy dose than low-density ionizing particles, i.e., particles with low LET. Among the former are α-particles, whereas x-rays and γ-rays, as well as electrons or β-rays, belong to the low-density ionizing types.

Radiation protection takes these facts into consideration by introducing a quality factor Q, which is normalized for low-density ionizing rays (x-rays and γ-rays, β-rays and electron beams) to a value of unity with α-particles having a relative value of ten in the Radiological Protection Ordinance (h). The product of energy dose D (unit: 1 rad = 10^{-2} Joule/kg) and quality factor Q yield the so-called dose equivalent H (unit: 1 rem) of the particular type of radiation in the particular tissue: h (rem) = Q X D (rad).

We should mention that radionuclides released during accidents or Class 9 events in nuclear power plants emit exclusively β- and γ-rays. Under these conditions, the fraction of radiation exposure of the population by α-radiation from actinides is very low, so that the total values of dose equivalent to body tissues determined by the dose model (see Section 7.3) correspond to the energy dose to these tissues.

As mentioned above, the form of the dose-risk relation depends on the type or quality of radiation. The dose-effect relations as described below for acute health effects and for somatic latent health effects are used as the basis for the analysis of accident consequences and thus relate primarily to the effects of low-density ionizing radiation or radiation with low LET.

7.5.2 Somatic Early Radiation Health Effects

Acute radiation dose absorbed by bone marrow is nearly the sole factor for the occurrence of acute, life-threatening illnesses resulting from a reactor accident. If a dose threshold is exceeded, blood formation is temporarily interrupted. The radiation effect in the bone marrow is recognizable after a few hours; but it is noticed only after a few weeks by the reduced number of granulated white blood cells and blood platelets in the blood count. The acute health injury of the patient is due almost exclusively to this change in blood count.

The extent of this interruption is so slight after radiation doses up to 200 rad that no severe health consequences are expected. For higher doses, the number of white blood cells can drop so much that the patient becomes highly susceptible to infections. The number of blood platelets can drop to such low values that there is danger of internal hemorrhage, a disease called "acute radiation illness." As a rule, the patient spontaneously recovers from this disease after 6-8 weeks. For high radiation doses, the progress of acute radiation illness can be so severe that death results within two months, in spite of treatment. From animal experiments, we know that mortality increases with radiation dose increase according to an S-shaped curve.

The dose-effect relation B (Figure 7-5) used in WASH-1400 is not based on direct experience with irradiated human beings, but is the result of complicated considerations and extrapolations. In this study, besides curve B, we also use a flatter dose-effect curve D (Figure 7-5) to calculate acute accident consequences. This curve considers that in the affected population there are groups with a greater radiation sensitivity.

These are primarily people with:

- infections, especially chronic infections, e.g., of the respiratory tract and urinary tract
- diseases tending toward hemorrhages, like many diseases of the stomach and intestine (e.g., stomach ulcer, etc.)

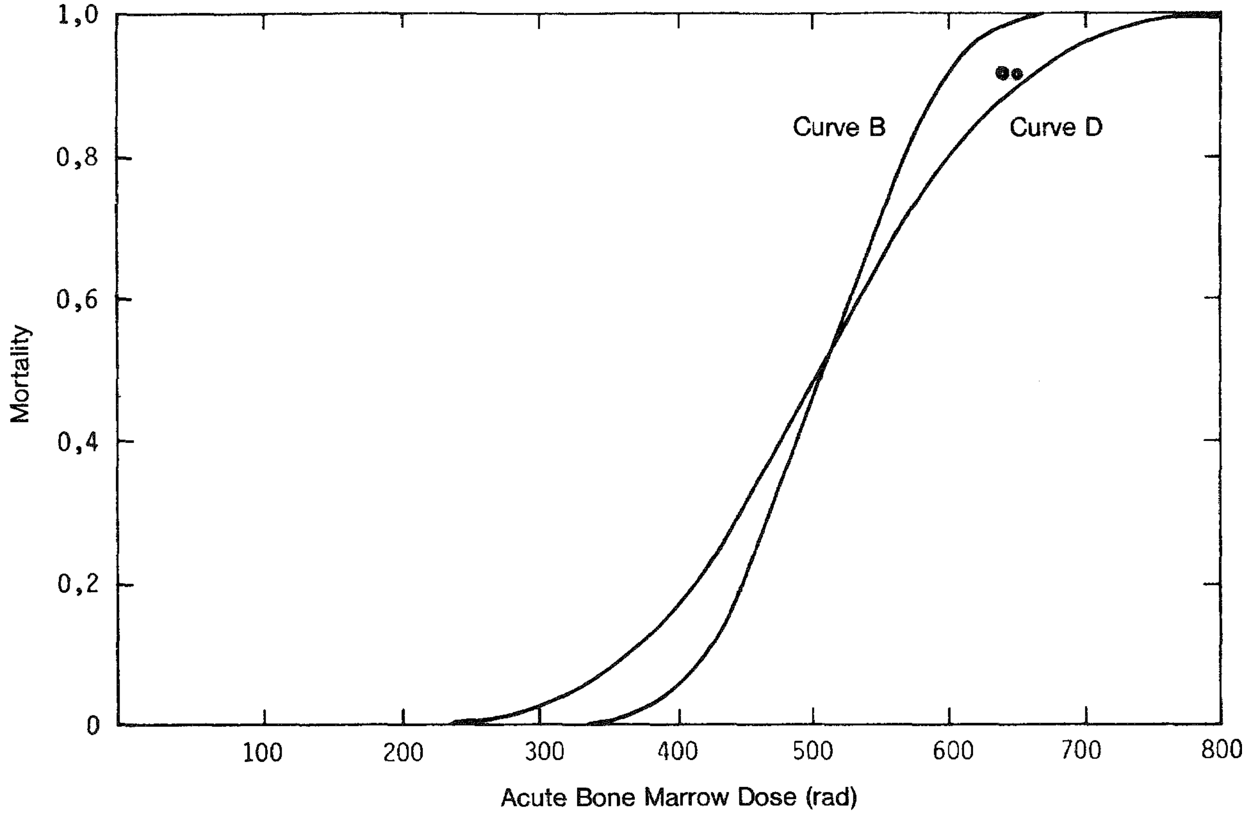


Figure 7-5. Mortality caused by acute radiation as a function of acute bone marrow dose. (Curve B: WASH-1400, 1975. Curve D: this study)

- injuries, burns, operations, and after-treatments with many drugs

- pregnant women.

Considering the annual fluctuations and different intensities of sensitive illnesses and conditions, it is assumed that 10% of the total population is more radiation-sensitive than the rest of the population. For this sensitive group, increased radiation sensitivity is quantified by the adoption of an average lethal dose of 340 rad, which corresponds to the average lethal radiation dose to invalids exposed to whole body radiation.

Advancements in treatment of leukemia and other severe illnesses with highly effective cytostatic drugs have frequently confronted medical oncolygists with the problem of comparatively acute and life-threatening interruption in blood formation.

For a large number of physicians, treatment of acute blood count disruption, which may be caused by the cytostatic drugs themselves or by accidental whole body radiation, is not a new problem. For this reason, we can assume that the vast majority of affected persons will receive appropriate medical treatment, even if the number of such persons is large. In addition, intensive treatment is usually not necessary until about one week after the radiation exposure. Therefore, most people who receive radiation doses between 200 and 500 rad should be saved without permanent injury.

The influence of sensitive diseases on the one hand and of modern medical treatment on the other can be taken into consideration by cumulative normal distribution of mortality as a function of bone marrow dosage with LD-1 (i) of 250 rad, an LD-50 of 510 rad, and an LD-99 of 770 rad. The dose-effect relation D is cut off at a dose threshold of 100 rad.

Besides life-threatening, acute radiation sickness, other morbidities and illnesses can occur that are limited, however, to a relatively short time span after radiation exposure. If acute radiation sickness is not fatal, the patient recovers completely. Other subsequent diseases of acute radiation exposure--except for cancer--are easily treatable, like for instance, radiation-induced thyroid disfunction.

7.5.3 Somatic Latent Radiation Health Effects

We pointed out that nonmalignant (i.e., noncancerous) latent radiation injury occurs only above a threshold dose (see Section 7.5.1). With regard to the dose magnitudes expected for accidents and the severity of these effects, the extent of this type of delayed radiation injury is small compared with injury caused by carcinogenic radiation effects. Malignant latent radiation injury, i.e., cancer and leukemia, must receive primary consideration in the analysis of the risk of somatic latent health effects resulting from reactor accidents.

Recognition of a relationship between radiation effect and cancer frequency is impeded by the fact that cancerous tumors caused by radiation cannot be clinically distinguished from cancerous tumors caused by other natural or civilization-induced effects. An increase in cancer frequency due to radiation is therefore noticed only if the frequency of a particular cancer type increases above that normally caused by natural and other civilization-induced effects. One must consider that the observed, normal cancer rate in the population depends not only on age and sex, but also on location and time, whereby the causal factors of this variation are generally unknown. Average values of normal frequency of several types of cancer in the population of the FRG are compiled in Table 7-3.

The most important source of our knowledge about cancer risk to man are long-term cancer rate studies of survivors of the atomic bomb blasts on Hiroshima and Nagasaki who were exposed for a short time to relatively high, whole body radiation of neutrons and γ-rays. In addition, results of numerous epidemiological studies on radiation cancer risk in groups of patients after treatment by radiation are now available, especially as they pertain to diagnostic and therapeutic treatment by X-rays. Conversely, there are only a few observations of an increased cancer risk for occupationally exposed persons: reliable findings are limited exclusively to groups of persons whose radiation exposure was above the prevailing dose limits for occupationally exposed persons. The most important example is the observed, increased lung cancer frequency in miners working in mines containing high radon concentrations. From the findings on cancer rates among populations living in regions with an increased background radiation level, no significant increase in normal cancer risk has yet been found.

A detailed and extensive presentation of present information on radiation cancer risk in man is given by the report of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) of 1977. Present observations on the

Table 7-3. Average cancer risk (mortality) to the German population, based on the age distribution and observed cancer statistics 1974/75 (according to general health data [13])

Manifested location (tissue)	Frequency (cancer cases per million persons per year)	Relative frequency in percent[a]	Mortality risk[b] (cancer cases per million persons)
Respiratory tract	400	3.3	29,000
Stomach	350	2.9	25,000
Intestine	350	2.9	25,000
Breast[c]	180	1.5	13,000
Lymphatic and blood-forming organs	150	1.2	11,000
Leukemia	70	0.6	5,000
Pancreas	75	0.6	5,400
Kidneys	70	0.6	5,000
Liver	45	0.4	3,200
Bones[d]	10 - 15	ca 0.1	700 - 1100
Thyroid[d]	5 - 10	ca 0.05	350 - 700
Other tissues including nonidentified types	ca. 800	6.6	ca. 56,000
Whole body (total risk)	2,400	20	173,000

[a]Relative to total number of fatalities.

[b]Using an average life span of 72 years.

[c]Averaged over both sexes.

[d]Estimated values according to data of the Saar cancer record.

carcinogenic effect of ionizing radiation in man are limited--with a few exceptions--to groups of persons receiving doses above about 50 rem. The dose magnitudes resulting from the dosimetric model applied to reactor accidents lead to expected doses below 10 rem for the majority of the affected population. To estimate the potential radiation cancer risk we need to extrapolate from the risk values observed for higher doses down to low doses.

In WASH-1400, three extrapolation laws were considered:

1. an "upper bound estimate," which designates purely proportional dose-risk relations $R = aD$ (a = risk coefficient)

2. a "central estimate," which is a piecewise linear relation for dose-risk with reduced risk coefficients for lower doses and/or dose rates

3. a "lower bound estimate," which is a dose-risk relation with a threshold dose of 10 or 25 rem.

In calculating accident consequences, the American study used the central estimate, which approximates a so-called linear quadratic dose-risk relation $R = \alpha D + \beta D^2$. In comparison to a simple proportional relationship, a smaller value of $\alpha = a/5$ was assigned for the linear term. Furthermore, it was assumed that the coefficient β of the quadratic term decreases to 0 with decreasing dose rate.

Observations on the dose dependency of radiation cancer risk for atomic bomb survivors and findings from animal experiments and cytologic studies indicate that for low-density ionizing radiation (radiation with low LET) this type of linear-quadratic relation can approximate the dose-risk relation. From findings on humans with regard to derived risk data, the statistical uncertainty range of the risk value increases severely with decreasing dose. A purely proportional dose-risk relation for carcinogenic effect can therefore not be excluded for low-density, ionizing radiation.

Consistent with recommendations of the International Commission on Radiological Protection (14), the present study uses a purely proportional dose-risk relation $R = aD$ for low-density ionizing radiation (X-rays and γ-rays). This assumption corresponds to the upper-bound estimate of WASH-1400.

Figure 7-6 illustrates the proportional and linear-quadratic extrapolation from an observed risk value to low doses.

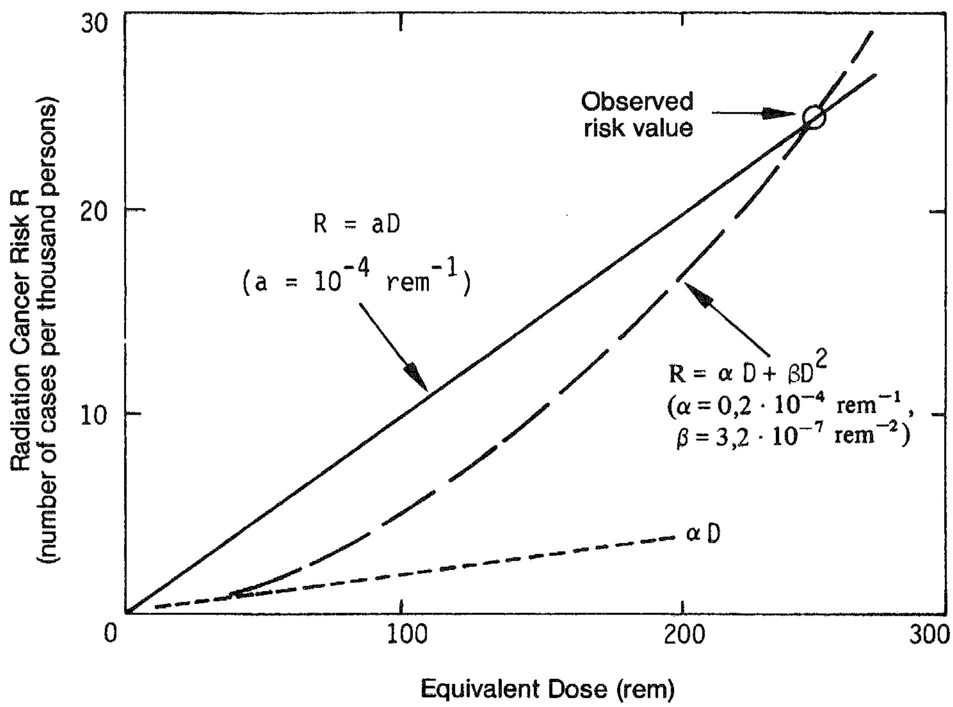


Figure 7-6. Extrapolation to small doses: comparison between proportional and linear-quadratic dose-risk relationship (example: α = a/5).

The ICRP points out in their recommendations that the use of a purely proportional dose-risk relation for low-LET ionizing radiation will probably lead to overestimating the actual expected risk for low dose rates or low doses when experimental values of relatively high doses and high-dose commitment are extrapolated (see Figure 7-6). The proportional extrapolation of radiation cancer risk applied in the present study for low doses can be considered a conservative assumption.

In the establishment of risk coefficients a = R/D for relevant, radiation-induced types of cancer, we proceeded from the observed values of the dose-risk relation in man, whereby scientifically confirmed results were used. In Table 7-4, column 2, the available data analysis gives expected values or an anticipated range of the dose-risk relation for the additional cancer risk caused by radiation (mortality risk) in human tissue. The values are averages for age and sex, and take into consideration the present age distribution of the German people.

The third column of this table gives the reference values of risk coefficients recommended by the ICRP for purposes of radiation protection. They are approximately in the middle of the expected range, which results from present experiences on radiation cancer risk in man. To estimate the mortality risk due to malignant, somatic latent radiation injury, we use the recently recommended reference values of risk coefficients for individual body tissues given by the ICRP (1977) (14).

The above values lead to a radiation cancer risk of about 100 premature cancer fatalities among 1,000,000 persons, each receiving one rem whole body dose or one rem effective dose. With respect to cancer mortality, the red cell bone marrow (leukemia), lungs, and the female breast are considered the most sensitive body regions for radiation.

The expected long-term doses for these tissues are used to calculate accident consequences, in the case of red cell bone marrow (leukemia), lungs (lung cancer), bone surface (bone cancer), thyroid (malignant thyroid tumors), and female breast (breast cancer). For other body tissues, the average long-term dose in the whole body is used as a representative dose.

The analysis of accident consequences takes into consideration the relative age dependence of radiation cancer risk for the individual types of cancer. In order to do this, a procedure is used that corresponds to one in WASH-1400. Specifically, it employs the higher leukemia risk in children and greater radiation sensitivity of the fetus for irradiation of the uterus. In the latter case, a total cancer risk coefficient of 2.5×10^{-4} is used.

Table 7-4. Expected value of the risk-equivalent dose-ratio for additional mortality due to cancer in body tissues resulting from radiation averaged over age and both sexes

Organ or tissue	Radiation cancer risk-equivalent dose[a] (number of cases per million persons per rem): Expected range[b]	Radiation cancer risk-equivalent dose[a] (number of cases per million persons per rem): Reference value ICRP 26 (1977)
Cell bone marrow (leukemia)	15 - 40	20
Breast	15 - 40	25
Lung	10 - 30	20
Bone surface (bone cancer)	≤ 5	5
Thyroid	5 - 10	5
Digestive organs, total	20 - 50	50 (digestive and other organs combined)
Other organs, total	10 - 30	
Total cancer risk for uniform whole body radiation	80 - 200	125

[a]Average equivalent dose to the effected tissues.

[b]Values considering the age distribution of the German population.

With respect to the chronological distribution of cancer incidence, a significant difference exists between leukemia and solid types of cancer. In the case of radiation-induced leukemia, the average latent period is about 10 years, whereas for the other types of cancer it is in a range of 20-30 years. The distribution function of the latent period approximately corresponds to a logarithmic normal distribution. For simplification, this function is approximated as in WASH-1400 by a "plateau" model, i.e., a constant cancer incidence is assumed during the manifestation period.

Description of the results for reactor accidents includes on one hand the individual risk for somatic latent health effects. On the other hand, a total number of cases--the so-called collective latent health effect risk--is given to evaluate the total hazard for the population. For the assumed proportional dose-risk relation, this population radiation risk is proportional to the population dose (unit: 1 man-rem, see footnote (a) at the end of this chapter). This population dose results from integration of the spatially dependent doses multiplied by the number of exposed persons with integration performed over the entire affected region.

To place into perspective the radiation cancer risk expected from the accident, the normal cancer rate in the population can be used (see Table 7-3). In Figure 7-7, the additional anticipated radiation cancer risk, assuming a proportional dose-risk relation, is given as a function of whole body exposure or effective dose. It follows that for a mean lifetime dose of 10 rem, which corresponds to an average natural lifetime radiation exposure of the population, a radiation cancer risk of about 0.1% (1 in 1000) would be expected. For comparison, the observed normal cancer frequency (mortality) in the German population is about 20% (1 in 5). An additional effective dose of 1 rem would thus increase the cancer rate from 20% to about 20.01%.

7.5.4 Genetic Radiation Effects

Genetic radiation effects have not yet been found in exposed population groups. The extent of genetic radiation effects can be estimated with similar reliability as for somatic latent effects from results of animal experiments. Some radiation-induced mutations will lead to hereditary disorders in children and grandchildren (primarily dominant mutations). The recessive mutations usually do not become pronounced for many generations, that is, until they have mixed into the genetic

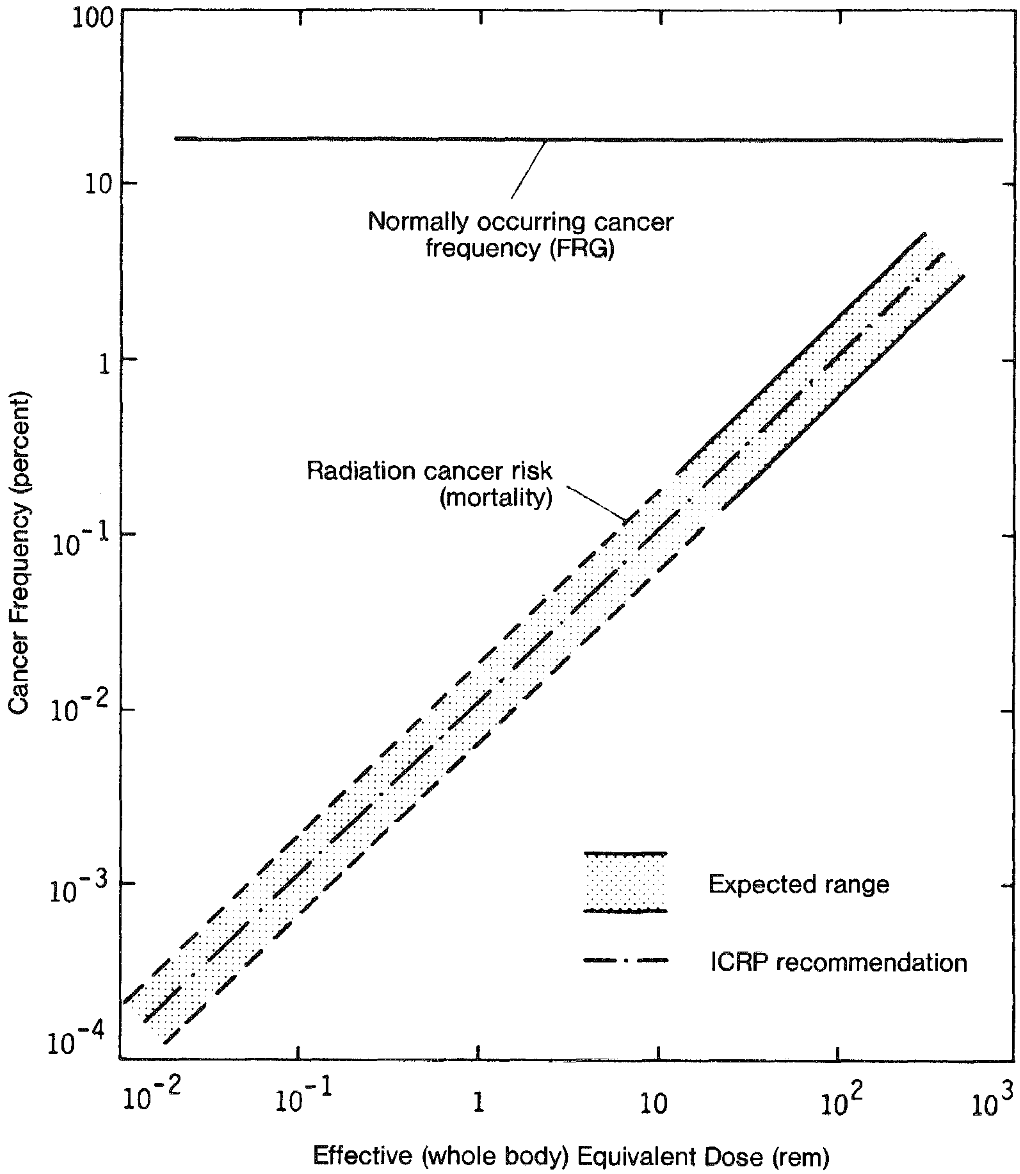


Figure 7-7. Expected value of the radiation cancer risk to the population as a function of the effective (whole body) equivalent dose assuming a proportional dose-risk relationship without threshold. For comparison, the present "normal" cancer frequency among the population in the FRG population is given.

"pool" of the total population. "Spontaneous" mutations, mutations caused by natural and man-made radiation burdens from preceding generations, and mutations caused by chemical pollutants determine the number of hereditarily affected children.

The extent of hereditary health injury can range from minor, metabolic, and shape changes often recognizable only by special methods, to severe disorders associated with life-long infirmity. This multiplicity of potential genetic consequences defies simple classification of effects on the fate of those affected. Even if we restricted the discussion to hereditary diseases with clinical relevance, they still could not be compared to the somatic consequences calculated in Sections 7.5.2 and 7.5.3 on a common basis.

For this reason, to prevent possible misinterpretation, we did not make a numeric determination of hereditarily injured children in subsequent generations. However, the collective genetically significant population dose determining the extent of all hereditary injury was calculated. This population dose relates to a population of 670,000,000 and can be compared with the collective, genetically significant population dose from other radiation sources. This method does allow comparison on a common basis of genetic effects of the studied reactor accidents with other, genetically relevent radiation exposures of the population.

With the genetic risk coefficients given in the scientific literature, we can estimate--from the genetically significant population dose--the order of magnitude number of children born in the next two generations with clinically significant hereditary injury due to radiation burdens. The ICRP states that risk for the first few generations would be less than 10^{-4} per man-rem (14). To estimate the individual genetic risk for the appearance of clinically significant hereditary injury in the children of exposed persons, a risk coefficient of 4×10^{-5} per rem gonad dose can be used.

7.5.5 Calculation of Accident Consequences

The resulting individual probability of injury S is calculated first on the basis of the expected doses of the dose model (see Section 7.3). These are the probabilities with which persons at the respective location die due to the expected doses. Using a particular population as a basis, the collective injury can then be determined, i.e., the total expected cases of health effects.

The calculation of probability of somatic early radiation health effects is based on the short-term dose to the bone marrow (see Section 7.3). Determination of injury probability then results from the dose-effect relations as illustrated in Section 7.5.2.

To calculate probability of latent somatic injury, we proceed from long-term doses to the following organs (see Section 7.3):

- bone marrow
- bone surface
- lung
- thyroid
- breast
- whole body (representative "organ" for the remainder of the body).

Probability determination is then based on the linear dose-risk relation without a threshold dose as illustrated in Section 7.5.3. Genetic effects calculations are discussed in Section 7.5.4.

By integration of the health effect probability multiplied by the number of affected persons in a particular region, we finally obtain the expected total early and the latent health effects, i.e., the number of expected cases. For this integration, we use a site-specific population distribution to a range of 80 km (j). From 80 km to 540 km (j), we assume a uniform population density of 250 people per square kilometer (representative of Central Europe). For the region beyond 540 km, where the remainder of the radioactivity is deposited, an average population density of 25 per square kilometer is used (representative of a 2500 km circle around Central Europe, including all European land and water areas).

FOOTNOTES

(a) These distances corresponds to the 50 and 350-mile range respectively used in WASH-1400.

(b) 1 rad = 0.01 J/kg or erg/g. From an energy dose of 1 rad, a dose equivalent is derived which considers the biological effectiveness of the different types of radiation. Its unit is the "rem" (see also Section 7.5.1 and the appendix). To simplify the text, the word "dose" is used below for "dose equivalent" as well. By using the units "rad" and "rem," we can then tell whether we are dealing with energy dose or equivalent dose.

(c) Selection of these organs is explained in Section 7.5.

(d) By "protective actions," we mean in general those measures that can be easily implemented, e.g., remaining indoors. Countermeasures require special preparations, for instance, population evacuation plans.

(e) The population dose is given in man-rad units, and the population dose equivalent in man-rem units. The suffix "man" is not a dimension in the physical sense; it is only used to express that we are dealing with the sum of individual doses. To simplify the text, we use the phrase "population dose" for population dose equivalent below. By reference to the units man-rad or man-rem, one can readily tell if we are dealing with energy dose or population dose.

(f) Corresponds to surface area of 33 km^2.

(g) $DF = \frac{\text{radioactivity before decontamination}}{\text{radioactivity after decontimination}}$

(h) The International Commission on Radiological Protection (ICRP) has recently recommended a quality factor of 20 for α-particles.

(i) LD-1: dosage resulting in 1% mortality (LD-50 corresponds to 50%, LD-99 corresponds to 99%.

(j) Selection of the ranges 80 km (50 miles) and 540 km (350 miles) is discussed in Section 7.1.

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Section 8

RESULTS AND UNCERTAINTIES IN THE RESULTS

8.1 RESULTS

8.1.1 Introduction

Section 5 studies the particular event sequences that can lead to a melt of the reactor core. Section 6 concerns the core melt process through release of fission products from the containment. The entire spectrum of possible releases from a controlled LOCA to a core melt accident with early failure of the containment is divided into eight categories. Table 6-3 shows the characteristics of the eight release categories. In part, these characteristics consist of representative values for the released quantities of radionuclides and for thermal energy, as well as the chronology of release. Also, they include the calculated frequency of a given release.

In Section 7 ("Accident Consequence Model") the computer model to determine accident consequences is illustrated. As described there, 115 weather sequences were used for each of the four meteorological siting regions of the FRG; these represent all potential weather sequences in a satisfactory manner. It is assumed these weather sequences will occur with the same probability in each of the 36 wind directions (10 degree intervals).

These dispersion calculations were applied to the given 19 sites with attendant population distributions. It was assumed that at these sites a total of 25 reactor units are operating (see Section 7.1).

On the basis of these conditions there are a total of 8 X (115 X 36) X 19 = 629,280 "release category-weather sequence-wind direction-site" combinations. The necessary accident consequence calculations were performed for them. In order to account in the evaluation for the assumed 25 reactor units, the results for sites were weighted according to whether there were one or more units at each site.

The results of these accident consequence calculations must be viewed in connection with the associated frequencies.

If we consider one plant, eight release categories, 115 weather sequences, and 36 wind directions, then by using the above method we would have to calculate consequences from 8 X 115 X 36 = 33,120 individual events, i.e., combinations of release category, weather sequence, and wind direction.

The probability of each of the potential combinations of weather sequence and wind direction is (1/115) X (1/36) = 2.4 X 10^{-4}, i.e., the frequency of all individual events belonging to a release category is the same. The calculation scheme for the frequency of individual events for all release categories is given in Table 8-1.

Previous considerations pertained to one plant. If we include all 25 plants in the discussion, then the number of individual events is increased by a factor of 25 from 33,120 to 828,000, since now the population distribution around each of the 25 plants must be considered. The frequencies and probabilities named in Table 8-1 do not change, however, since they relate to one year of operation of a particular plant and thus are independent of the total number of plants under discussion.

At sites with two or three reactor units, the actual population distribution is multiplied by a factor of two or three. In this case, calculations are performed for only one reactor unit, and the frequencies are multiplied by a factor of two or three. The consequence calculation is thus reduced from 828,000 individual events to the number given above, 629,280.

If several individual events from different release categories, weather sequences, wind directions, or sites lead to the same consequence magnitude, then the attendant frequencies are added. Thus, each consequence magnitude is clearly connected to a frequency. The resulting function is called the frequency density function of consequences from 25 plants.

Section 8.4.1 will discuss several important model properties that cause uncertainties in the results.

Table 8-1. Occurrence frequencies of release categories and of specific calculated consequences, per plant operating year

Release category	Frequency per operating year[1)] $[a^{-1}]$	x	Probability for weather sequence 1/115	x	Probability for wind direction 1/36	=	Frequency of a specific consequence per operating year $[a^{-1}]$
FK 1	2×10^{-6}		8.7×10^{-3}		2.8×10^{-2}		4.8×10^{-10}
FK 2	6×10^{-7}		8.7×10^{-3}		2.8×10^{-2}		1.4×10^{-10}
FK 3	6×10^{-7}		8.7×10^{-3}		2.8×10^{-2}		1.4×10^{-10}
FK 4	3×10^{-6}		8.7×10^{-3}		2.8×10^{-2}		7.2×10^{-10}
FK 5	2×10^{-5}		8.7×10^{-3}		2.8×10^{-2}		4.8×10^{-9}
FK 6	7×10^{-5}		8.7×10^{-3}		2.8×10^{-2}		1.7×10^{-8}
FK 7	1×10^{-4}		8.7×10^{-3}		2.8×10^{-2}		2.4×10^{-8}
FK 8	1×10^{-3}		8.7×10^{-3}		2.8×10^{-2}		2.4×10^{-7}
Frequency for release from one of the categories 1 to 6	$1.0 \times 10^{-4}\ a^{-1}$						
Frequency for release from one of the categories 1 to 6 for 25 nuclear power plants	$2.5 \times 10^{-3}\ a^{-1}$						

1) These values are rounded off and contain additions on the order of 10% from the neighboring categories (see Section 6.6.3).

8.1.2 Probability Density Functions for Collective Injuries and Doses and Their Complementary Cumulative Distributions

As in WASH-1400, frequencies are illustrated as a function of the magnitude of the calculated collective injuries (or population dose) in the form of complementary cumulative distribution functions.

To determine these distribution functions, we first calculate the probability density function according to the method described in the previous section.

The probability density function tells the frequency of occurrence of this injury KS (or this dose KD) (a) for each collective injury KS (or for each collective dose KD).

Within the framework of this study we are primarily interested in knowing with which frequency a given magnitude of consequence is to be expected in a given interval. Therefore, it is not the frequency density function for each order of consequences, but rather the frequency density of consequence intervals (so-called consequence classes) that is illustrated.

The complementary cumulative distribution function finally gives a frequency for each collective injury KS or for each collective dose KD) that this consequence KS (or this dose KD) or a greater one will occur.

This function is obtained by summation of class frequencies of each collective injury (or collective dose) group that is greater than or equal to a predetermined collective injury (KS or collective dose KD). The summation continues to the greatest possible injury that can occur.

The accident consequence calculations determined the collective injury (or collective dose) for the following health effects and doses:

- early health effects (fatalities due to acute radiation syndrome)
- latent health effects (fatalities due to leukemia and cancer)
- genetic effects (genetically significant collective dose).

The probability density functions for consequence classes and complementary cumulative distribution functions for collective injuries (or collective dose) for 25 units were calculated and presented.

In addition, characteristic data are compiled for the areas and persons affected by the countermeasures "evacuation" (region A), "fast relocation" (regions B_1 and B_2), and "relocation" (region C).

8.1.2.1 Early Health Effects (Deaths Due to Acute Radiation Syndrome). Early injuries can occur only above a threshold dose of 100 rad, according to the German dose-effect relation (curve D in Figure 7-5). Such injuries are therefore limited to a region near the site (see Section 8.1.3).

Summarized over all release categories or over all categories except FK1, the probability density functions for consequence classes of acute fatalities are illustrated in Figure 8-1 (b).

From these curves, we can read the occurrence frequency for the magnitude of various consequence classes. In constructing these curves, each logarithmic decade was subdivided into 10 equal size intervals. The complementary cumulative distribution functions (CCDFs) of early fatalities are illustrated by release categories in Figure 8-2.

Figure 8-3 shows the CCDFs (with and without release category 1) summed over all release categories. For comparison, in addition to the distribution based upon the dose-effect relation (D) used in the German risk study, the distribution based upon the dose-effect relation (B) in WASH-1400 is also shown (c). As we can see from Figure 8-3, the results for the two dose-effect relations differ only slightly.

The characteristic quantities of both distribution functions are summarized again in Table 8-2.

From the figures, we see that early injuries occur only for release categories FK1, FK2, FK3 and FK4, and then only under selected environmental conditions. Table 8-3 discusses the following:

- conditional probability of an injury (acute fatality) given a release for each release category
- injury occurrence frequency per year with regard to the occurrence frequency of the particular release categories, and
- average injury (number of acute fatalities) given a particular release.

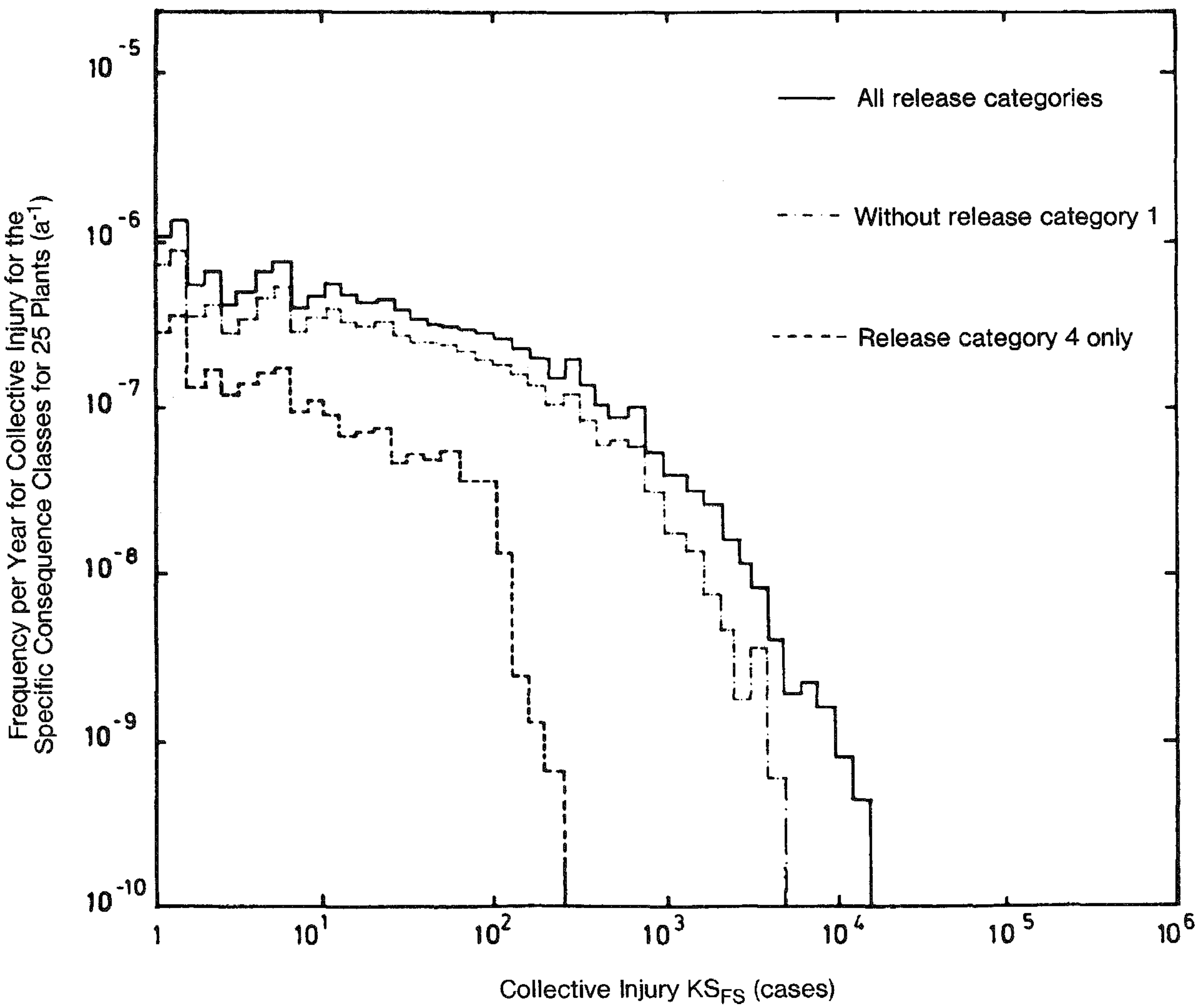


Figure 8-1. Frequency distribution density functions for early fatalities by consequence classes. Release category 4 is given as an example of an individual curve

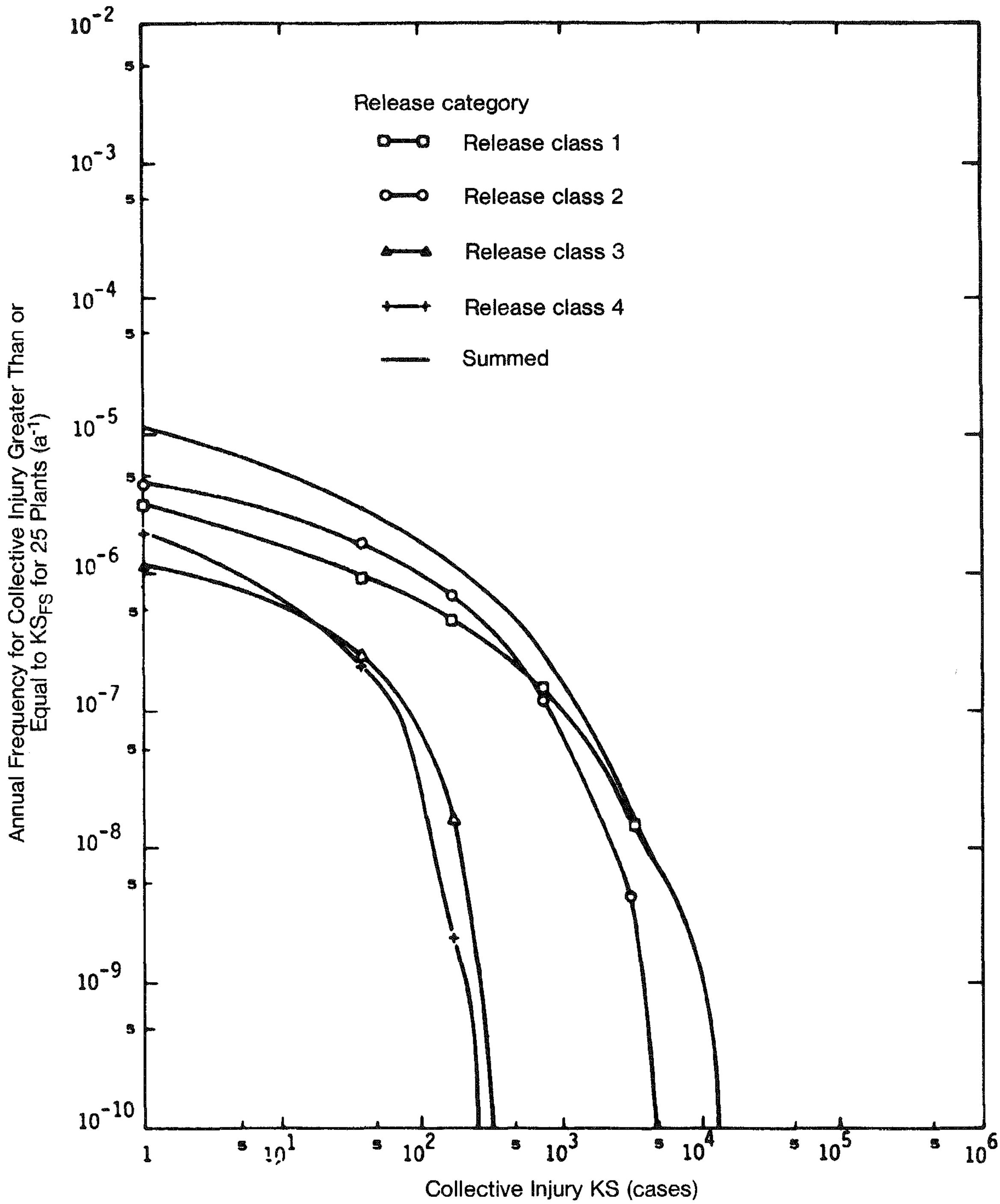


Figure 8-2. CCDF for early health effects, keyed according to release categories (German study dose-risk relationship)

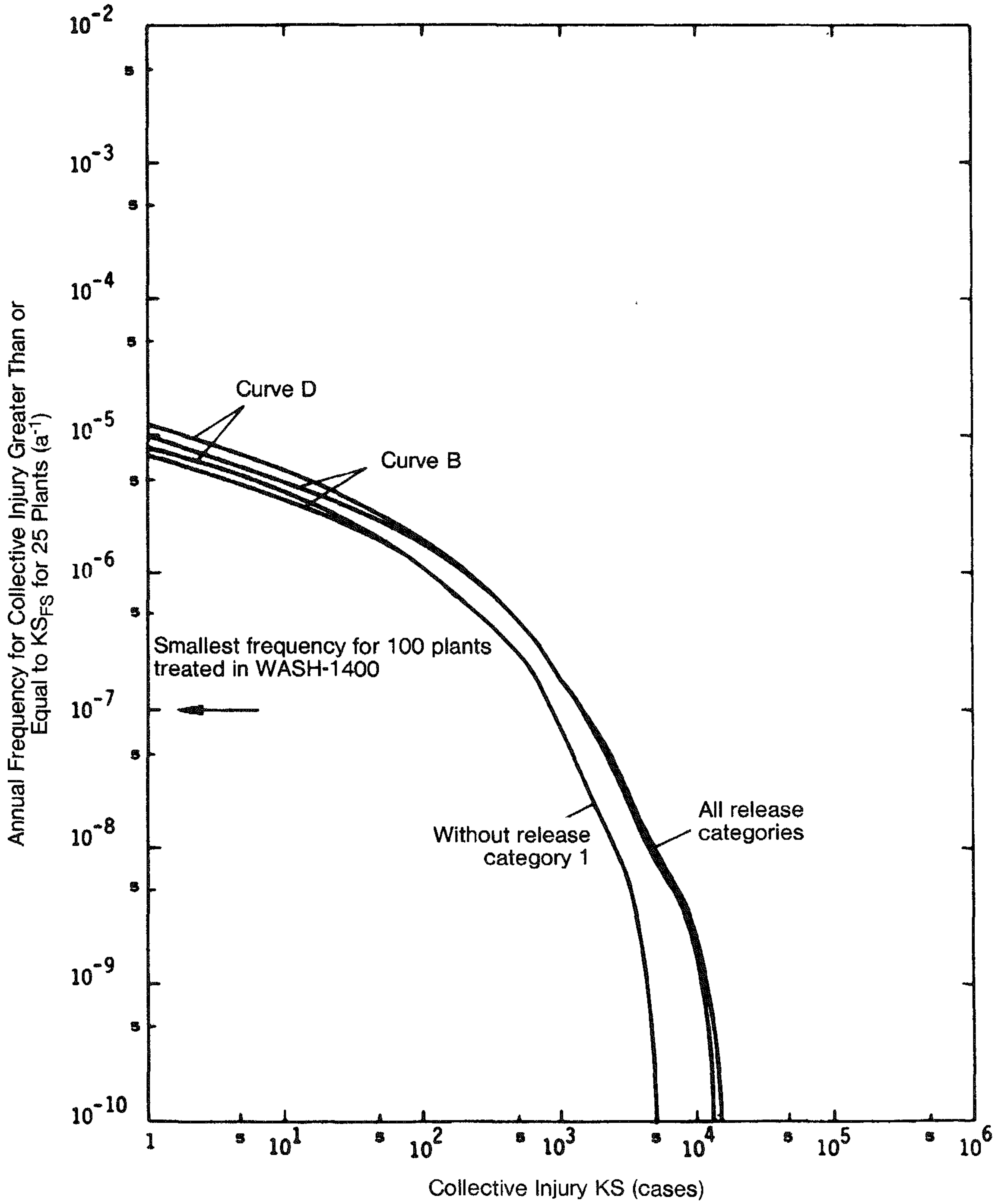


Figure 8-3. CCDF of early health effects calculated with the dose-effect relationship of curves B and D from Figure 7-5

Table 8-2. Characteristic quantities of the CCDF for early health effects (societal health effects) for 25 plants

Release category	German study dose-risk relation (curve D)				American study dose-risk relation (curve B)			
	Probability of KS > 0	Smallest societal health effect KS_{min}(a)**	Greatest societal health effect KS_{max}	Average societal health effect $\overline{KS}$ (b/c)#	Probability of KS > 0	Smallest societal health effect KS_{min}(a)**	Greatest societal health effect* KS_{max}	Average societal health effect $\overline{KS}$ (b/c)#
FK 1	7.1%	0 (92.9%)	14,500	9.4 (96.5%/3.5 %)	5.2 %	0 (95.8%)	12,200	8.2 (97.0%/3.0%)
FK 2	33.1%	0 (66.9%)	5,100	32 (97.9%/2.1%)	27.6%	0 (72.4%)	4,900	29 (98.0%/2.0%)
FK 3	8.2%	0 (91.8%)	320	2.1 (92.8%/7.2%)	6.8%	0 (93.2%)	330	1.7 (93.3%/6.7%)
FK 4	2.8%	0 (97.2%)	260	0.4 (97.2%/2.8%)	2.3%	0 (97.7%)	270	0.3 (97.7%/2.3%)
FK 5	0%	--	--	--	0%	--	--	--
FK 6	0%	--	--	--	0%	--	--	--
FK 7	0%	--	--	--	0%	--	--	--
FK 8	0%	--	--	--	0%	--	--	--

* The greatest number of societal health effects KS_{max} resulting from one of 115 x 36 x 19 = 78,660 considered accident sequences.

** A% of the results leads to the smallest number of societal health effects KS_{min}.

\# At B% of the cases, the societal health effects are less than KS; at C% they are greater than KS.

Note: All probabilities given in this table are conditional probabilities, i.e., a release was presumed.

Table 8-3. Occurrence frequency of a release, conditional probability of an injury, occurrence frequency for an injury, and average number of acute fatalities for different release categories

Release category	Occurrence frequency per operating year $[a^{-1}]$	Probability of injury given the release	Occurrence frequency of injury per operating year $[a^{-1}]$	Occurrence frequency of injury for 25 units* $[a^{-1}]$	Average injury given the release (fatalities)
FK 1	2×10^{-6}	7.1×10^{-2}	1.4×10^{-7}	3.5×10^{-6}	9.4
FK 2	6×10^{-7}	3.3×10^{-1}	2×10^{-7}	5.0×10^{-6}	32.0
FK 3	6×10^{-7}	8.2×10^{-2}	4.9×10^{-8}	1.2×10^{-6}	2.1
FK 4	3×10^{-6}	2.8×10^{-2}	8.4×10^{-8}	2.1×10^{-6}	0.4
FK 5	2×10^{-5}	--	--	--	--
FK 6	7×10^{-5}	--	--	--	--
Sum	1×10^{-4}	--	4.7×10^{-7}	1.2×10^{-5}	--

* Example for release category FK 1 for 25 units: $2 \times 10^{-6} \times 7.1 \times 10^{-2} \times 25\ a^{-1} = 3.5 \times 10^{-6}\ a^{-1}$

We took the German dose-effect relationship (GDER) as a basis (curve D from Figure 7-5).

In terms of unit operating years, the occurrence frequency of acute fatality is 4.7 X 10^{-7}/year (sum of fatality occurrence frequencies [fourth column in Table 8-3] for release categories FK1-FK4). Comparison with the occurrence frequency for a core melt accident (sum of occurrence frequencies for release categories FK1-FK6) of 1 X 10^{-4} per reactor year shows that early injury (acute fatality) will occur in far less than 1% of core melt accidents.

The fact that early fatalities occur only for a very small proportion of postulated accidents and that there is large variation in possible numbers of fatalities (which is, for example, expressed by a large ratio of maximum number of fatalities to average number of fatalities--see Table 8-2), can be explained as follows:

(a) The different weather sequences cause very different spatial distributions of radionuclide concentrations, resulting in different doses. Even for large releases, the threshold dose for early fatality (100 rad) is not exceeded in most cases.

(b) Population distribution density varies near the various reactor sites. In addition, at very short range the radioactive plume is still relatively small compared to the size of populated regions (villages and cities). The calculated results therefore depend greatly on population distribution and wind direction.

A few examples are provided with respect to Figures 8-2 and 8-3. If we take the GDER as a basis, the following values for frequency of early fatalities greater than or equal to KS are found (Table 8-4).

Figures 8-2 and 8-3 illustrate the results up to maximum number of early fatalities calculated. This is defined as the greatest number of early fatalities that could be demonstrated in calculations for a total of 629,280 simulated accident sequences. The maximum number of early fatalities occurs for an accident sequence having the worst release, weather, and population distribution conditions (d).

Including the release category FK1 (steam explosion), the maximum number of early fatalities is about 14,500 (GDER). The maximum number is caused by release category FK1 and has a calculated occurrence frequency of 4.8 X 10^{-10}/year.

Excluding the release category FK1, the maximum number of fatalities (about 5,100) is caused by release category FK2, and has a calculated (GDER) frequency of 1.4×10^{-10}/year.

Table 8-4. Selected points on the CCDFs for early health effects for 25 units (GDER)

Occurrence frequency per year	Collective injury > KS	
	Including FK 1 (steam explosion) KS	Excluding FK 1 (steam explosion) KS
1/ 100,000	2	< 1
1/ 1,000,000	200	120
1/ 10,000,000	1,400	870
1/ 100,000,000	4,000	2,500
1/ 1,000,000,000	11,000	4,300

There are many early fatalities when a large release occurs at sites with a relatively high population density, when the wind is blowing into the sector of greatest population density, and when it rains near the plant, contaminating the ground.

The studies in WASH-1400 are based on 100 plants. The results published there, normalized to one plant, extend to a frequency of 10^{-9}/year. Therefore, for 100 plants the lower frequency limit will be 100×10^{-9}/year = 10^{-7}/year. This lower limit of consideration, based on arguments of statistical accuracy of random samples, is so indicated in Figure 8-3, as well as in the following curves of complementary cumulative distribution functions.

8.1.2.2 Latent Somatic Health Effects (Deaths Due to Leukemia and Cancer). Latent somatic health effects can occur at all dose values under the assumed linear dose-risk relation (see Section 7.5). They are determined wherever the population is exposed to radiation through radioactivity transport. Latent effects are therefore long range and not--as in early health effects--limited to the proximity of the site (see Section 8.1.3). According to the model, consequence magnitude is determined primarily by the large number of persons receiving a small dose.

Analogous to Figure 8-1, Figure 8-4 presents probability densities for consequence classes of latent somatic injury, with and without inclusion of release category FK1 ("steam explosion"). The shapes of these curves are generally more irregular than the corresponding curves for acute fatalities, and they exhibit greater numbers of local maxima.

This complex structure results from the addition of probability density functions for the individual release categories. The characteristic shape of these individual curves is illustrated from the example curve for release category FK4. Between the minimum and maximum numbers of deaths, the occurrence frequency increases over a very broad range. This curve shape is due to relatively small radioactive concentrations in the air and ground that are much more frequent than high concentrations. But since the extent of countermeasures is smallest for low concentrations, the greatest number of fatalities results for these concentrations.

The complementary cumulative distribution functions for latent cancer fatalities are illustrated in Figure 8-5 by release categories. Table 8-5 states the characteristic parameters of these distributions. As shown in the table, latent cancer fatalities occur for all release categories with the exception of category FK8. For FK8, the largest number of latent cancer fatalities is much less than one.

In contrast to early deaths (see Table 8-5), the relatively small ratio of maximum to average latent cancer fatalities is striking. This is because the numbers of latent cancer fatalities vary only slightly for the following reasons:

- The assumed dose-risk relation for latent cancer fatality is linear and has no threshold value. Therefore, the different weather sequences causing activity dispersion are less influential than the total deposited radioactivity. Aside from certain fluctuations in the transport time, this is primarily proportional to the released quantity of activity.

- A large fraction of latent cancer fatalities is caused by widespread radioactivity of low concentration. At long range, the radioactive plume is already relatively broad and extends over wide regions. Variations in the population density thus lose their significance.

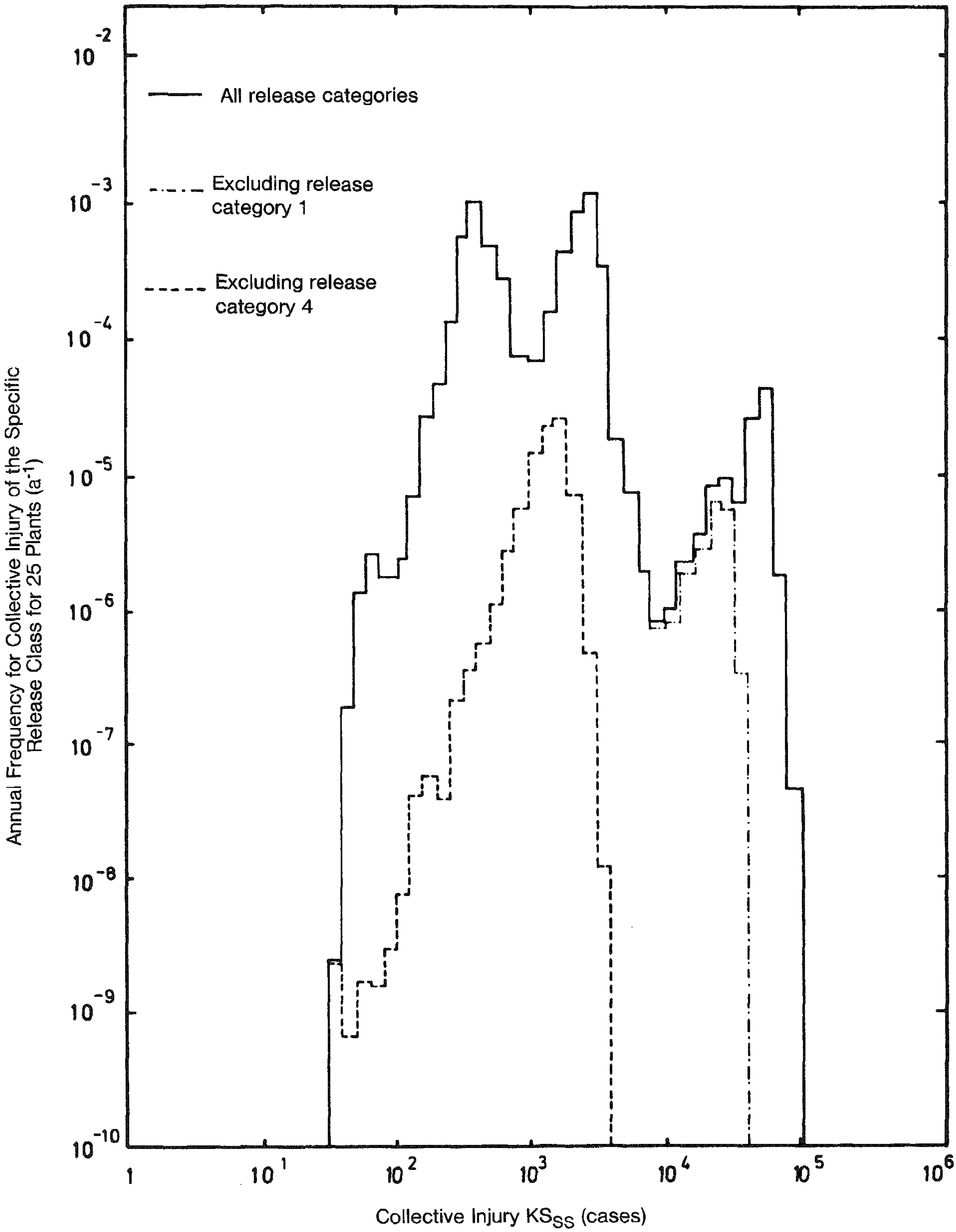


Figure 8-4. Frequency distribution density functions for somatic latent health effects by release classes. As an example of an individual curve, that of release category 4 has been shown.

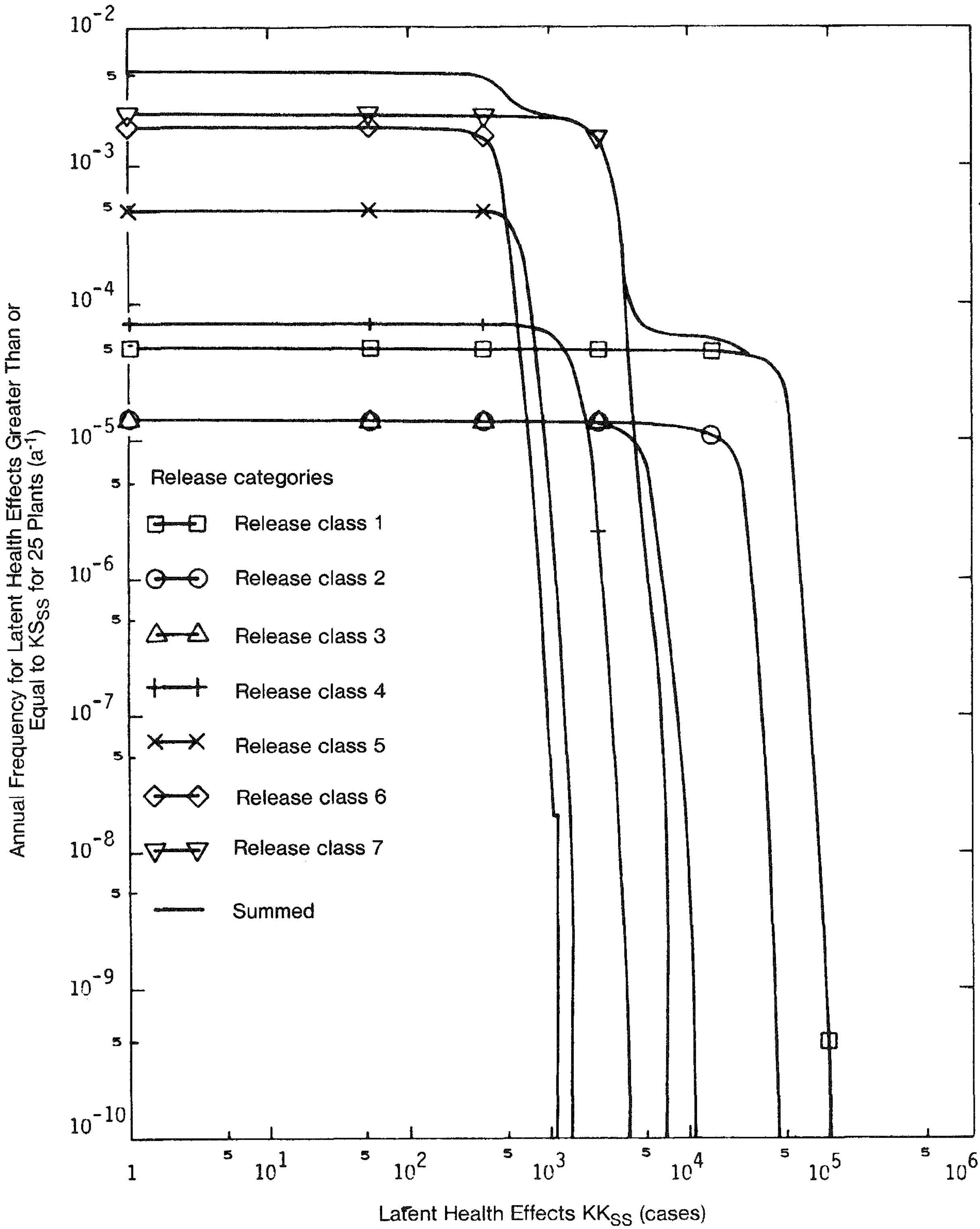


Figure 8-5. CCDF for latent health effects, keyed according to release categories

Table 8-5. Characteristic quantities of the CCDFs of latent health effects (collective injury KS) for 25 units

Release category	Smallest number of fatalities KS_{min}	Greatest number of fatalities KS_{max}	Average number of fatalities $\overline{KS}$ (b/c)*
FK 1	3,200	104,000	49,000 (27.7%/72.3%)
FK 2	1,000	44,000	22,000 (43.5%/56.5%)
FK 3	160	11,300	5,000 (50.4%/49.6%)
FK 4	30	3,700	1,600 (51.6%/48.4%)
FK 5	160	1,500	660 (55.6%/44.4%)
FK 6	80	1,200	420 (47.2%/52.8%)
FK 7	130	7,000	2,400 (42.8%/57.2%)
FK 8	0	< 1	0.02

*The smallest and greatest number of fatalities (KS_{min} and KS_{max}) result from one of 115 x 36 x 19 = 78,660 considered accident sequences.

Note: For b% of cases, the collective injuries are less than $\overline{KS}$; for c% they are greater than $\overline{KS}$.

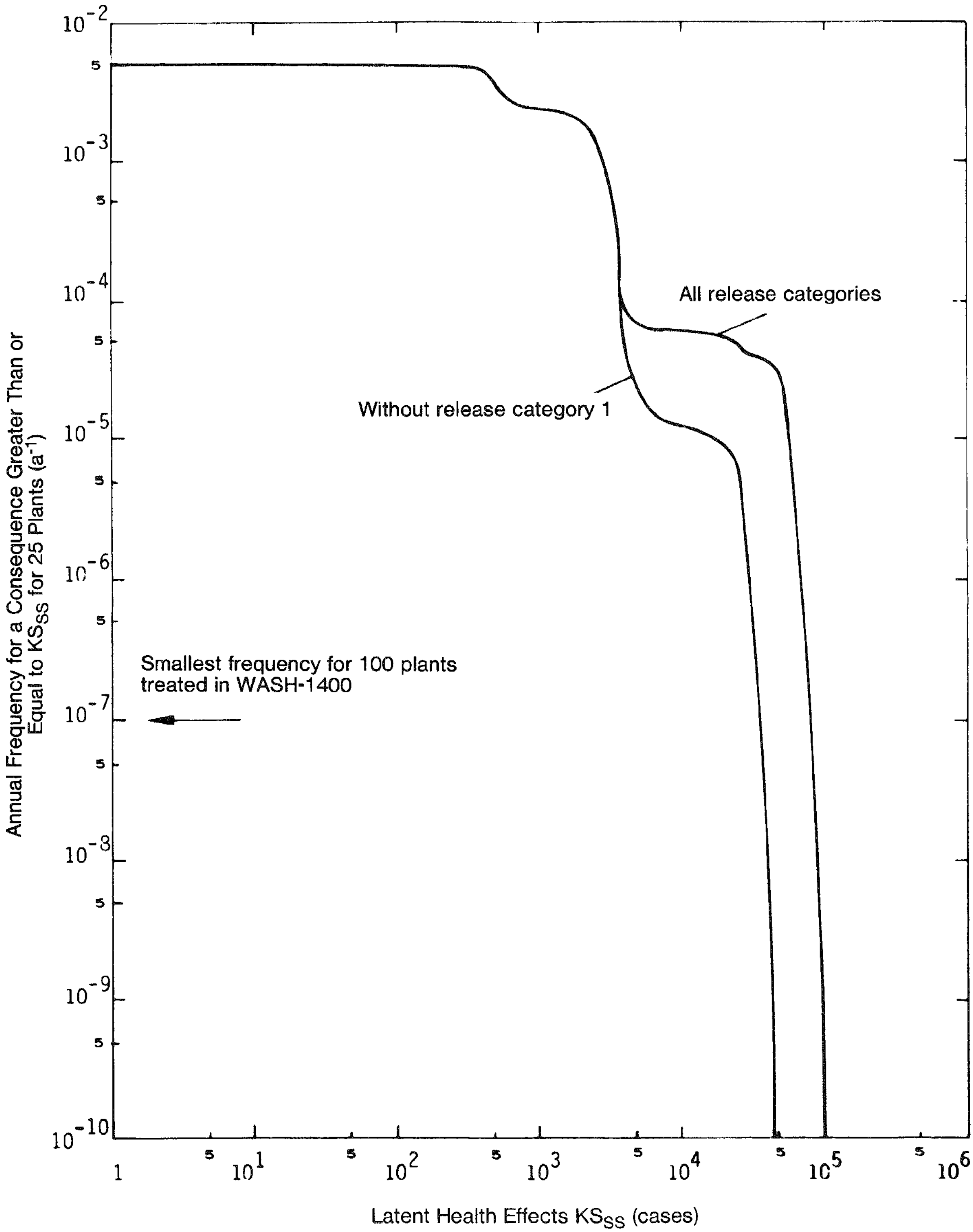


Figure 8-6. Complementary cumulative distribution frequency for latent health effects

The overall complementary cumulative distribution functions (CCDFs) for latent cancer fatalities are illustrated in Figure 8-6, both with and without inclusion of release category FK1.

To interpret these figures we will give a few numerical values. The values for the frequency of latent cancer fatalities greater than or equal to KS can be seen in Table 8-6.

The figures show all calculated results up to the maximum numbers. If we include release category FK1 ("steam explosion"), then we have about 104,000 fatalities; excluding release category FK1, we have about 44,000 fatalities. To be sure, the calculated frequencies of maximum latent cancer fatalities agree with those of the maximum early deaths, but we should note that for latent cancer fatality, the numbers of deaths are greater, even for a significantly higher frequency of occurrence (see Table 8-6).

In the study, a large number of latent cancer fatalities was always determined when--after large releases--the weather conditions were such that the radioactivity concentrations were so low over relatively large regions that protective actions and countermeasures were not initiated.

Table 8-7 shows that only for release categories FK1-FK3 are the latent cancer fatalities greater for persons receiving radiation doses over 5 rem than for under 5 rem. For all other release categories--especially for FK7, which provides the greatest contribution to the calculated risk for latent cancer fatality (see also Table 8-14)--about 90% of the calculated fatalities result from accident-induced radiation doses, which are smaller than the radiation dose received through natural exposure in the course of one's life.

Early fatalities and latent cancer fatalities of the same frequency may not be added since those deaths, contributing to the same frequency in the distributions, generally belong to different accident sequences. This is particularly true for large consequences. In those cases where the number of early deaths is large, the number of latent cancer deaths is relatively small, and vice-versa.

Comparison with Fatalities due to Leukemia and Cancer due to Natural and Man-Made Causes

Leukemia and cancer occur even without the effects of accidental radiation, so that a comparison with the normal frequency of fatalities due to leukemia and cancer caused by natural and man-made radiation is possible.

Table 8-6. Selected points on the CCDFs for latent somatic health effects for 25 units

Occurrence frequency per year	Collective injury ≥ KS	
	Including FK 1 (steam explosion) KS	Excluding FK 1 (steam explosion) KS
1/ 190	1	1
1/ 1,000	2,700	2,700
1/ 10,000	3,900	3,900
1/ 100,000	54,000	20,000
1/ 1,000,000	65,000	31,000
1/ 10,000,000	72,000	36,000
1/ 100,000,000	83,000	41,000
1/ 1,000,000,000	94,000	44,000

Table 8-7. Percentage of fatalities due to cancer and leukemia caused by accident radiation dose below or above 5 rem

Release Category	Percent fatalities due to radiation dose	
	(Greater than 5 rem)	(Fewer than 5 rem)
FK 1	95	5
FK 2	67	33
FK 3	33	67
FK 4	11	89
FK 5	5	95
FK 6	2	98
FK 7	11	89

As determined in the Introduction, the number of latent fatalities for the nuclear power plant accidents examined in the study is governed primarily by small doses received by a large population. To this extent, simply by using the same risk-dose relation, a comparison can be drawn between fatalities due to leukemia and cancer that have their origin in natural radiation exposure.

On the basis of extensive radioactivity dispersion, on the average, about half of the latent fatalities calculated in the study would occur outside the boundaries of the FRG. For this reason, we use the population of Europe as a basis for these comparisons.

The contribution of leukemia and cancer to all natural and man-made fatalities is about 20% (1). Therefore, using the total population of Europe (670 million) we obtain the following number of fatalities due to this cause:

$$6.7 \times 10^8 \times 0.2 = 1.34 \times 10^8$$

Given an average life expectancy of 71 years (1) there results:

$$6.7 \times 10^8 \times 0.2 \times (1/71) = 1.89 \times 10^6$$

fatalities per year caused by leukemia or cancer.

The number of fatalities caused by leukemia and cancer due to natural radiation exposure of 0.1 rem/year over an average life span of 71 years, when correlated to the 670 million population becomes:

$$6.7 \times 10^8 \times 0.1 \text{ rem/year} \times 1.25 \times 10^{-4}\text{/rem} \times 71 \text{ year} = 595{,}000$$

fatalities with 1.25×10^{-4}/rem risk coefficient for whole body radiation (see Section 7.3).

Natural radiation exposure for a calendar year accordingly causes:

$$6.7 \times 10^8 \times 0.1 \times 1.25 \times 10^{-4}\text{/rem} = 8{,}400 \text{ fatalities per year}$$

through the mechanisms of leukemia or cancer.

This result is based on the dose-risk relation for latent cancer fatalities as applied in the study.

The numerical values for the frequency and magnitude of latent cancer fatalities given in Tables 8-5 and 8-6 and in Figures 8-4 and 8-6 should therefore be compared with the normally occurring 1.89 million fatalities per year due to leukemia or cancer and with the 8400 fatalities per year due to leukemia or cancer caused only by natural radiation exposure, calculated by the dose-risk relation used in the study (e).

A comparison of the risk (f) of death due to leukemia or cancer from natural and man-made causes with the risk of death due to nuclear power plant accidents in 25 units is shown in Section 8.1.3.

We should point out that this comparison is based on 25 nuclear units located in the FRG. This picture can be completed only when corresponding studies are also available for other European nations operating nuclear power plants.

8.1.2.3 Genetic Effects (Genetically Significant Collective Dose). We will not present the probability densities for consequence classes or the CCDFs broken down by release categories (see appendix), since this provides a picture similar to that of latent cancer fatalities (Figures 8-4 and 8-5). The characteristic quantities of these individual distributions are listed in Table 8-8.

The average genetically significant collective doses occurring for these release categories in the course of several decades after the accident are summarized below (see Table 8-9).

Table 8-8. Characteristic quantities of the CCDFs of the genetically significant collective dose for 25 units

Release category	Smallest collective dose KD_{min} (man-rem)	Greatest collective dose KD_{max} (man-rem)	Average collective dose $\overline{KD}$ (b/c)* (man-rem)
FK 1	20.0×10^6	420×10^6	260×10^6 (35.1%/64.9%)
FK 2	6.3×10^6	280×10^6	140×10^6 (44.1%/55.9%)
FK 3	0.6×10^6	78×10^6	32×10^6 (49.2%/50.8%)
FK 4	0.2×10^6	23×10^6	8.2×10^6 (55.5%/44.5%)
FK 5	0.3×10^6	10×10^6	2.8×10^6 (56.8%/43.2%)
FK 6	0.2×10^6	7×10^6	1.3×10^6 (61.5%/38.5%)
FK 7	1×10^6	54×10^6	18×10^6 (50.5%/49.5%)
FK 8	$< 1 \times 10^3$	1×10^3	0.14×10^3 --

* In b% of cases the collective dose is less than $\overline{KD}$; in c% it is greater than $\overline{KD}$.

Table 8-9. Average genetically significant collective dose for various release categories

Release category	Average genetically significant collective dose (man-rem)
FK 1	2.6×10^8
FK 2	1.4×10^8
FK 3	3.2×10^7
FK 4	8.2×10^6
FK 5	2.8×10^6
FK 6	1.3×10^6
FK 7	1.8×10^7
FK 8	1.4×10^2

Figure 8-7 illustrates the summation curves of the CCDFs, both with and without considering release category FK1 ("steam explosion"). The interpretation of this figure follows the interpretation of the corresponding figure for latent deaths. All calculated results up to the calculated maximum genetically significant collective dose are entered. Including release category FK1, this is about 4.2×10^8 man-rem, excluding release category FK1, it is about 2.8×10^8 man-rem. The frequency of maximum genetically significant population dose results in a manner analogous to the discussions of early deaths. As in latent cancer fatalities, large genetically significant population doses are always calculated when--after large releases--the weather conditions were such that the radioactivity concentrations were so low over relatively large regions that protective actions and countermeasures were not initiated.

Comparison to the Genetically Significant Population Dose From Natural Radiation

By using the same reasoning given in Section 8.1.2.2 for latent somatic health effects--that the collective dose is determined primarily by a large population receiving small doses--a comparison with the genetically significant collective dose due to natural radiation exposure can also be drawn here.

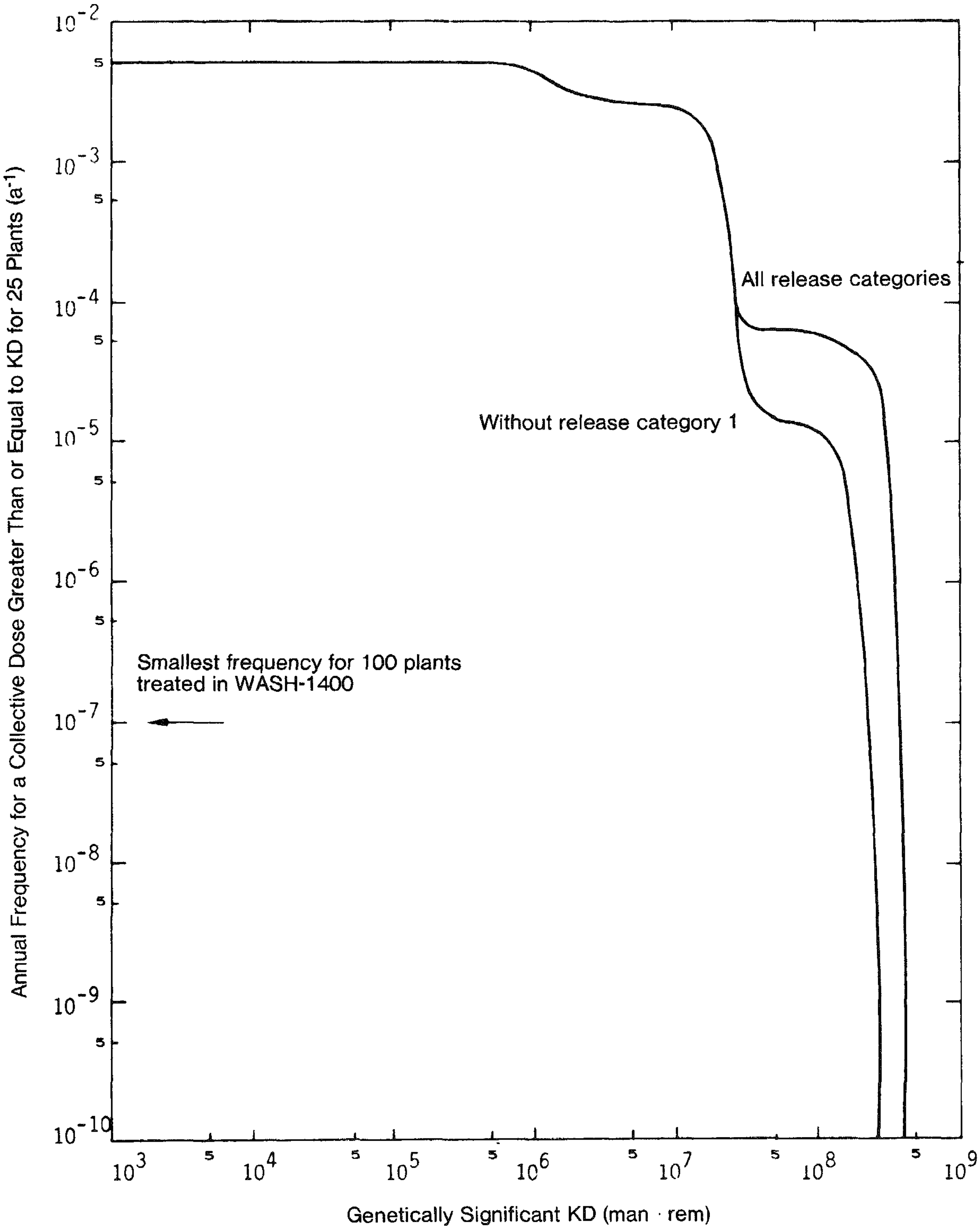


Figure 8-7. CCDF of the genetically significant collective dose

The genetically significant natural dose to the population presently living in Europe is

$$6.7 \times 10^8 \times 0.1 \text{ rem/yr} = 6.7 \times 10^7 \text{ man-rem per year}$$

on the basis of the average natural radiation exposure of 0.1 rem/yr.

The calculated results given in Tables 8-8 and 8-9 for the genetically significant collective dose and its frequency should therefore be compared with this numerical value.

Section 8.1.3 compares the expected genetically significant collective dose from natural radiation with the expected genetically significant collective dose from core melt accidents at 25 units.

The same restrictions in Section 8.1.2.2, "Latent Somatic Health Effects," which were established with regard to nuclear power plants located in other European countries, are also valid here.

8.1.2.4 Areas and Persons Affected by the Countermeasures "Evacuation," "Fast Relocation" and "Resettlement." In Section 7.4, "Model of Protective Actions and Countermeasures," the following countermeasures were described among others:

- evacuation (Region A)
- fast relocation (Regions B_1 and B_2)
- resettlement (Region C).

The results calculated for areas and persons affected by these measures were determined in the usual manner as CCDFs. Characteristic quantities are illustrated in Tables 8-10 to 8-12.

Evacuation (Region A)

The study assumes for all releases under discussion here that evacuation takes place in a region measuring F = 33.3 km^2. In two-thirds of all cases, the number evacuated is fewer than 6800 persons. Additional details are found in Table 8-10.

Table 8-10. Characteristic quantities of the CCDFs for the areas F and persons P affected in region A by the countermeasure "evacuation"

Release category	Probability for	Surface area			Person P		
		Smallest area	Largest area	Average area	Smallest number of persons	Greatest number of persons	Average number of persons
	F>0 for P>0	F_{min}	F_{max}	$\overline{F}$	P_{min}	P_{max}	$\overline{P}$ (b/c)*
FK 1...8	100%		33.3		360	42,000	6,800 (ca. 60%/ ca.40%)

Note: All probabilities given in this table are conditional probabilities, i.e., a radioactive release is presumed.

* In b% of cases the number of persons is less than $\overline{P}$; in c% it is greater than $\overline{P}$.

Fast Relocation (Regions B_1 and B_2)

The characteristic quantities of the CCDFs for persons and areas affected by this countermeasure are listed in Table 8-11. This table shows that only in release categories FK1, FK2, and FK3 will radiation doses occur outside of Region A that lead to a "fast relocation" according to the findings of the study.

Considering the total frequency of release categories with preceding core melts FK1 to FK6, only 1% of the releases of these categories affect formation of regions B_1 + B_2.

Other details can be found in Table 8-11.

Resettlement (Region C)

The characteristic quantities of the CCDFs for persons and areas affected by this countermeasure are shown in Table 8-12.

Table 8-11. Characteristic quantities of the CCDF for the areas F and persons P affected in regions B_1 and B_2 by the countermeasure "rapid relocation"

Release category	Annual frequency of regions B_1 and B_2 occurring for 25 plants	Probability of occurrence of a region $B_1 + B_2$ after a postulated release	Affected areas F [km^2]			Persons P		
			Smallest areas $F_{min}(a)$*	Largest areas F_{max}	Average surface area $\overline{F}$ (b/c)**	Smallest number of persons $P_{min}(a)$*	Largest number of persons P_{max}	Average number of persons $\overline{P}$ (b/c)**
FK 1	1.9×10^{-5}	37.8%	0(62.2%)	379	20 (80.5%/19.5%)	0(62.2%)	1,010,000	5,200 (83.7%/16.3%)
FK 2	6.2×10^{-6}	41.3%	0(58.7%)	125	7.8 (77.4%/22.6%)	0(58.7%)	280,000	2,200 (81.2%/18.8%)
FK 3	7.2×10^{-7}	4.9%	0(95.1%)	4	0.2 (95.1%/4.9%)	0(95.1%)	18,500	55 (95.4%/ 4.6%)
FK 4	0	0	--	--	--	--	--	--
FK 5	0	0	--	--	--	--	--	--
FK 6	0	0	--	--	--	--	--	--
FK 7	0	0	--	--	--	--	--	--
FK 8	0	0	--	--	--	--	--	--

* A% of results leads to the smallest area F_{min} or to the smallest number of persons P_{min}.

** In B% of cases the area is less than $\overline{F}$ or the number of persons less than $\overline{P}$; in C% the area is greater than $\overline{F}$ or greater than $\overline{P}$.

Table 8-12. Characteristic quantities of the CCDF for the areas F and persons P affected in region C by the countermeasure "relocation"

Release category	Annual frequency, of formation of a region C for 25 plants	Probability of occurrence of a region C after a postulated release	Affected areas			Persons P		
			Smallest areas $F_{min}(a)$*	Largest areas F_{max}	Average surface area $\overline{F}$ (b/c)**	Smallest number of persons $P_{min}(a)$*	Largest number of persons P_{max}	Average number of persons $\overline{P}$ (b/c)**
FK 1	5×10^{-5}	ca. 100%	0(<0.1%)	5,680	680 (65.7%/34.3%)	0(<0.1%)	2,910,000	180,000 (69.8%/30.2%)
FK 2	1.5×10^{-5}	ca. 100%	0(<0.1%)	1,950	340 (65.0%/35.0%)	0(<0.1%)	2,400,000	90,000 (71.9%/28.1%)
FK 3	1.4×10^{-5}	92.9%	0(7.1%)	230	30 (68.2%/31.8%)	0(7.1%)	660,000	7,600 (76.7%/23.3%)
FK 4	4.5×10^{-5}	58.3%	0(41.7%)	13	2.7 (58.0%/42.0%)	0(41.7%)	36,000	600 (77.3%/22.7%)
FK 5	9.3×10^{-5}	18.4%	0(81.6%)	2	0.3 (87.0%/13.0%)	0(81.6%)	9,100	50 (51.1%/48.9%)
FK 6	1.1×10^{-5}	0.6%	0(99.4%)	2	0.01 --	0(99.4%)	7,600	2 (99.4%/ 0.6%)
FK 7	1.7×10^{-3}	66.5%	0(33.5%)	49	4.7 (75.0%/25.0%)	0(33.5%)	150,000	1,100 (78.4%/21.6%)
FK 8	0	0%	--	--	-- --	--	--	-- --

* A% of results leads to the smallest area F_{min} or to the smallest number of persons P_{min}.

** In B% of cases the area is less than $\overline{F}$ or the number of persons less than $\overline{P}$; in C% of cases it is greater than $\overline{F}$ or greater than $\overline{P}$.

According to this table, in all release categories except FK8 weather sequences occur in which there are potential radiation doses greater than those assumed as the criteria for resettlement.

With regard to the total frequency of release categories with preceding core melts FK1 to FK6 (Table 8-1), region C will be resettled in only 9% of the releases of these categories.

Table 8-12 indicates that the criterion in the study for initiating resettlement can affect a very large number of people in an extreme case. Since numbers of this magnitude are anticipated only for large cities and congested areas, the figures cited are too high for the following reasons:

- The study used shielding factors valid for an average mix of large, medium, and small houses. In contrast, in large cities and congested areas, tall, multi-story buildings are more frequent, and these afford a better shielding.
- Rainfall will also affect the calculations. In densely populated regions with large areas of roofing, concrete, and asphalt, much of the radioactive material will flow with the rainwater into the sewer system and become much less effective. The study does not contain any model to simulate rainfall runoff; thus, this effect is not taken into account.
- The study provides for resettlement even after decontamination if the accumulated whole body dose exceeds 12.5 rad (25 rad potential whole body dose) for 30 years of normal activity. This value corresponds to twice the annual dose limit for occupationally exposed persons and is thus relatively low. In the course of phase B of the present study, we will consider in detail whether such a large population movement is justified for such a low initiating dose.

The first two deficiencies of the model cause an overestimation of doses in large cities and congested areas. Their correction will reduce the number of persons affected without changing the countermeasure model. In addition, it seems appropriate to check the basic criteria for resettlement.

8.1.3 Collective and Individual Risks and Mean Collective Doses

In addition to the complementary cumulative distribution functions and probability densities for consequence classes of collective injuries and collective doses, illustrated in Section 8.1.2, the expected values of these quantities are also of interest.

The expected value of the collective injury <KS> (g) (this is the risk; see Chapter 2) denotes the mean injury per reactor-year; the expected value of the collective dose <KD> gives the mean collective dose.

The expected value of the collective injury (collective risk) due to nuclear power plant accidents from 25 reactor units $<KS^{25}>$ (or for the collective dose $<KD^{25}>$) is formed by aggregating the injuries (or collective doses) weighted by the frequencies. The summation is performed for all accident situations, i.e., for all release categories, weather situations, and population distributions.

Table 8-13 gives the resulting expected collective injury and collective doses according to individual release categories. Accordingly, the total risk for acute death is:

$$<KS^{25}_{FS}> = 1 \times 10^{-3} \text{ per year;}$$

for leukemia or cancer, it is:

$$<KS^{25}_{SS}> = 10.1 \text{ per year.}$$

The total expected value for the genetic dose is:

$$<DK^{25}> = 6.6 \times 10^{4} \text{ man-rem per year.}$$

The individual release categories FK contribute the following fractions to the aggregate values (see Table 8-14).

This compilation indicates that release categories FK1 and FK7 contribute the greatest fractions to the overall expected latent health effects and genetically significant dose. The main contribution comes from release category FK7. Accidents that fall into this category include a LOCA controlled by the emergency cooling systems so that no severe damage to fuel elements is expected, but a failure of the isolation valves of the containment occurs. As discussed in

Table 8-13. Collective risks and expected values of collective doses for 25 units

	Collective risk $\langle KS^{25} \rangle$ (a^{-1})			Average genetically significant collective dose $\langle KD^{25} \rangle$
	Early health effects $\langle KS^{25}_{FS} \rangle$		Latent health effects $\langle KS^{25}_{SS} \rangle$	
Release category	German study dose-effect relation-ship	American study dose-effect relation-ship		[man-rem a^{-1}]
FK 1	4.7×10^{-4}	4.1×10^{-4}	2.4	1.3×10^{4}
FK 2	4.8×10^{-4}	4.3×10^{-4}	3.3×10^{-1}	2.1×10^{3}
FK 3	3.1×10^{-5}	2.5×10^{-5}	7.4×10^{-2}	4.8×10^{2}
FK 4	2.9×10^{-5}	2.3×10^{-5}	1.2×10^{-1}	6.2×10^{2}
FK 5	0	0	3.3×10^{-1}	1.4×10^{3}
FK 6	0	0	8.4×10^{-1}	2.6×10^{3}
FK 7	0	0	6.0	4.6×10^{4}
FK 8	0	0	4.8×10^{-4}	3.4
Summed	1.0×10^{-3}	8.9×10^{-4}	10.1	6.6×10^{4}
Summed without Category 1	5.4×10^{-4}	4.8×10^{-4}	7.7	5.3×10^{4}

Table 8-14. Relative contribution of release categories to the overall expected risks

Release category	Risk of health effect type		
	Acute fatalities	Latent cancer fatalities	Genetically significant dose
FK 1	46.5%	24.0%	19.9%
FK 2	47.5%	3.3%	3.1%
FK 3	3.1%	0.7%	0.7%
FK 4	2.9%	1.2%	0.9%
FK 5	--	3.3%	2.1%
FK 6	--	8.3%	3.9%
FK 7	--	59.3%	69.3%
FK 8	--	5×10^{-3}%	5×10^{-3}%

Chapter 6, the frequency for this release category was estimated to be 3×10^{-7} per reactor-year. It was also mentioned that in accordance with the objective of phase A of the study and as in WASH-1400, a 10% carry-over of release frequency from the neighboring release category FK8 was performed. For release category FK7, regardless of the detailed analysis, the release frequency was increased by more than two orders of magnitude. Since the study estimated the frequency of LOCA controlled by emergency cooling facilities to be 1×10^{-3}/year, this result could be interpreted as a failure of the containment isolation valve as postulated on every tenth LOCA. Neither operating experience nor detailed system analysis shows justification for such a pessimistic assumption.

Were we to assume, for instance, that the containment isolation valve fails once in 100 successfully controlled LOCAs, then the expected value for latent health effects would be halved by this fact alone. It will therefore be necessary to consider this state of affairs in a more realistic manner in phase B of this study.

The accident-caused collective risk from latent health effects can be compared with the expected values for leukemia and cancer caused by natural and man-made radiation.

The collective risk of death due to leukemia and cancer from nuclear power plant accidents in 25 units of $\langle KS_{SS}^{25} \rangle = 10.1$ per year is compared with the expected value for normally occurring leukemia and cancer to $\langle KSnat \rangle = 1{,}890{,}000$ per year (see Section 8.1.2.2) or, due to natural radiation exposure of $\langle KSnat\ rad \rangle = 8{,}400$ per year (see Section 8.1.2.2).

The mean genetically significant collective dose due to nuclear power plant accidents for 25 units of $\langle KD^{25} \rangle = 6.6 \times 10^4$ man-rem per year may be compared with the genetically significant collective dose due to natural radiation exposure of $\langle KDnat\ rad \rangle = 6.7 \times 10^7$ man-rem per year (see Section 8.1.2.3).

From this comparison, it is clear that the estimated collective risks due to nuclear power plant accidents in 25 units lies several orders of magnitude below that due to natural radiation exposure.

In addition to the collective risk, the distance-dependent average individual risk is calculated.

The average individual risk (expected value for the individual injury <S>) is stated as the average individual injury per reactor-year. It is understood to be the average value for all persons located the same distance from a nuclear power plant.

The distance-dependent individual risk is an expected value standardized to one reactor unit, formed from the location-dependent individual injury weighted by the frequencies. Again, all release categories, as well as weather sequences and population distributions, were considered.

In Figures 8-8 and 8-9, the distance-dependent individual risks for early and latent health effects are shown by release category. The curves for the individual release categories run approximately parallel. For the reasons already discussed in Section 8.1.2.1, early deaths are limited to the proximity of the site. Therefore, they exhibit a particularly steep decrease with increasing distance. The expected value for latent cancer fatality is less steep, i.e., it decreases almost inversely proportional to distance.

The aggregate curves of individual risk are illustrated in Figure 8-10, both with and without inclusion of release category FK1 ("steam explosion"). As we can see from this figure, individual risks for early death are far below those for death due to leukemia or cancer.

The distance-dependent individual risk for latent health effects illustrated in Figure 8-10 can be related to the location-independent risk of normally occurring leukemia and cancer. For an average life expectancy of 71 years, this becomes:

$\langle S_{nat} \rangle = 0.2 \times (1/71 \text{ yr}) = 2.8 \times 10^{-3}$ per year (see Section 8.1.2.2).

The expected value for leukemia and cancer due to natural radiation exposure is:

$\langle S_{nat\ rad} \rangle = 0.1 \text{ rem/yr} \times 1.25 \times 10^{-4}/\text{rem} = 1.25 \times 10^{-5}$ per year (see Section 8.1.2.2).

These values are also shown in Figure 8-10. We can see that the individual risks due to reactor accidents lie far below the corresponding risks for leukemia and cancer due to natural or man-made causes.

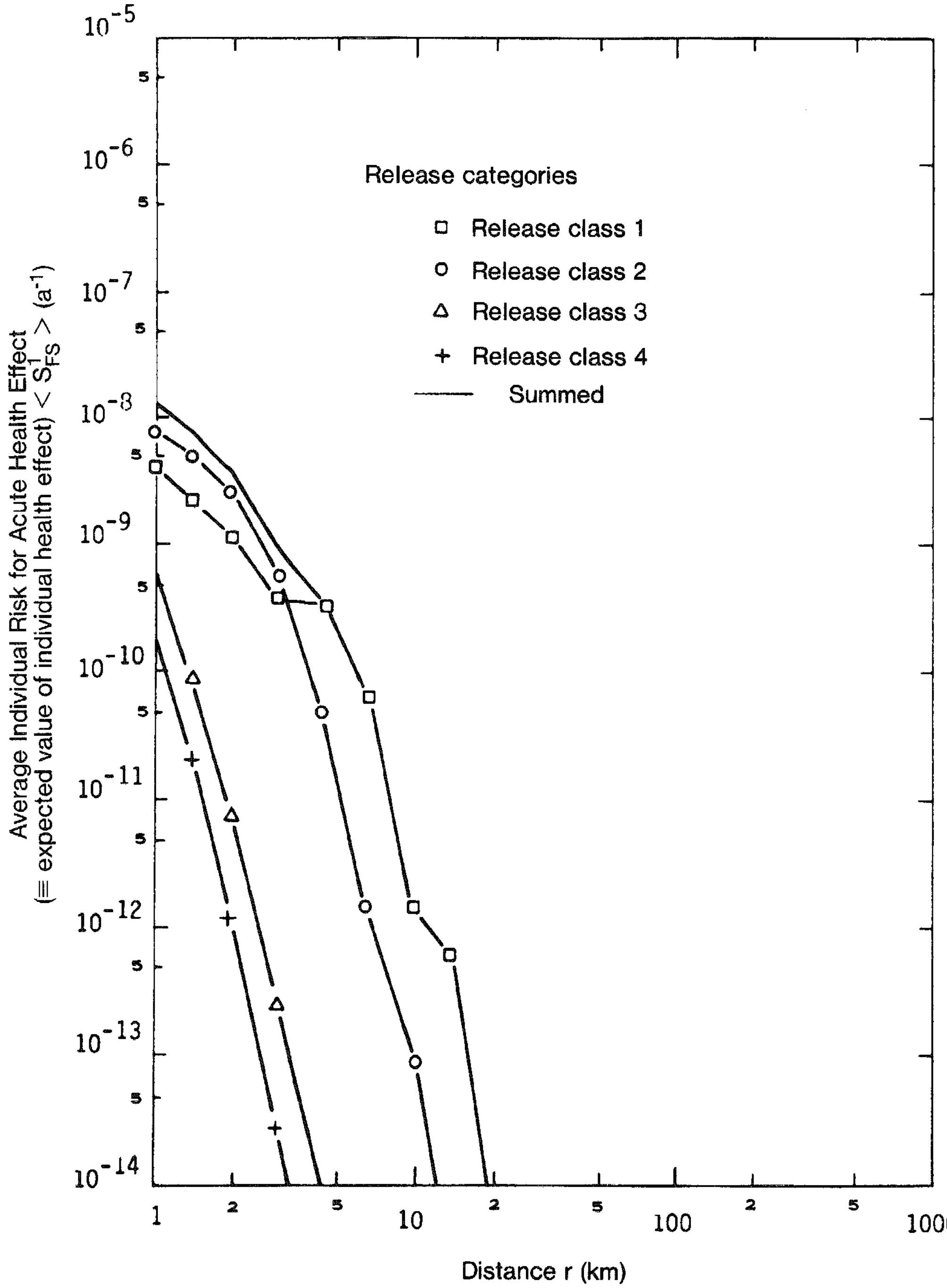


Figure 8-8. Expected probability of acute health effect for an individual as function of distance from plant (equals average individual risk for acute health effect), normalized to one plant and keyed according to release categories (German study dose-effect relationship)

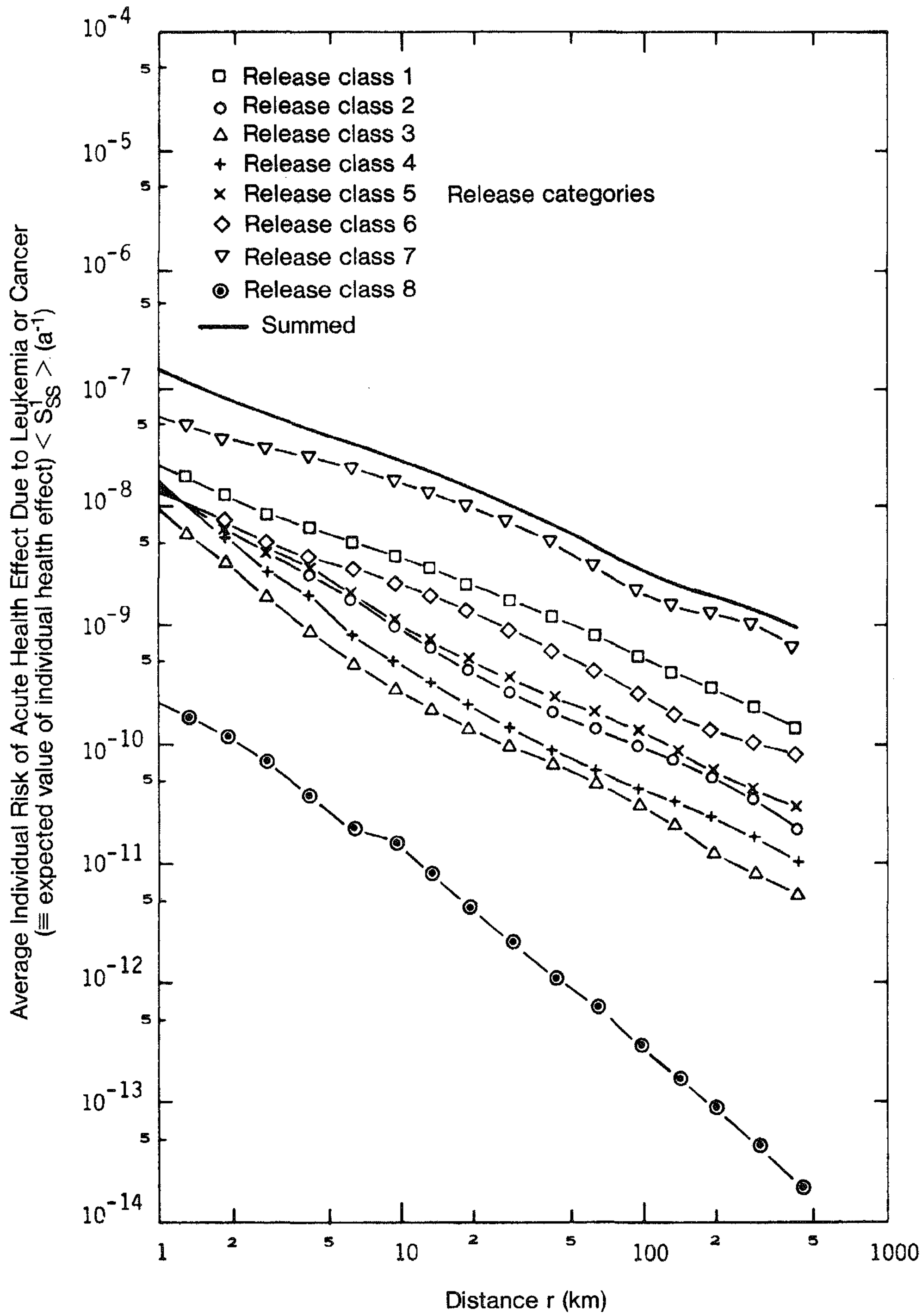


Figure 8-9. Expected probability of latent health effect for an individual as a function of distance from the plant (equals average individual risk of acute health effect due to leukemia or cancer), normalized to one plant and keyed according to release categories

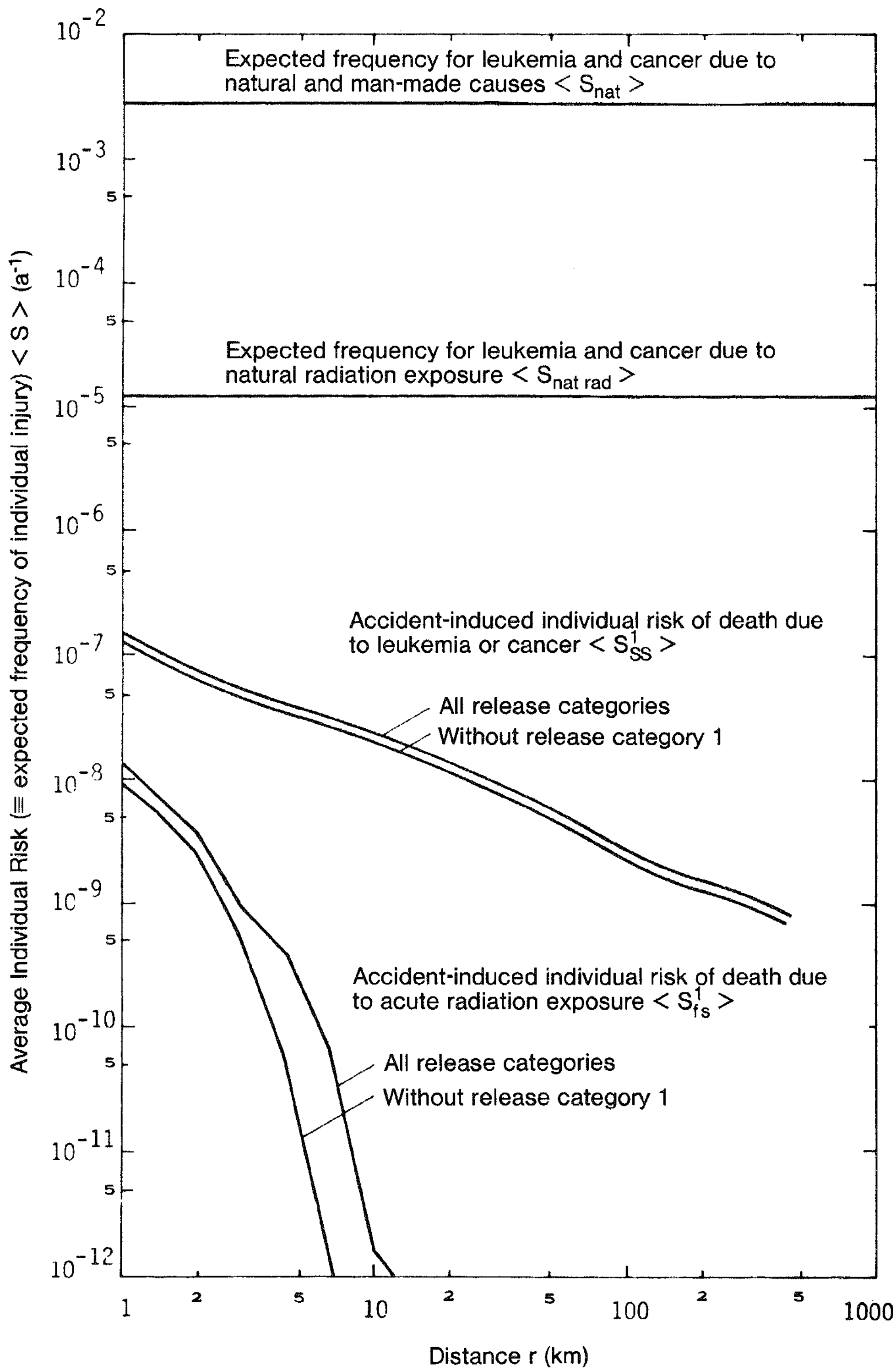


Figure 8-10. Expected probability versus distance for individual injury (equals average individual risk), normalized to one plant, and comparison with individual risks based on natural and man-made causes.

8.1.4 Substance of Findings

The objective of this study and the substance of its findings were already discussed in the first chapter. In order to avoid misinterpretation, this subject will once again be addressed with direct reference to the presentation of results.

In accordance with the contract, the parameters and models of the American study WASH-1400 were generally adopted in phase A. This concept required modification at several points (system technology, weather, and population data, etc.), but in principle it was followed and often retained, even when more suitable parameter values and models (ingestion model, rain run-off model, etc.) were already known.

The characteristics of the accident consequence model and the uncertainty of specific input data do not permit the results of this study to be applied to specific nuclear power plant sites. For instance, the same weather sequences were used for all sites within a region, even though the distances are sometimes considerable and the site-specific peculiarities of the traffic system were excluded in the evacuation model because inclusion would have required thousands of complicated evacuation simulations. Because of the statistical character of the results, no deterministic statements about the occurrence of a particular injury or individual destiny can be derived.

This study serves to estimate the collective risk attached to the operation of nuclear power plants with 25 light water reactors in the FRG. The studies are methodologically similar to the American Reactor Safety Study (WASH-1400). The accident consequence model satisfies these objectives and boundary conditions. The decoupling from the American model (WASH-1400), the improvement of individual submodels, and the utilization of improved input data corresponding to the present state of technology are retained for phase B of this study.

8.2 VALIDITY OF THE RESULTS

Figures 8-3 and 8-6 indicate for each magnitude of consequences X the expected frequency per year of consequences greater than or equal to X to be caused by 25 units of the type under discussion (h). In the case of latent heath effects, the calculated injury occurs at the given annual frequency, but its effects are not noticed for years. For rare events, the present method of risk presentation separating the frequency and magnitude of consequences is required. Therefore, the risk is expressed both by the expectation (sum of the products of frequency and magnitude of the individual consequences) and by means of the so-called complementary cumulative distribution function of the consequence. We use the complementary CDF because it denotes the frequency of consequence greater than or equal to X, whereas the CDF itself denotes consequences less than or equal to X.

The curves represent about 600,000 different simulated accident sequences. Each accident sequence consists of:

- The plant-internal event sequence, which runs from the initiating event to the release, and
- The plant-external exposure sequence, which includes dispersion and deposition of pollutants, local distribution of exposed persons, and harmful effects, as well as protective actions and countermeasures (see Figure 2-6).

To reduce computer time, the results of the event sequences were summarized for purposes of accident simulation into eight release categories. Each simulated accident sequence thus combines the representative characteristics of the particular release category and one exposure sequence.

The frequency of a certain consequence magnitude X from the CCDF is the sum of frequencies of those simulated accidents that cause consequences greater than or equal to X. Both the expected frequency of the simulated accident and its estimated consequences are affected by uncertainties in estimation (see Section 4.7). For example, if we select a different value for a fixed but inaccurately known quantity or a different functional description for an inaccurately known phenomenon, then one or several combinations of release and exposure sequence yield a different frequency or a different consequence magnitude, and thus also a different CCDF. For a given combination of release and exposure sequences, because of uncertainties, only regions in the frequency/consequence diagram (see Figure 2-7) where the contribution of this combination to the CCDF will be within a certain confidence level (i) can be given. For a given confidence of 90%, for instance, we obtain a band (called the global 90% confidence interval) in which the particular CCDF will run with just this confidence, provided all unquantified uncertainties in the estimation are negligible. For the sake of simplicity, we did not estimate global confidence intervals for the resulting curves (Figures 8-3 and 8-6); rather, at specific points along the cumulative complementary distribution function we found appropriate local confidence bands. Whereas global intervals indicate boundary lines between which the entire particular CCDF will run with a given confidence, from local bands we can find the following information for fixed (thus local) values of frequency or magnitude of consequences:

- 90% confidence interval for frequency H: for a fixed magnitude of consequence X'; this is the range in which it is 90% certain that the particular frequency will be found with a consequence magnitude greater than or equal to X' (vertical, dashed line in Figure 8-11).
- 90% confidence interval for consequence magnitude X; this denotes the range for a fixed frequency H* in which there is a 90% probability that the particular consequence magnitude will equal or exceed the frequency H* (horizontal, dashed line in Figure 8-11).

Extensive, complex risk estimations of rare events are generally affected by many uncertainties. Section 8.2.1 lists estimation uncertainties judged important and therefore quantified.

Section 8.2.2 briefly digresses on the methodology to convert these uncertainties into local subjective confidence intervals along the CCDF of consequence. Section 8.2.3 illustrates and explains the resulting confidence intervals. Sections 4.7 and 8.2.1 show why the uncertainty or confidence interval is considered subjective here.

8.2.1 Quantified Uncertainties in the Assessment

Besides release frequencies, uncertainties were quantified by assigning so-called fractile values. Accordingly, with the 90% fractile, e.g., dry atmospheric fallout rate of iodine, we state that, according to expert opinion, there is 90% certainty of the "best value" of this quantity being below 0.05 m/sec. By "best value" we mean that single fixed value which, in the opinion of the experts, best befits the example analysis (in this case the dispersion calculation). Fractile values of 10%, 50%, etc., are interpreted analogously. The fractiles of the expected release frequencies are derived from distributions obtained by Monte-Carlo simulation from functions of several probabilities and frequencies subject to estimation uncertainties (see Section 4.7.2).

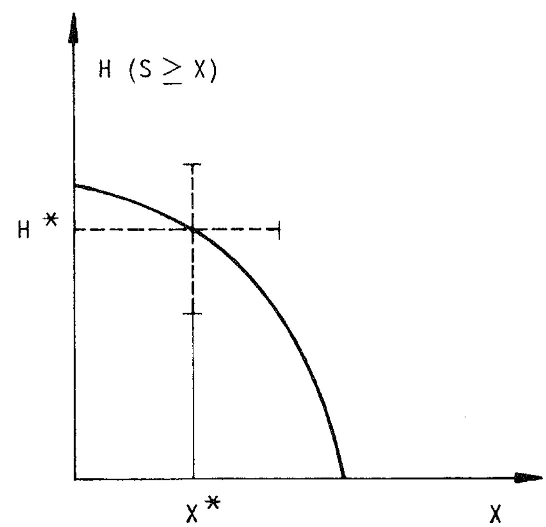


Figure 8-11. Illustration of local uncertainty bands on frequency and consequences

For the individual, quantified estimation uncertainties:

- release frequencies (j) (see table)

Release category	5%	Fractile 50% (median)	95%	Reference value (expectation)
FK 1	4. E-8	4. E-7	7. E-6	2. E-6
FK 2	7. E-8	3. E-7	2. E-6	6. E-7
FK 3	8. E-8	3. E-7	2. E-6	6. E-7
FK 4	3. E-7	1. E-6	9. E-6	3. E-6
FK 5	2. E-6	9. E-6	7. E-5	2. E-5
FK 6	5. E-6	2. E-5	2. E-4	7. E-5
FK 7	9. E-6	6. E-5	4. E-4	1. E-4
FK 8	9. E-5	6. E-4	4. E-3	1. E-3

Note: We estimated the expected annual frequency, averaged over several plants of the type analyzed. The fractiles express the estimation uncertainty of this average value.

- released energy in 10^6 kJ/h (plume rise): see table.

Release category	Fractile 5%	Fractile 50%	Fractile 95%	Reference value
FK 1 --	50	460	4,200	540
FK 2 --	1	5	20	15
FK 5 (Third phase)	30	160	840	200
FK 6 (Third phase)	30	160	840	200
FK 7 --	1	5	20	9

Note: Estimation uncertainty of the "best value".

- calculated plume rise: fractile of the correction factor for the reference value (see table).

10%	50%	90%
0.5	1.0	1.75

Note: The plume rise is calculated by equations that only approximately describe the process. The correction factor is designed to express potential error. The estimation uncertainty of the "best value" of this factor is quantified.

- dry atmospheric fallout rate in m/sec: (see table)

For iodine			For aerosols		
Fractile			Fractile		
10%	50%	90%	10%	50%	90%
0.002	0.01	0.05	0.001	0.005	0.025
Reference value			Reference value		
0.01			0.01		

Note: The fallout rate during an exposure sequence is subject to many, not explicitly defined, random influences. The uncertainty given here is to express the estimated uncertainty of the "best value"

- wet atmospheric fallout; "wash-out" (scavenging) coefficient Λ(1/sec) for various rainfall rates (see table).

Fractile	Rainfall in mm/h		
	0-1	1-3	> 3
10%	2.0 E-5	1.0 E-4	2.0 E-4
50%	1.0 E-4	5.0 E-4	1.0 E-3
90%	5.0 E-4	2.5 E-3	5.0 E-3
Reference value equals 50% fractile			

Note: Even the scavenging coefficient is a "best value" for the corresponding rainfall rate.

The fractile information expresses the uncertainty of the "best value."

- time periods from identification of a release to "going and remaining indoors" expressed in hours: see table.

Fractile		
10%	50%	90%
1.5	2	4
Reference value 2		

Note: Estimation uncertainty of the "best value"

- percent population outdoors before the order "stay indoors."

Fractile		
10%	50%	90%
1	3	9
Reference value 3		

Note: See above.

- percent of population remaining outdoors beyond the time periods given above: (see table)

Fractile		
10%	50%	90%
1	3	6
Reference value 3		

Note: As above.

- dose-effect relationship for early fatalities: see table.

Fractile		
10%	50%	90%
O	B	P
Reference value B		

O : F_O (330) = 1%; F_O (610) = 50%

F_O (x) = 0 for all x with F_O (x) less than F_B (100).

B : F_B (250) = 1%; F_B (510) = 50%

F_B (x) = 0 for x less than 100 rad.

P : F_P (200) = 1%; F_P (410) = 50%

F_P (x) = 0 for all x with (x) less than F_B (100).

Note: In accordance with a cumulative normal distribution designated by its 1% and 50% fractiles, the different dose values in rad are assigned to percentages. We are dealing with expected values whose estimated uncertainties are expressed here by three alternative normal distributions (Figure 8-12).

- dose-effect relationship for latent fatalities: see table.

Fractile		
10%	50%	90%
0.5 x Y	1.0 x Y	2.0 x Y
Reference value Y		

Y = risk coefficient according to ICRP-26 (see Chapter 7).

Note: The risk coefficients are expected values. Their estimated uncertainties are expressed by fractile numbers.

8.2.2 Determination of Subjective Confidence Intervals

We should stress here that only parameter uncertainties (see Section 4.7) contribute to confidence intervals of the risk assessment. However, each uncertainty concerning the outcome of an accident sequence which is based on random variation of components of the accident simulation represents an element of risk. Therefore, it must be expressed by the consequence CCDF or in the numerical value of risk, and not by confidence intervals.

Example: The fraction of the population outdoors (before the evacuation begins) is given by a fixed percentage as a "best value." Since we are dealing with a fixed value for accident simulation, its estimated uncertainty contributes to the confidence interval. If another "best value" is selected, a different CCDF is obtained.

In actual situations, because of numerous random effects, different percentages of the population will be outdoors, and only the average of many accident sequences will yield a fraction of 3%. Were we to include these random variations, each simulated accident sequence would have to be repeated with a whole spectrum of different percentages, and the expected frequency would have to be multiplied by the probability of actual occurrence. Thus, the random variation of this percentage (still averaged over time and place) would be included in the CCDF and would not contribute to the confidence interval. At the same time, the number of simulated accident sequences would multiply.

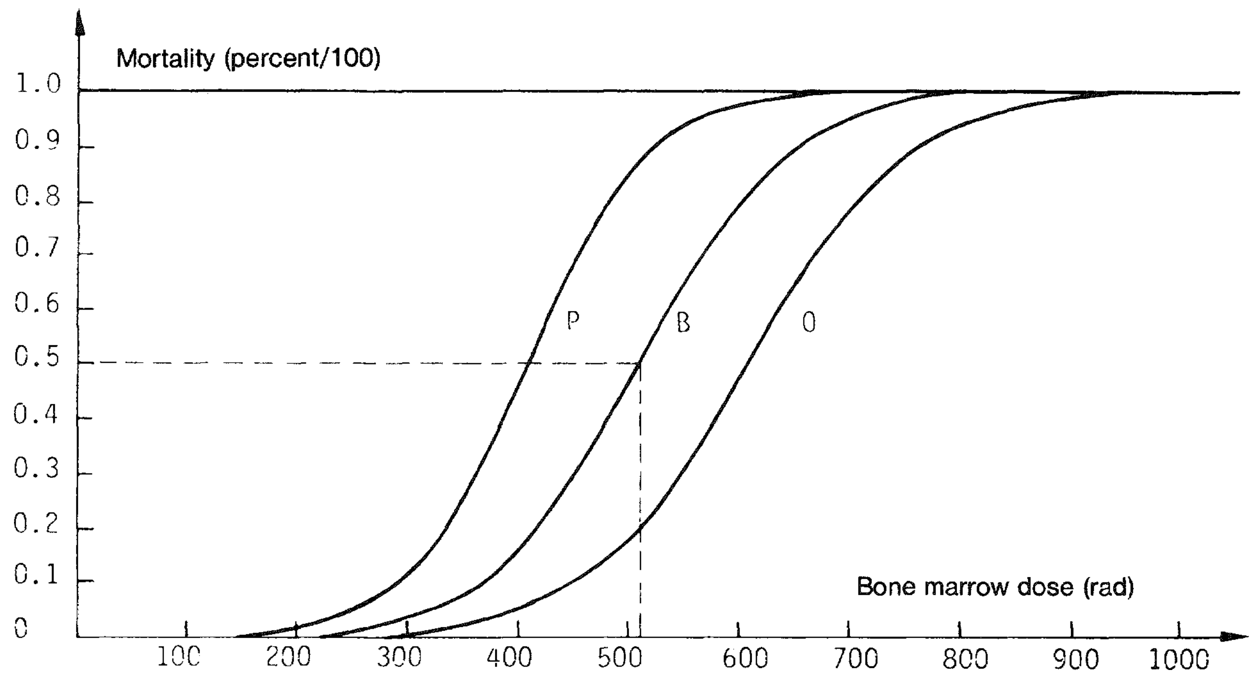


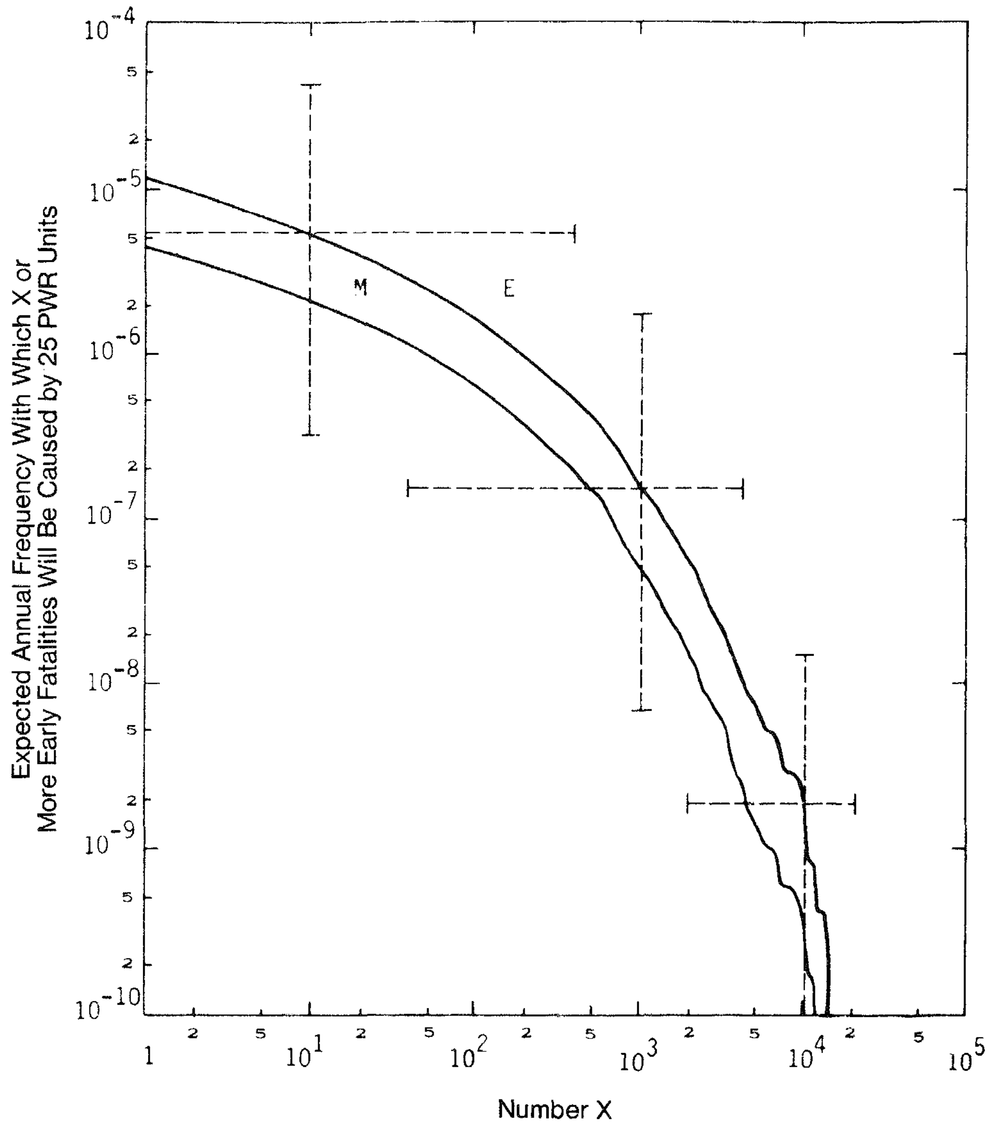
Figure 8-12. Dose-effect relationships P, B, and O for early fatalities

Quantifying the effect of parameter uncertainties in the form of confidence intervals also multiplies the calculations by requiring determination of CCDFs for a relatively large number of randomly selected combinations of values for the variables. To avoid determining these CCDFs through the study's complicated accident consequence model, a so-called response function was constructed approximating change in the consequence CCDF to change in parameter values ($\underline{k}$). With respect to most parameters from Section 8.2.1, this surface response function represents a local, piece-wise linear approximation of the frequency distribution and is accurate with respect to the eight release frequencies. Its coefficients were estimated by means of a defined number of CCDFs from the accident consequence model on the basis of four (corresponding to the different meteorological siting regions) of the 19 sites studied. The response function can be evaluated much faster than the accident consequence model of the study. Its use for a great number of value combinations of the variables can approximately determine the CCDF.

One thousand sets of values, each consisting of one value for each parameter, were randomly selected to estimate the confidence interval in accordance with the parameter distributions. With the exception of parameter pairs, "dry deposition rate and washout coefficient" and "percentage of the population outdoors before and after the signal 'stay indoors'," selection of parameter values was mutually exclusive. The distributions of release frequencies required for this were obtained from quantified estimated uncertainties in failure rates, probabilities, and anticipated frequencies of initiating events, expressed by distributions and propagated to the release frequency via fault trees and event trees. For distributions of the other parameters, log normal distributions with the given 10% and 90%, or 5% and 95% confidence intervals were used. Expert information on the estimated uncertainties of these parameters in the risk calculation is not so detailed that special distributions can be tailored. The result of selecting log normal distribution as a distribution type was that, for most expert opinions, the quotients from the 50% and 5% (or 10%) confidence intervals, and from the 95% (or 90% and 50%) confidence intervals are nearly equal.

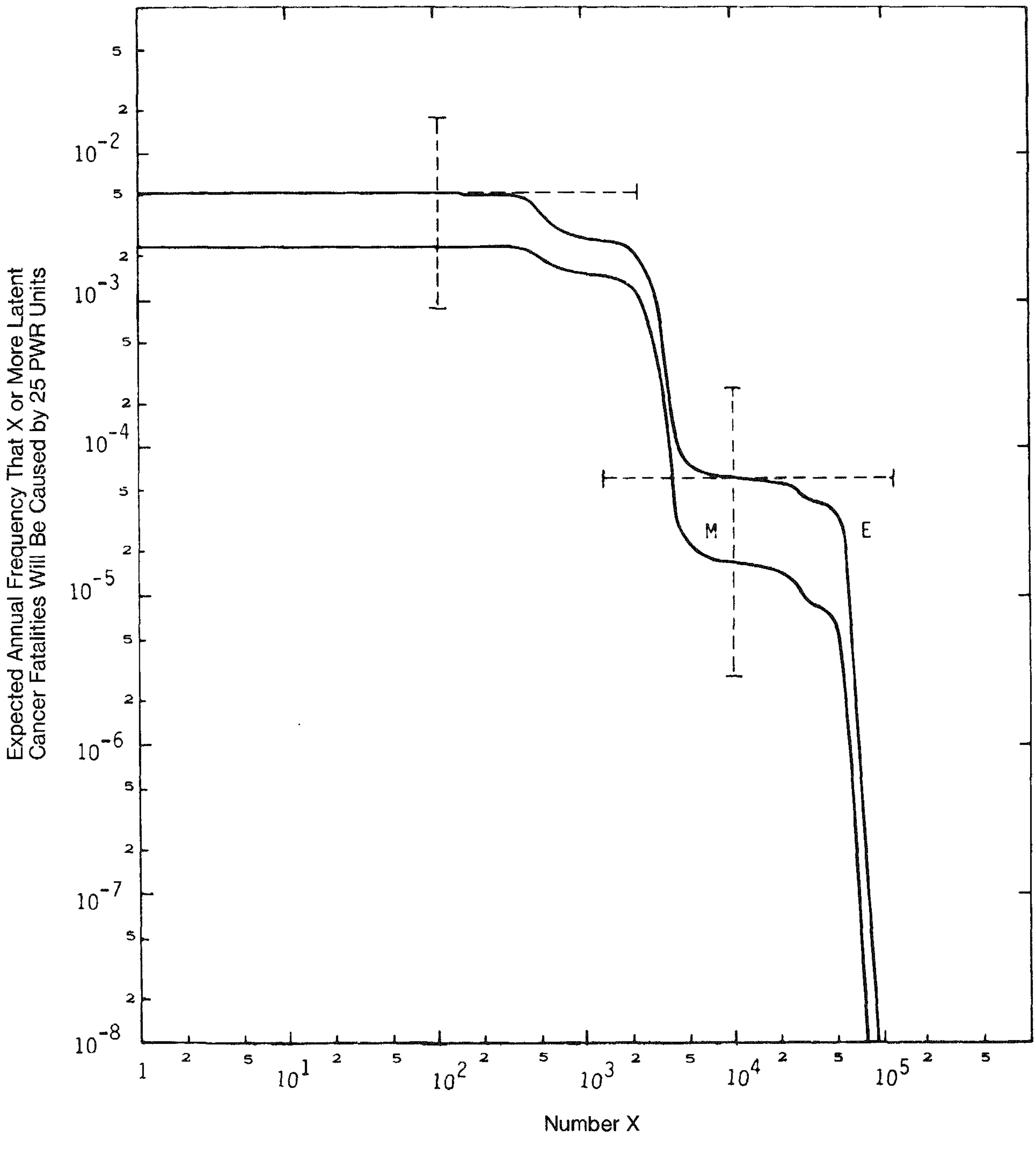
Evaluating the response function for each of the 1,000 value sets provided local (i.e., for each of the particular consequence magnitudes X selected) approximate values of the attendant CCDF. Thus we have:

- 1000 frequencies for each given consequence X*
- 1000 consequences for each given frequency H*.



Determined by:
The median probabilities of release categories (curve M)
The expected probabilities (curve E)
The dashed lines provide subjective 90% confidence intervals. The real value lies within these bands with 90% subjective uncertainty, provided all nonquantified estimation uncertainties are negligible.

Figure 8-13. CCDF for early fatalities



Determined with:

The median probabilities for release categories (curve M)

The expected probabilities for release categories (curve E)

The dashed lines give subjective 90% confidence intervals. The correct value lies with a 90% subjective certainty within these bands, provided all nonquantified estimation uncertainties are negligible.

Uncertainty band and curve profile are greatly affected at low consequences X by the 10% frequency addition (see section 8.1 and footnote in section 9.2) of category 8 to category 7.

Figure 8-14. CCDF for latent cancer fatalities

If we rank the 1000 frequencies or consequences in descending order, then the 50th value from top and bottom serves as the bounds of a subjective local 90% confidence interval. Figures 8-13 and 8-14 show several local confidence intervals (dashed line). Accordingly, the correct value of the particular quantity (consequence or frequency) is 90% certain to lie within this region, provided all non-quantified estimation uncertainties are negligible.

8.2.3 Subjective Confidence Interval for the Resulting Curves

Figure 8-13 shows with what expected annual frequency X or more early fatalities are caused by 25 PWR units, according to calculation. Curve (E) is based on the parameter values given as reference values in Section 8.2.1. Curve (M), obtained by using the 50% fractiles (medians, see Section 4.7.2) of the release frequencies (analogous to the procedure in WASH-1400) and of the reference parameter values from Section 8.2.1, is also entered. The confidence intervals, of course, are unaffected by this selection.

The dashed bars show local, subjective 90% confidence intervals. That is, on the basis of quantified estimation uncertainties, the expected annual frequency that, for instance, more than one thousand early fatalities will result from the 25 units analyzed, lies between 7×10^{-9} and 2×10^{-6}, with 90% subjective confidence (or with 95% subjective confidence for less than 2×10^{-6}).

With respect to the consequence direction, the subjective confidence intervals indicate, for instance, that the number of early fatalities X lies between 80 and 5,100 with a 90% subjective confidence (or with 95%, below 5,100) for the 25 units with the expected annual frequency of 10^{-7}.

Figure 8-14 gives analogous information on "latent cancer fatalities."

FOOTNOTES

(a) The probability density function is called "probability density" below.

(b) Release category FK1 represents core melt accidents that lead to a considerable release of fission products assuming a steam explosion. Such accident sequences are extremely improbable and cannot be absolutely precluded given the present status of the studies; the results of the accident sequence calculations are given with and without inclusion of release category FK1.

(c) The following abbreviations are used below:

GDER = German Study Dose-Effect Relation (curve D from Figure 7-5);
ADER = American Study Dose-Effect Relation (curve B from Figure 7-5).

(d) To verify that those weather sequences causing the greatest consequences were sufficiently represented by the 115 samples, the number of samples was increased. Results showed no greater maximum, and it was concluded that the 115 weather sequences selected were sufficiently representative.

(e) If we consider conditions only in the FRG, then the above numbers of leukemia or cancer fatalities due to natural or man-made causes and by natural radiation exposure, respectively, should be divided by 11.

(f) The definition of the term "risk" was given in Chapter 2.

(g) The mean of the collective injury is also called the collective risk.

(h) The "expected annual frequency" (see Section 2.4) should not be confused with the random quantity "frequency," which is always a whole number (e.g., "frequency in year Y"). We often use the designation "frequency" for the sake of brevity, this always means "the expected annual frequency."

(i) With respect to statistics, by confidence we mean, for instance, the probability that the value of a random sample will be within a certain interval (confidence interval) which contains the particular value of a fixed, but inaccurately known quantity. In a 90% confidence interval based on random sampling, then we are 90% certain that it contains the desired value (even though it will actually be contained with probability of either 1 or 0). If the confidence interval is predominately based not on random sampling, but on expert opinion (see Section 2.4.4), then the validity or confidence is designated as subjective.

(j) Not every event sequence studied has been defined in detail. Therefore, an entire spectrum of potential fission product release fractions is given. Thus, the released fission product fractions correspond with a certain probability to representative values of the next higher or lower category. The expected frequency of the event sequence, weighted with this probability, must also be added to the expected frequencies of the neighboring categories. This addition is already included in the fractiles and reference values given above. Addition and weighting were performed as in WASH-1400, i.e., 10% of the frequency was added to each of the two neighboring categories, 1% to each of the two categories beyond that, etc. Thus, the event sequence in itself would only contribute with a probability of 0.78 in each category to which it was originally assigned. As in WASH-1400, 0.78 is rounded off to 1.0. No addition to event sequences not leading to core melt occurs from categories connected with core melt.

(k) The following response function was selected:

$$F(a,pjx) = \sum_{i=1}^{r} \left(\sum_{j=1}^{s} \left(H\,(1,\bar{p};\ v_i;\ m_j,X) + \sum_{k=1}^{t} \frac{\partial H}{\partial P_k} \right|_{(1,\bar{p};\ u_i,\ m;,x)} (P_k - \bar{p}_k) \right) g_j\ \alpha_i$$

a = vector of release frequency (a_1, a_r), $\bar{a}$ is the vector of the reference value

p = vector of the other, uncertain parameters ($p_1, \ldots, p_t$) from 8.2.1, $\bar{p}$ is--with the exception of the dry deposition rate for aerosols--the vector of the reference value

a_i, $i = 1, 2, \ldots, r$ and p_k, $k = 1, 1, \ldots, t$ were randomly selected in accordance with their distributions.

u = vector of the release categories ($u_1, \ldots, u_r$)

m = vector of the meteorologic site regions ($m_1, \ldots, m_s$)

g = vector of the weights ($g_1, \ldots, g_s$) of the meteorologic siting regions (they depend on the number of plants in the siting region).

x = vector of the abscissa value of the CCDF for which the response function is to be evaluated

$H(1, p; u, m_j, x)$ =

discreet points of conditional CCDFs from the accident consequence model (under the assumption that a release of the particular category takes place).

$\partial H/\partial p$ = vector of the partial differential quotients of the conditional CCDFs. By means of the results of the accident consequence model with 10% and 90% (or 5% and 95%) fractiles of the parameter p_k, $k = 1, 2, \ldots, t$, two differential quotients were determined for each parameter value. In order to evaluate the response function, the respective difference quotients (dependent on the location of the randomly selected parameter values with respect to the corresponding component of $\bar{p}$) were utilized instead of the partial differential quotients.

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Section 9

CONCLUSIONS

The results of the study are documented in detail in Sections 5, 6, and 8. This Section attempts to draw conclusions and to comment on experiences gained during the course of this study.

Accurate interpretation of any analysis requires knowledge of the relationships and limitations that affect the results. Therefore, it is perhaps appropriate at this point to comment on the discussion in the preceding Sections. Since it is not the objective of the study to express an opinion on the acceptability of risk or to compare benefit and risk, the results will not be interpreted with reference to these items.

9.1 LIMITATIONS AND SIMPLIFICATIONS

The study is concerned with risk caused by accidents in nuclear power plants. Risks due to normal operation of nuclear power plants or the attendant fuel cycle are not discussed.

The design conditions of a reference plant constitute the foundation for the plant system analysis. However, data, which for the most part were not derived from the reference plant, were used to investigate reliability. We also had to rely on data obtained for comparable components in other industrial plants. Consequence calculations were based on typical German site conditions, represented by a large number of actual FRG sites. The results therefore do not relate to a particular plant at a real site, but are only models for plants of a particular type at comparable sites.

To interpret the results, one should note that the present study deals not with an exact risk calculation, but with a risk assessment subject to considerable estimation uncertainties. Insofar as a useful basis existed, we attempted to quantify these estimation uncertainties (see Section 8.2). Other (and under certain circumstances significant) uncertainties remain which were not quantified within the framework of this study. In order to cover such uncertainties in the assessment (unavoidable, given the present state of knowledge), a pessimistic assessment procedure is often called for. A typical example of this is the assumption of a steam explosion which can damage the containment. This hypothetical accident sequence strongly influences the maximum extent of consequences for a core melt accident. But steam explosion is not considered a realistic possibility, but only an extremely pessimistic estimation. This procedure means that the risk is more likely over-estimated than under-estimated.

In evaluating available results, the following points must be kept in mind:

- The study is, in many respects, based on existing research. For instance, the simulation of accidents as performed within the framework of nuclear licensing procedure forms the foundation for establishing minimum requirements of safety systems for this study.

- In accordance with the contract and similar to WASH-1400, the risk contribution of war, sabotage, and similar events is not considered. A series of other effects (e.g., fire, flood) were handled on an aggregate basis.

- Human error was considered in the reliability investigations to the extent that plans provided for in the operating handbook will affect the handling of accidents. Unplanned interference by operating personnel that could result in the initiation or control of accidents was not considered. Such interference affects only the probability of event sequences and can have a positive, as well as negative, effect. It is hardly possible that they will lead, in principle, to new consequences not covered by the study.

- For the sake of simplification it was assumed that a partial failure or a delayed response of safety systems that resulted in insufficient cooling caused a complete melt of the reactor core. This method tends to overestimate the frequency of core melt accidents. A differentiated, less pessimistic consideration would also have to treat preliminary stages of a complete melt (cladding damage, partial melt).

- To investigate processes during and after a core melt accident, by and large only simplifying models are available. In many areas, gaps in knowledge must be covered by pessimistic assumptions. For plant-internal processes, this is relatively simple. To calculate accident consequences, the influence of important assumptions on the accuracy of results was quantified.

This enumeration of limitations and simplifications makes it clear that, because of pessimistic assumptions, the results are only estimates, indeed, overestimates of risk.

9.2 PROBLEMS IN CLASSIFICATION OF RISKS

Meaningful evaluation and classification of risks presume a suitable scale for comparison. We are primarily interested in comparing technologies that achieve the same purpose, in this case a supply of energy or electricity; but we are also interested in comparing entirely different technologies like nuclear, chemistry, or aircraft. Finally, the risks in a particular technology can be compared with its benefits or with the risks inherent in its nonuse.

Many risks are characterized by the fact that injuries may occur relatively frequently but their number is comparatively small in each case. Such risks can be evaluated without special difficulty by statistical assessment and evaluation of injury cases.

However, characteristic of risk due to accidents in nuclear power plants--as of a number of other risks--is that on the one hand, because of precautionary measures, injuries are seldom expected; and on the other hand, large-scale catastrophes cannot be entirely excluded. A comparative evaluation of such risks is made more difficult by the fact that quantitative risk studies like the present one have previously been performed only for nuclear power plants.

One of the few exceptions is, for instance, a risk estimation for a large petrochemical plant that was recently completed in Great Britain (Canvey Island Study) (1). In Figure 9-1, important results of this study are shown. Through improvements suggested in (1) for the particular plants and considered in curve B, the frequency of, but not the estimated maximum number of deaths is reduced.

Here, characteristic of theoretical risk analyses in general, this study manifests that for all accidents that cannot be absolutely precluded, probabilities greater then zero are found. If the hazard potential remains the same, the probability for harm can be reduced by additional safety precautions; but basically, the maximum extent of harm cannot be decisively affected.

Since the methods and assumptions in the Canvey Island Study differ greatly from those in this study, a direct comparison of results is not possible.

In technologies that have existed much longer or on a much larger scale than nuclear technology, risks can also be empirically determined, with certain reservations. This also applies to numerous natural risks. However, comparing such

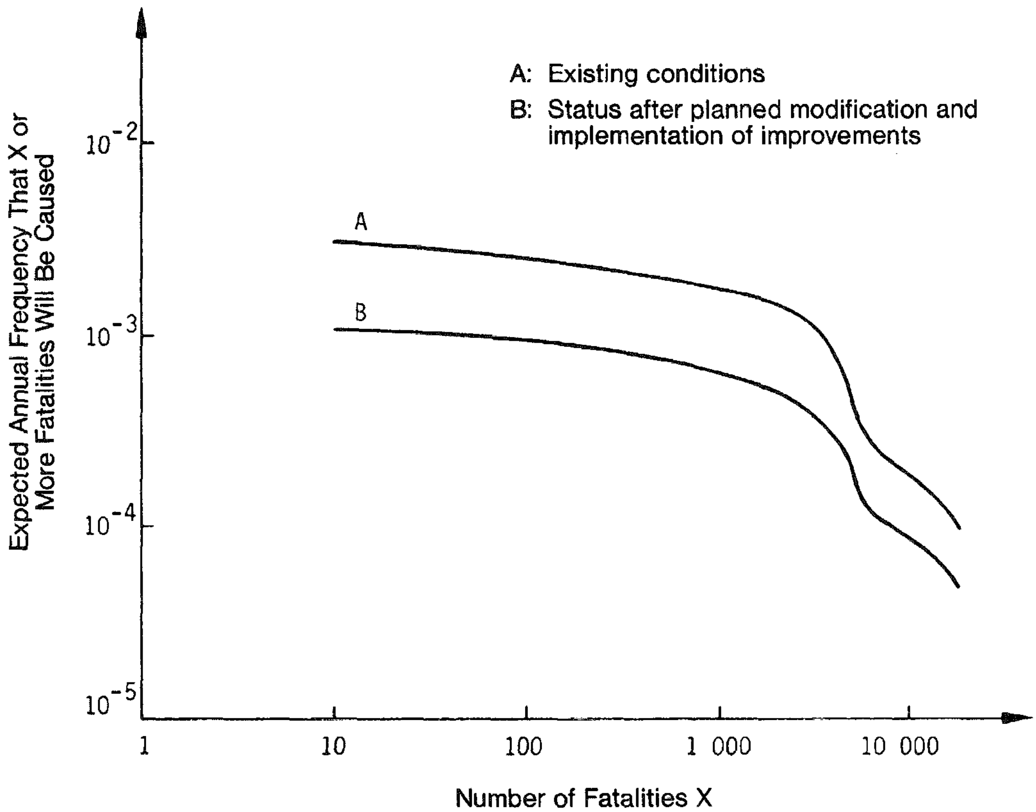


Figure 9-1. CCDF for fatalities due to postulated accidents in chemical plants on Canvey Island (Great Britain) according to (1)

values from experience with analytically estimated risks could lead to incorrect conclusions. It must be noted that because of safety precautions taken in all potentially dangerous technologies and against most natural disasters, major disastors in general are seldom anticipated.

If major disasters have never occurred in a particular area, this does not necessarily mean they are physically impossible.

This is explained by Figure 9-2, which is based on a compilation of empirical data on the frequency of accidents having a large number of fatalities (2). The solid part of the curves is based exclusively on findings in Great Britain. For example, fires and explosions there have taken a maximum of 50 lives in a single event. In (2) these curves are supplemented by worldwide findings (dashed part of the curves in Figure 9-2). According to these data, fires or explosions with more than 1000 fatalities--even though very infrequent--are quite possible. The same is true for other types of accidents.

In this regard we must ask to what extent event sequences of extremely low probability can be important to a risk evaluation. Theoretical risk analyses for complex systems, which by nature are characterized as estimations, become even more uncertain with increasing consequence and decreasing probability. It is doubtful whether the state of knowledge is sufficient to determine reliable results for events with a probability of one-in-a-billion-per-year or less. Large consequences occur only for a combination of unfavorable circumstances whose potential interactions become increasingly difficult to quantify.

In addition, very small probabilities or very rare events are difficult to organize into human experience (see also Section 2). Normally, such events, even if they can lead to considerable consequences, are not perceived as real dangers. For instance, a crash of civilian aircraft, which is very unlikely when related to a single flight, does not cause most people to relinquish the use of aircraft. In other words, this means that the probability of an airplane crash of about one-in-one-million-per-flight, which is the worldwide crash rate today, is generally considered an acceptable risk.

But crashing aircraft can also injure third parties. To be sure, only a few people indoors or outdoors have been injured or killed in this manner. But it is quite possible that an aircraft might crash over a densely populated area. In an extreme case, it is even conceivable that a fueled aircraft might crash into a

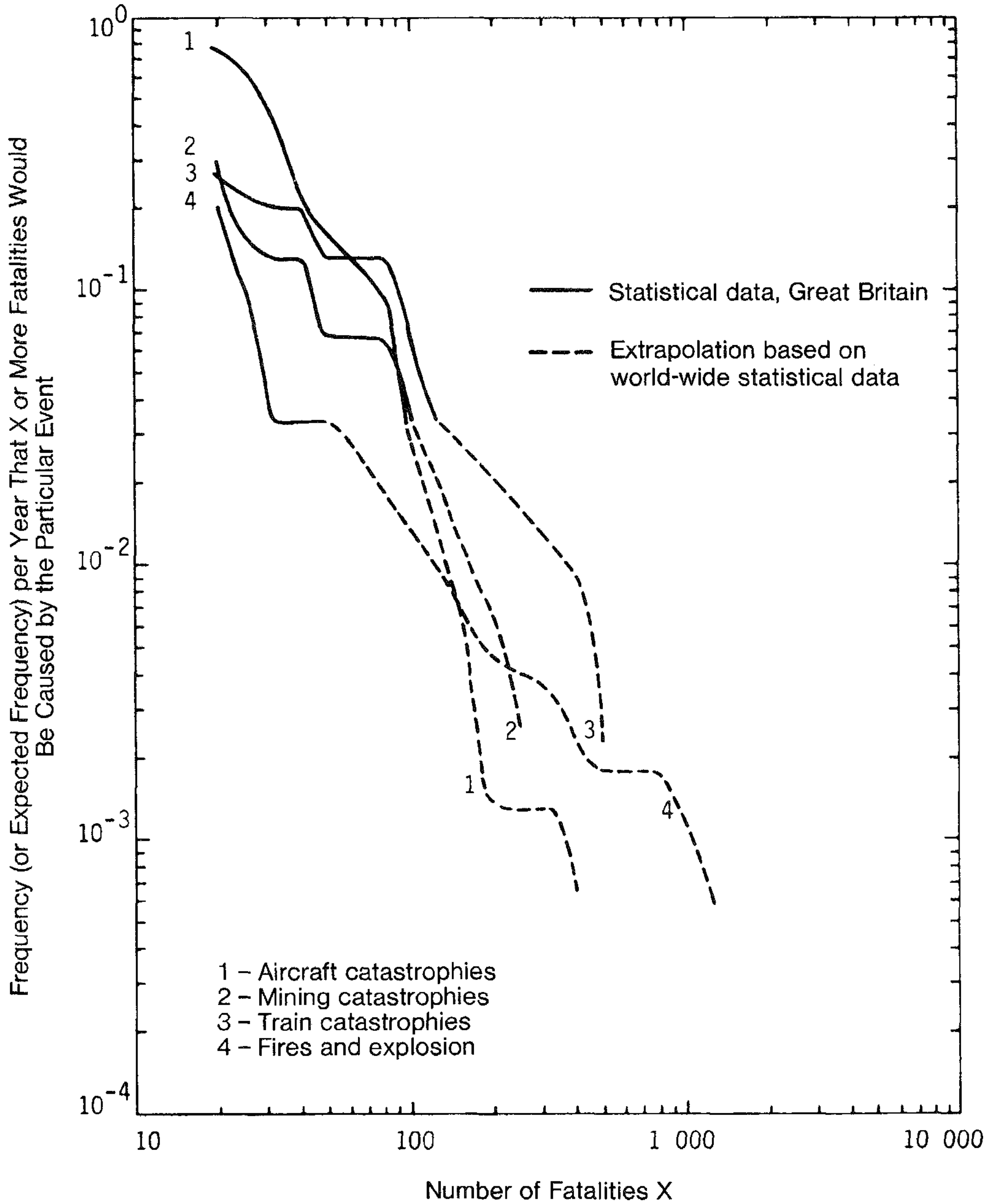


Figure 9-2. CCDF for fatalities due to various types of accidents in Great Britain (2)

crowded stadium and cause considerable damage. From experience for the crash probability of aircraft (including military aircraft) and the relative frequency with which soccer stadiums are filled, we can estimate the frequency of an aircraft crash onto a filled soccer stadium in Germany as 10^{-8} to 10^{-10} per year. Although such an event could cause tremendous damage, no one demands protective measures.

Detailed analysis gives events that are quite impossible by human experience a realistic character. Potential dangers that in all probability will never cause real injury and that in the minds of most people are of no importance are thereby called into question. Thus, the paradox arises that while certain risks are proven minimal, the fear of them grows merely as a result of this proof. However, significantly greater risks (which may not have been studied in detail) are frequently ignored.

9.3 ASSERTIONS OF THE STUDY

It was not the objective of the study to conduct a systematic comparison of risk. However, to permit rough classification of the determined risks, Table 9-1 compares average individual risks derived from this study for persons near nuclear power plants with individual risks from other sources. This classification creates a basis to assess risks to which people are exposed, whether voluntarily or involuntarily.

Besides these quantitative, but crude comparisons, a number of qualitative statements can be derived from the study and even concrete conclusions from parts of the analysis.

In agreement with WASH-1400 the study concludes that the containment of a nuclear power plant considerably reduces the consequences of a core melt accident with high probability. The probability per reactor-year for a core melt accident was estimated at about one in 10,000. In 93% of all core melt accidents, the release of fission products from the containment is so limited that acute fatality cannot result. In the remaining 7%, the frequency of injury is further reduced by environmental conditions (weather effects, population distribution). Thus, in more than 99% of all core melt accidents, no early fatalities are to be anticipated. This is because even in severe accidents, time is still available to initiate emergency protective actions. According to the results of this study, the danger of acute health effects exists only for individuals within a limited area.

Table 9-1. Various individual risks (fatal risk values per 1 million persons in one year)

Type of Risk	Risk (average fatalities per 1 million persons per year)
Fatal accidents due to	
Occupational hazards (average)	130
Occupation in mining	540
Occupation in health care services	40
Domestic and leisure activity	230
Driving (75 minutes per day)	240
Using commercial aircraft (one hour per week)	50
Using other, nonmilitary aircraft (one hour per week)	1,000
Lightning strikes	0.6
Electricity	4
Death due to cancer or leukemia (due to natural and man-made causes)	2,700
Postulated accidents in nuclear power plants (Average value according to this study for the proximity of a nuclear power plant)	
Death due to acute radiation syndrome (early health effects)	0.01
Death due to cancer or leukemia (latent somatic health effects)	0.2

Nonnuclear hazards are largely taken from (3).

Conditions for latent deaths are quite different. Latent health effects on a large scale are calculated not only for core melt accidents, but also for accidents that do not lead to core melt, but for which a failure of the containment is assumed (release category 7). However, a considerable fraction of latent effects (depending on release category, between 40% and 95%) results from accident-induced radiation levels that lie below or on the same magnitude as natural radiation. In addition, it must be remembered that such events are very rare. Therefore, in the proximity of nuclear power plants, the average overall risk of contracting cancer or leukemia increases by clearly less than 0.1% through potential nuclear accident (see Figure 8-10). Since only average values can be derived from the study, concrete assessments of the risk to individual persons are not possible.

The total number of calculated latent fatalities is distributed over a period of 30 years and over very large regions. Under these assumptions, about half the fatalities would occur outside the FRG.

The plant-system aspect of risk assessment permits a generally objective evaluation of the accident spectrum for which the safety features of a nuclear power plant are designed. The study shows, for instance, that failure of the safety systems during a large leak in a primary coolant pipeline contributes very little to core melt frequency. This important design accident considerably influences plant safety features (Figure 9-3). This can also be attributed to subsequent efforts to manage such a postulated accident.

An uncontrolled small leak in the primary coolant pipeline contributes by far the greatest amount to the frequency of core melt accidents. On the one hand, small leaks are more frequent than large leaks. On the other hand, the ability to cope with a small leak is considerably influenced by human intervention. Under the above assumptions, these small leaks cause a relatively high failure probability of the needed system functions in the present case. The greatest contribution to the frequency of core melt is thus due to human error in the handling of small leaks (Figure 9-4). The specific causes for this and possible improvements were discussed in Section 5.2.1.3.

Component failure during a small leak in a primary coolant line and during a power failure contribute the next greater amount. In these cases as well, guidelines for improvement can be derived from the plant systems analysis (see Sections 5.2.1.3 and 5.2.2.2).

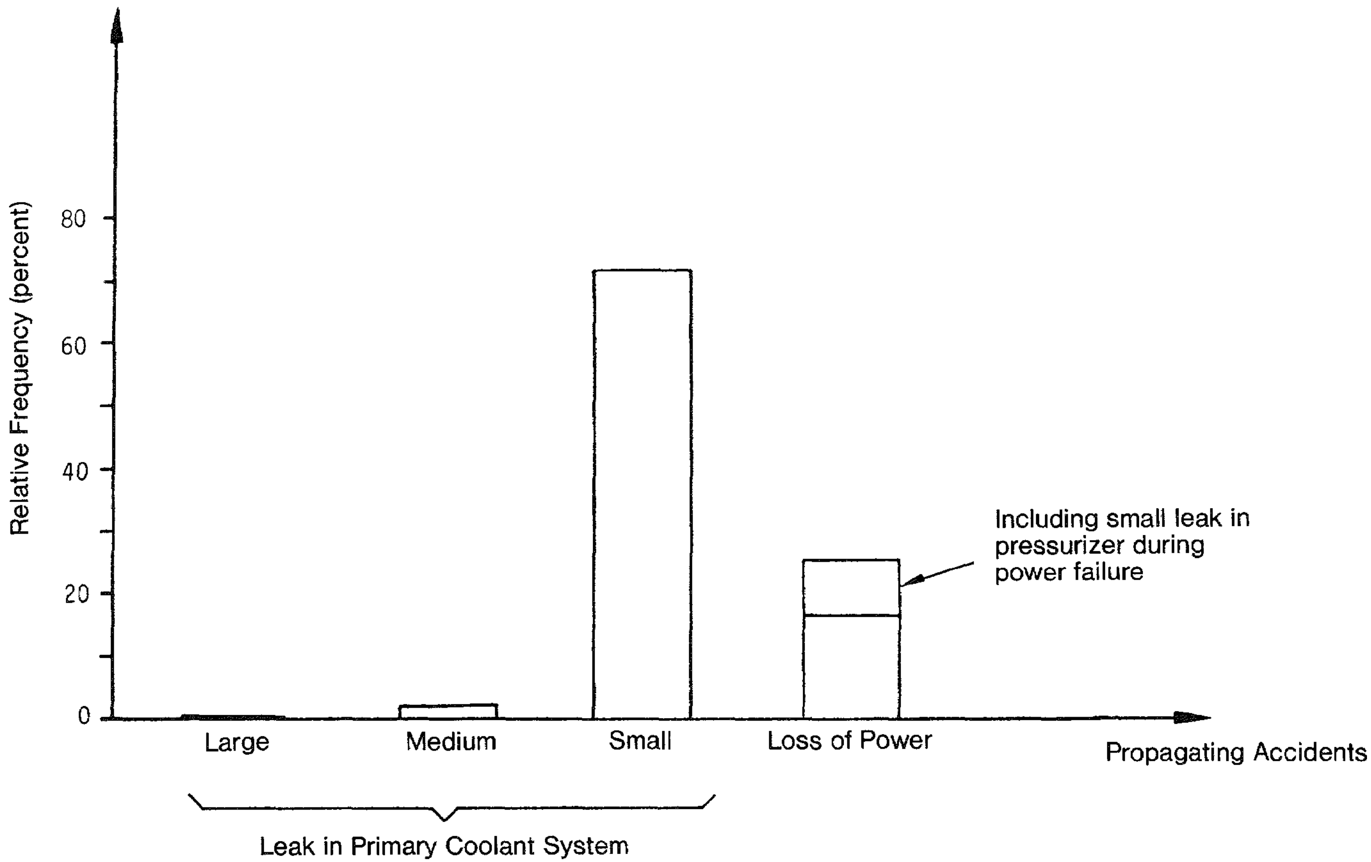
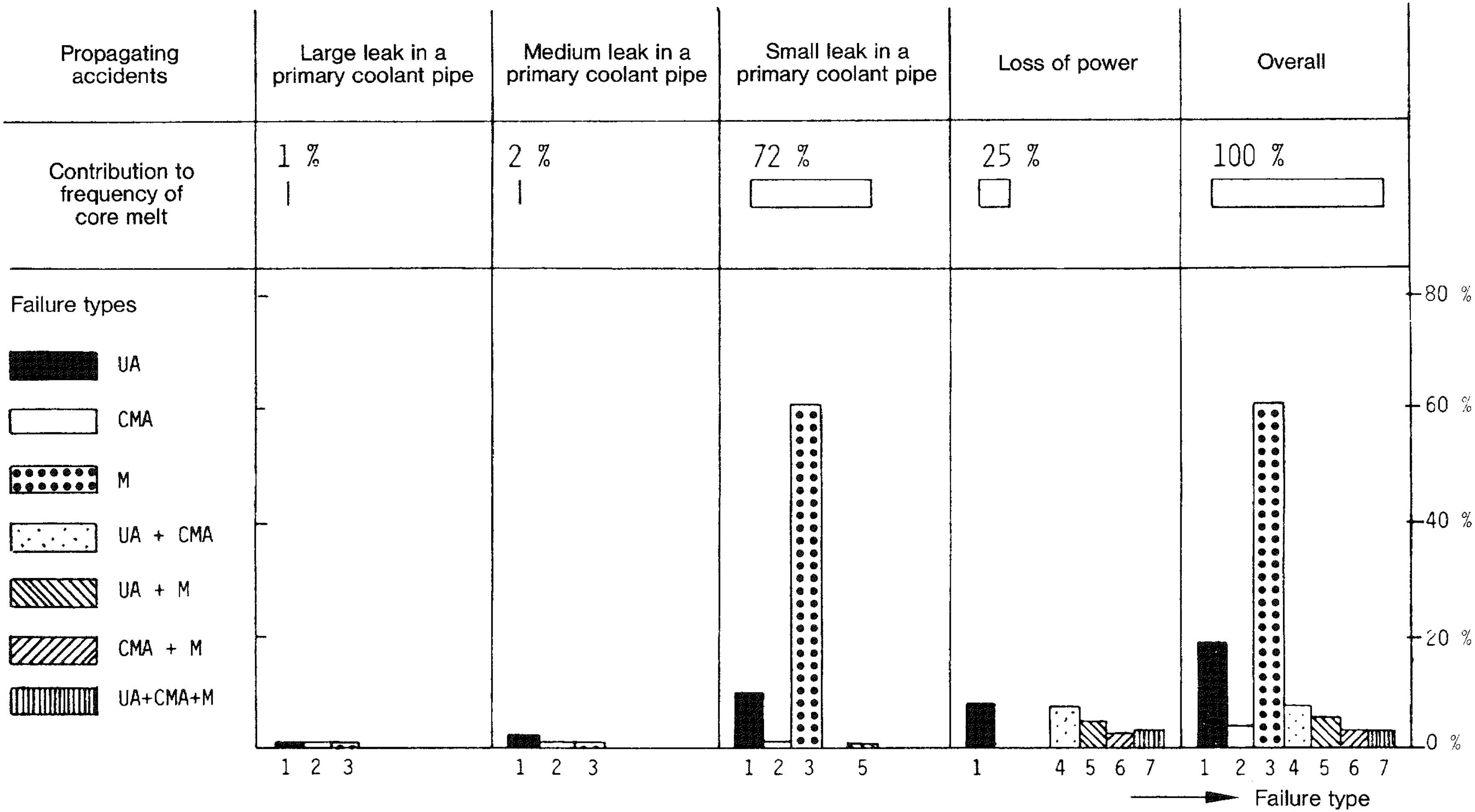


Figure 9-3. Relative contribution of various propagating accidents to the frequency of core melt



UA = Independent random failures of components
CMA = Common cause failures of components
M = Human error including common cause failures due to incorrect calibration of instrumentation

Figure 9-4. Contributions of different failure types to the frequency of core melt

In accordance with the contract, the study was conducted according to methods and assumptions closely associated with WASH-1400. Still, detailed in the preceding Sections, repeated deviations from the procedure in WASH-1400 were necessary. Several of these deviations, in both plant systems studies and the accident consequence model, significantly affect the results. This fact must be taken into account for a direct comparison of the estimated risk.

Nevertheless, it can generally be concluded from a comparison of the results of the study that risks from accidents in PWR nuclear power plants in the FRG and the U.S.A do not significantly differ. Deviations are within the corresponding range of uncertainties in the assessment. The greater population density in the FRG, which initially implied higher collective risk, is thus compensated for and is not reflected in a higher estimated risk.

9.4. USE OF THE RESULTS

During the study it was found that often very detailed analyses were needed to recognize important points. For instance, the probability of human error for a small leak (important to the frequency of core melt) depends heavily on the design of the control room and the instruments available there. In these cases, a prerequisite for detailed analysis is accurate documentation of the system design, which is normally available only for completed plants.

Because of the statistical nature of the analysis, the frequencies found in Section 5 even for the reference plant, have only limited applicability, particularly since the analysis includes much data not specific to the plant. However, relative evaluation of the results reveals deficiencies in design, as determined by realistic event sequences in the particular plant. It is just this knowledge that cannot be extrapolated per se from the reference plant to other plants, since differences in systems can sometimes affect results.

This reservation, however, might affect more the determined probabilities or frequencies and less the calculations of consequence. For plants of the same type and similar design, it can be assumed that the sequence of accidents will not be decisively affected by the detailed design. Studies for one reference plant can thus serve as a model for similar plants.

Findings indicate that problems tend to occur at interfaces between different systems and between different technical disciplines. Since risk analyses, especially systematic event sequence and reliability analyses, require a systematic and interdisciplinary procedure for all important parts of the system, such problems are identified with greater certainty than in ordinary deterministic safety evaluations.

Information on typical accident sequences can be used to plan emergency protective measures. They provide, for instance, conceptions about time frames during which protective measures are possible or necessary; to a certain extent they permit the effects of various measures to be estimated. However, the limitations of the method must also be remembered here. For instance, conclusions about the extent of affected regions can be drawn from this study only with reservations (see Section 8.1).

Results of risk analyses should be used as criteria for the evaluation of research and development projects in the area of reactor safety. Since risk analyses are particularly suitable to finding weaknesses in plant system design, meaningful priorities can be established and significance of results can be evaluated.

9.5 METHODOLOGICAL IMPROVEMENTS IN PHASE B OF THE STUDY

In implementing this study, it was found that the methods required for a theoretical risk assessment are in principle available. In numerous respects, however, further developments to improve confidence in the study results are both meaningful and possible.

In phase B of this study, envisioned to follow-up on the studies performed to date and documented in this report, the following issues, among others, will be examined in greater detail:

Evaluation of Operating Experiences

Existing reliability data for components are in part subject to considerable uncertainties. Only a limited amount of data is available to evaluate common cause failures and human reliability. Progress can be achieved here primarily through intensive evaluation of operating experiences in German nuclear power plants, not only with regard to components, but also with systems.

Detailed Study of Additional Accidents

As in WASH-1400, accidents from which no notable contribution to core melt frequency was expected either were not considered in the analysis or were included only by representative discussion of similar accidents. A number of accidents were treated cursorily, as in WASH-1400. These primarily concern potential releases of radioactive inventory outside the reactor core, as well as overlapping effects originating inside the plant (like fire) or outside the plant (like earthquake, flood). Even risk contributions due to accidents that do not lead to core melt and thus generally result in small fission product release were considered by only a rough investigation of controlled LOCAs. It is necessary to more accurately check potential risk contributions from accidents that have not previously been studied in detail.

Greater Differentiation of Event Sequences

To evaluate event sequences in this phase of the study, system were regarded as either fully operational or completely failed; partial failure was treated as total failure. Similarly, with regard to the status of the core, we distinguished only between "fully intact" (except for a certain degree of cladding damage) and "completely molten." The sequence of core melt accidents resulting from failure to control a large leak in a primary coolant pipeline was treated as representative for core melt accidents from all other causes. The occurrence of core melting after a longer delay can result in an overestimation of risk. In this regard, we shall attempt to obtain greater differentiation on the basis of improved accident simulation.

Evaluation of Accident Simulation Accuracy

The accuracy of the accident simulation used in plant system analysis has not yet been quantitatively evaluated. The use of pessimistic assumptions to cover uncertainties in this regard will, as much as possible, be replaced in Phase B by uncertainty quantification.

Improvements in the Accident Consequence Model

The model to calculate accident consequences describes actual conditions only by simplifying assumptions in a number of respects. For instance, the dispersion calculations neglect fluctuations in wind direction. The dose-risk relationships for latent health effects, which were used to calculate the risk of death from cancer or leukemia from the radiation dose, are based on average values for all age groups. Provisions for protective actions and countermeasures in the consequence analysis are characterized by relatively stringent criteria.

Simplifying model assumptions to calculate accident sequences generally tends to overestimate the frequency or extent of consequences, and thus risk. Phase B will attempt to formulate the model in a more complex and flexible manner.

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Section 10

THE ACCIDENT AT THREE MILE ISLAND NUCLEAR POWER PLANT

10.1 INTRODUCTION

Since the risk study concerns hypothetical accident sequences that have never occurred, results can be compared with practical findings only in isolated cases or, at most, small individual areas. Even the accident in Unit 2 of the Three Mile Island (TMI) nuclear power plant does not alter the situation. Nevertheless, whether the significant aspects of this accident were treated in the study should be checked. Evaluation of the study from the viewpoint of the TMI accident also raises the question of how the study can be used to prevent similar accidents. Phase A research presented in this report was almost completed when the TMI accident occurred.

The following individual points must be examined:

- Is the methodology applied in the risk study generally suitable to describe and examine an event sequence like that at TMI?
- To what extent were conditions important for the event sequence at TMI correctly recognized and considered in the theoretical investigations?
- Is the quantitative probability valuation of these conditions compatible with the experience at TMI?

In order to discuss these items in detail, it is first necessary to discuss the event sequence at TMI.

10.2 DESCRIPTION OF THE SEQUENCE OF EVENTS AT THREE MILE ISLAND

Only that part of the event sequence at TMI of interest for the risk study is illustrated below. Detailed discussion is limited to the first three hours of the accident. For further information, the reader is referred to the publications by the USNRC.

The accident involved a "small leak in the pressurizer" triggered by a transient, with loss of coolant through an open relief valve, whereby the core was insufficiently cooled. Such event sequences are thoroughly treated in the study. However, Phase A pessimistically assumes that insufficient cooling of the reactor core necessarily leads to reactor melt. Important aspects of the accident sequence at TMI can be described as follows:

- At 4 a.m., March 28, 1979, the primary feedwater system in Unit 2 of the TMI nuclear power plant failed as a result of a malfunction of the condensate purification system, interrupting the feedwater to the two steam generators (a). (All times given below relate to the timing of this initiating event.)

- The diminished heat removal through the boilers led to a pressure increase in the reactor coolant system. According to design, this caused the pressurizer relief valve to open after three seconds and initiated the reactor scram after eight seconds. Consequently, the pressure again dropped and, after 13 seconds, reached the closure pressure of the relief valve.

- The relief valve failed to close. In the power plant control room, a button that does indicate the triggering of the closing signal but does not directly show the position of the valve erroneously indicated the valve to be closed.

- In the emergency feedwater system, which assumes secondary heat removal from the steam generators upon failure of the primary feedwater supply, pumps were at full delivery pressure after about 40 seconds. However, the water did not arrive at the steam generators because, in a previous inspection of the emergency feedwater system, the block valves had been left closed by mistake. This condition was initially unnoticed so that the steam boilers evaporated out within a few minutes. About eight minutes after the accident began, the error was discovered and the valves were manually opened. Thus, the emergency feedwater system was again available.

- The open relief valve caused a pressure drop in the reactor coolant system, whereby a few minutes later, the high-pressure injection of the emergency cooling system automatically began operation.

- The displayed pressurizer water level at first decreased slightly, then increased after about one minute, and reached the upper limit of the indicator range after about six minutes. The operating personnel in the control room concluded from this that the reactor coolant system was filled with water. After 4.5 minutes and 10.5 minutes, they shut off the pumps of the high-pressure injection. In addition, coolant was removed from the reactor coolant system through the volume control system.

- Pressure and temperature increased in the relief tank, in whose water pool the pressurizer valve discharges. After about 15 minutes, the rupture disk of the tank failed, and coolant was blown into the containment.

- The reactor coolant system was saturated after about six minutes as a result of the open relief valve; i.e., steam generation occurred. The continuing loss of coolant caused a progressive emptying of the reactor coolant system, but not a drop in pressurizer water level. This phenomenon was apparently not unnoticed by the operating personnel. They trusted the display of the pressurizer water level and either ignored or misinterpreted other readings and messages indicating a loss of coolant. The initial phase of the accident primarily involved pressure and temperature in the relief tank as well as the failure of the rupture disc, with subsequent increase in the containment pressure. In addition, the fact that the coolant was saturated for a long time would indicate steam in the reactor coolant system. The presence of saturation conditions is not immediately indicated by the instruments. Since the high-pressure injections did not occur at the required level, the important consequence was progressive emptying of the reactor cooling loop.

- After one hour 14 minutes, the primary coolant pumps of one of the two primary coolant loops was switched off; and after one hour 40 minutes, that of the other primary coolant loop was also shut down to prevent damage to the pumps. Thereafter, no natural circulation took place. At the coolant outlet from the core, the measured temperature increased rapidly. After two hours 11 minutes, it reached the display limit (327°C). This system behavior after pump shutdown also indicates an insufficient quantity of coolant in the reactor cooling loop. But no intensified high-pressure injections occurred thereafter.

- After two hours 20 minutes, the incorrect open position of the relief valve was noticed, and the relief line was blocked. To do this, the operating personnel gave a manual command to a block valve connected in series to the relief valve.

- Because of the processes given above, the reactor core was temporarily undercooled. The fuel rod cladding attained temperatures at which a rapid metal-water reaction began. One or more hydrogen bubbles then formed in the reactor cooling loop, which impeded the restoration of coolant circulation. Subsequently, the hydrogen concentration also increased in the containment, which resulted in a hydrogen puff.

- The automatic containment isolation valve is triggered at TMI by an overpressure in the containment of 0.28 bar or more. This value was not reached until after about four hours; an earlier isolation of the building by means of a manual command was not performed. Thus, large quantities of contaminated sump water were pumped through the sump drainage system into the auxiliary building. From there, radioactivity was released to the environment through the ventilating system. This represented the most important path for radioactivity releases during the accident.

- In the hours following the foregoing initial phase, various attempts were made to stabilize the reactor cooling. The pressurizer relief line opened and closed, the secondary steam generator water level increased, the primary coolant pumps started, and in particular, the high-pressure injection once again operated. After 11 hours, natural circulation in a primary coolant loop was reestablished; after 16 hours heat removal took place by forced circulation by a primary coolant pump. Removal of noncondensing gases from the reactor coolant system and the complete refilling of this system with water took several days.

It was of great importance to the accident sequence that the system operating mode was not correctly recognized. The open relief valve, which caused continuing loss of coolant and, as a result insufficient coolant in the reactor coolant system, went undetected. If one of these two states had been recognized in time, countermeasures to safely control the accident could have been initiated. This could have been done simply by closing the block valve in the relief line or by assuring continuous high-pressure injection.

Maintenance of high-pressure injection would have kept the reactor coolant system filled, even with a relief valve open, and sufficient heat removal would have been possible via the steam generators.

10.3 REFERENCE TO THE RISK STUDY

The event sequence at TMI is compared with the corresponding event sequences in the Risk Study, Phase A. These analyses are described in Sections 5 and 10.

Based on the methods used in the Risk Study, the following points are compared:

- consideration of the initiating event at TMI
- determination of corresponding event sequences in the event tree analyses
- determination of probabilities for failure of system functions by means of fault tree analysis.

10.3.1 Initiating Event

The accident at TMI was triggered by the "failure of the primary feedwater supply." This transient is considered an initiating event in the present study. Failure of the main feedwater supply in the reference plant does not, as a rule, result in opening of the pressurizer valves (see Section 5.2.1.4).

If the primary feedwater supply fails because the power fails, then a pressurizer relief valve opens. If, after the relief valve opens, the pressure in the reactor cooling loop continues to drop, and if the corresponding blowdown train is not isolated, a LOCA "small leak from the pressurizer" results. Other possible transient causes for such an accident were also evaluated in the present study, but they play a subordinate role compared with power failure (see Section 5.2.2.4). The initiating event of the TMI accident is thus correctly considered in the study.

10.3.2 Event Tree Analysis

- Event Sequence. The success or failure of system functions produce different event sequences that significantly affect the physical sequence for a given initiating event. The states of system functions existing at TMI during the first hours after the accident began can be characterized as follows: reactor scram occurred; the pressurizer relief valve failed to close, the emergency feedwater supply and main steam relief were operating; primary coolant pumps were operating, high-pressure injection did not occur.

  The system functions important for the reference plant are also treated in the risk study. The TMI accident thus gives no reason to question the suitability of the event tree analysis.

  Were thermohydraulic processes in such accidents more carefully studied, they would result in a great number of event sequences with different effects on the core. One may ask, for instance, what the effect would be of varying the times for failure of emergency feedwater supply or switchoff of primary coolant pumps and for closure of the blowdown train, or of assuming a different operating mode of the emergency feedwater supply and main steam relief or different delivery quantities of the high-pressure injection. In order to implement the thermohydraulic investigations necessary for this, no satisfactory and valid model is presently available. Therefore the study pessimistically assumes core melt if the minimum requirements of core cooling are not met.

- Illustration of the Event Sequence in the Event Tree Diagrams. The following description of the event sequence at TMI in the event tree diagrams of the risk study shows the detail with which this sequence can be reproduced in the simplified event trees of the study. We should point out that the TMI plant differs in important respects from the design of the reference plant. Therefore, the following discussion of the TMI event sequence is intended only for phenomenological purposes.

  Figure 5-6 shows the event sequences possible after a power failure. They contain all important aspects to be considered in the failure of primary feedwater supply at TMI. The event sequence at TMI corresponds to T_1S_2". The following comments regard the individual system functions given in Figure 5-6.

  T_1: The initiating event (the transient) occurs.

K: The reactor scram functions. (The abbreviations of the successful functions are shown in parenthesis below).

I: The main feedwater supply and main steam relief fail, in Figure 5-6, as a result of a power failure; at TMI it is the initiating event.

J: The emergency feedwater supply and main steam relief fail. The temporary failure of this system function for eight minutes causes the steam generator to dry out. The later, successful operation of the emergency feedwater system is taken into consideration by the system function "delayed feedwater supply and main steam relief" ($\underline{b}$).

L: The pressure relief valve of the reactor coolant system opens. During the period when no feedwater supply is available, coolant is blown through the pressurizer relief valve, limiting the pressure in the reactor coolant system.

M: The pressure relief valve fails, i.e., the relief valve stays open.

Q: Delayed feedwater supply and main steam relief occur.

R: This function is unimportant to the particular event sequence.

The result of this event sequence is a small leak at the pressurizer T_1S_2''. Below, we will use the event tree, i.e., Figure 5-8, for this leak.

K: Operation of the reactor scram was discussed above.

I: System functions I and J are presented once again in this event tree since different requirements are made of this function during the LOCA. In particular, in the Biblis B reference plant, this system function would have to cause a shutdown of the plant, i.e., a drop in coolant temperature which is to be initiated about two hours after the beginning of the accident. According to plan, the shutdown should begin 30 minutes after the initiation of the accident. This type of shutdown was apparently not provided for in the TMI plant because of other system technology.

J: Since the emergency feedwater supply and main steam heat removal were restored after a short time, the system function is available here.

B: The emergency cooling start-up signals are initiated. Thus, the containment isolation valve is triggered in the reference plant, which also shuts off the main coolant pumps.

C: The high-pressure injections do not take place. In the TMI plant the corresponding pumps were shut off. The extent to which this is applicable to Biblis B is discussed in Section 10.3; here we will identify only comparable event sequences.

The other system functions E, F, and G are of no interest for a comparison with TMI.

Overall, we now have the event sequence T_1S_2"IC. According to Figure 5-8, a core melt is assumed, i.e., further differentiation of the event sequence is not performed. In particular, no credit is given to the fact found at TMI that core cooling still occurs over a longer time even upon failure of the high-pressure injection and the resulting emptying of the reactor cooling loop.

- Conclusions for the Event Tree Analysis. The discussion shows that the actual conditions in the study are highly simplified and very pessimistic. For the reference plant, better cooling conditions are expected for the same sequence as in TMI. The reasons for this are better natural circulation resulting from the different configuration of the reactor coolant system and, especially, fast plant shutdown.

In the future, a detailed simulation of event sequences will be necessary for a realistic description. Partial failure of the required system functions should be studied. As a partial failure, the delayed or intermittent use of safety features should also be examined. But this presumes improved models to describe thermohydraulic processes.

From the risk contribution of transients and small leaks shown in this study, we see the importance of this type of realistic description of actual conditions. They determine the resulting frequency of core melt accidents by a fraction of 95%. In this type of comparatively slow dynamic process, failure of system functions does not immediately expose the core. In event sequences previously assumed to result in core melt accidents and where the core is not sufficiently cooled over longer periods of time limited cooling is actually in effect. That is, compared with core melt accidents, the reactor core releases much lower quantities of fission products. Thus, the risk determined in the study is overestimated as long as these event sequences assure that the retention of fission products by the containment is no worse than in core melt accidents.

The accident at TMI did not demonstrate the existence of processes that have a higher probability of degrading containment integrity than core melt accidents. According to information available, the earlier idea that gas bubbles could explode in the reactor primary cooling loop is unfounded. As at TMI, delayed containment isolation is not assumed for the reference system because of different system technology. The containment isolation during a LOCA automatically begins operation within minutes.

10.3.3 Fault Tree Analysis

Three of the events in the TMI sequence are of primary interest in evaluating the probability of event sequences performed in the risk study through fault tree analysis:

- unnoticed failure of the pressurizer relief valve to close
- human error
- incorrect reading of the pressurizer water level.

The fact no one noticed that the relief valve had failed to close was important for the sequence of the TMI accident. The indirect way of displaying a closed valve promoted this oversight. Initiation of the closing signal for the control valve was indicated, and not the position of the valve itself.

As explained in Section 3.2 above, this study examined event sequences with failure of the relief valve to close. In the fault tree analysis, failure of the relief valve to close because of defective determination of position can go undetected, whereby the redundant block valve in the relief line will not be closed. Such a possibility originally existed in the reference plant because--similar to the situation at TMI--the position of the control valve, and not that of the relief valve, was displayed in the power plant control room. From this control valve position, a closure signal was derived for the redundant block valve in the blowdown line. In addition, the block valve was not secured by emergency power. Functional testing for coolant pressure control that triggers the closing signals was not provided.

The results of these studies were already presented and discussed at the GRS Colloquium in 1977 (1). The determined frequency of LOCA caused by failure of pressurizer valves to close resulted in several changes in the reference plant: The positions of both the control valve and the relief valve are now monitored; the redundant block valve in the blowdown line is now supplied with emergency power, functional testing is also regularly performed for the coolant pressure control. Thus, the probability that the blowdown line will fail to close, i.e., the probability of a LOCA "small leak at the pressurizer" as a result of transients, has been significantly reduced.

The hiatus in the functioning of the emergency feedwater system was caused by incorrect closed position of valves. The present study considers such events. However, in the reference system they are extremely unlikely because of the separate-loop functional testing of the emergency feedwater system and because of control commands received by these valves upon triggering of the system which move then into the proper position.

In the risk study, a model is used to describe human error. This model takes into consideration potential errors made by the operating personnel in implementing the measures described in the operating handbook. On the other hand, spontaneous measures to cope with an accident can also have negative or positive effects. Such unplanned measures were, in general, not evaluated.

High-pressure injection did not occur during important phases of the TMI accident because the pumps had been switched off as a result of the high pressurizer water level. The risk study does not consider such manual measures. To verify this, the pressurizer water level of the reference system must not play so central a role as it did at TMI. High-pressure injection through the reactor protection system automatically operates even when the pressurizer water level does not drop. According to the operating handbook, high-pressure injections for small leaks are independent of the pressurizer water level. They shut off only when the plant shuts down at coolant temperatures below 150°C. Given the intended mode of operation, pressurizer water level has no effect on event sequence. The high pressurizer water level permits errors by the operating personnel, namely the unplanned shutdown of the high-pressure injection. Planned shutdown of the reference plant (i.e., beginning of shutdown about 30 minutes after beginning of accident), prevents a general exposure of the core by the pressurizer injection and thus does not significantly influence the frequency of core melt accidents in the reference plant.

The behavior of the pressurizer water level during a LOCA must be discussed within the framework of system design and related thermohydraulic analyses. Since phase A of the risk study rests on analyses used in the LOCA licensing procedure, protective actions like those discussed in the licensing procedure are assumed.

10.4 SUMMARIZED EVALUATION

Because no final results are yet available from the U.S.A. on the accident at TMI, a complete evaluation of the accident is not possible. The preliminary analysis shows the following:

- The methodology of the risk study is not called into question.
- The event that initiated the TMI accident was included in the study.
- Event sequences that qualitatively correspond to those occurring at TMI were included in the study.
- A simple distinction between event sequences with definite assurance of core cooling, on the one hand, and core melt on the other, leads to overestimation of risk. This problem is not new, intensified studies are planned for phase B.
- The study concretely contributes to preventing "small leak at the pressurizer" accidents comparable to the one that occurred at TMI.
- The behavior as well as possible consequences of the pressurizer water level for similar accidents in the reference plant must still be studied in detail. The influence of a constant pressurizer water level on the frequency of core melt accidents is, however, small.

FOOTNOTES

(a) The Three Mile Island Unit 2 Plant has only two primary coolant loops with straight tube steam generators, in contrast to the reference plant of this study. From the outlet side of the coolant, two cold primary coolant lines lead from each of the 2 boilers to the reactor pressure vessel. A primary coolant pump has been installed in each of these lines.

(b) Because of the greater water supply in the steam generators of reference plant Biblis B, no drying of the steam generators will occur within 8 minutes. Operation of the emergency feedwater system within this time span would therefore be considered as an intact emergency feedwater and main steam relief. This would have resulted in malfunction T_1S_2' where fewer system functions would be required.

REFERENCES

1. "Kernenergie und Risiko" - Fachvortraege - 1.GRS-Fachgespraech Munich, 3./4. November 1977 GRS-10 (March 1978).