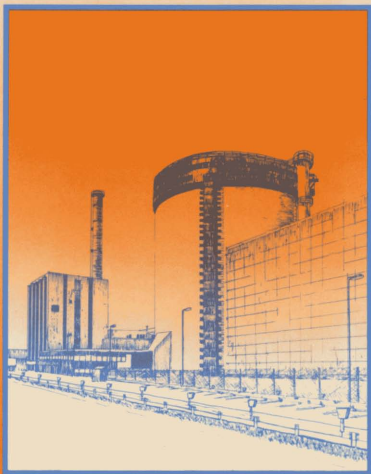


LIGHT WATER REACTOR SAFETY



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Light Water Reactor Safety

BENGT PERSHAGEN

Studsvik AB, Nyköping, Sweden

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Preface

Prerequisites for the utilization of nuclear power for the large-scale production of energy for industrial purposes are that it should be reliable, economically attractive and acceptable from the environmental point of view. Over the years opponents of the expanded use of nuclear power have focused attention on one or other—or even all three—of these prerequisites, and when speaking about the reliability and environmental acceptability of nuclear power they have emphasized the question of safety and that of how to handle the nuclear waste.

Bengt Pershagen, a Swedish nuclear engineer for almost 40 years, has devoted considerable time in the last few years to answering the question “How safe are light water reactors?” including a discussion on the impact of the 1986 accident in the USSR, which involved another type of reactor.

The magnitude of the literature on nuclear safety is such that the author felt it necessary to confine himself to scientific and technical matters. This explains the absence of descriptions of agreements concluded and organizational steps taken by different international and regional organizations in order to facilitate international co-operation in improving safety and reducing the consequences of accidents. It also explains why safety questions related to reprocessing of fuel and the storage of waste are not treated.

The book is the result of a systematic review of the physical processes which form the basis for the normal safe operation of a nuclear power reactor and for the propagation of disturbances which may arise for different reasons and have different consequences. The book is intended to be a sourcebook on light water reactor safety for both professionals and informed laymen. At the end of each chapter there is a collection of references which constitutes a valuable summary of the research and development work done in the field under consideration, and at the end of the book there is a comprehensive index. The author deserves particular credit for these two features of the book.

The Swedish publications listed in Chapter 2 may well be of interest to technicians and politicians outside Sweden wishing to form an opinion about the official Swedish policy on nuclear power.

Pershagen, a former employee of AB Atomenergi and later of its successor Studsvik AB, has continuously been concerned with problems arising in relation to Swedish research and power reactors within the organizations where he worked. He was intimately involved with the Ågesta dual purpose pressurized heavy water reactor (10 MW(e) and 55 MW(th), in operation

from 1964 to 1974) and the Marviken boiling heavy water reactor (200 MW(e)). Marviken was a very advanced project, including nuclear superheating, but was abandoned prior to criticality, partly owing to risks of instabilities. The almost completed reactor has instead been used as a full-scale test rig for various international safety experiments without nuclear heating.

The author is a man of great experience who, when asked the favourite question of some journalists, "Are nuclear reactors safe?" tends to couch his answer in terms of the basic principles of reactor safety and the behaviour of different safety-related systems used by designers and manufacturers of power reactors: he does not give simplistic answers. In his book, two chapters are devoted to the deterministic and probabilistic analysis of how the whole reactor system reacts in abnormal situations for both pressurized water and boiling water reactors.

The principles of consequence analysis are presented, with a description of the sources and characteristics of the radioactive materials in a reactor. Estimates of doses which people in risk zones may receive are possible if the concentration of radioactive substances in the air or on the ground is known. Information of this type is fundamental in any accident leading to the dispersion of radioactivity outside the protective barriers, a situation prone to create panic if correct information is not available.

One cannot over-emphasize how important it is that population groups which risk being exposed to radiation following a reactor accident be informed about their true situation. All people are scared of the unknown, but our fears may be reduced if we know that the increased radiation to which we are being, or may be, exposed was comparable with—say—the slightly higher radiation levels we would encounter and probably accept without hesitation if we moved from our present living place to one in an area with a higher background owing to natural radioactivity.

In spite of the increasing number of power reactors in the world, it is reasonable to expect that the consequences of accidents in the future will decrease and become more manageable. Even if nuclear energy plays a role in the provision of energy in the world for only a limited period, it can be expected to do so with less complications than are associated with the use of some fossil fuel sources.

A couple of years ago, in a review of the Swedish version of *Light Water Reactor Safety*, I expressed my conviction that an English version would be a bestseller. I am even more convinced now, for the book is an invaluable guide for many people—from utility managers, reactor operators and students of nuclear technology to journalists and laymen who wish to penetrate what lies behind the reports spread by the mass media.

Vienna, December 1988

SIGVARD EKLUND

Director General Emeritus
International Atomic Energy Agency

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The English edition would not have been realized without the encouragement and enthusiasm of Dr Sigvard Eklund, my respected friend and former teacher who once introduced me to the field of reactor technology. It gives me great satisfaction that he kindly agreed to write the Preface.

I am indebted to Monica Bowen for her efficient translation from the Swedish. I also want to thank Paula Granath and Katarina Pörn for their painstaking typing and retyping of the many versions of the manuscript.

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Studsvik, April 1988

BENGT PERSHAGEN

1

Introduction

Nuclear power has proved to be a reliable and economic source of energy. Like any other large-scale energy technology, however, it involves risks to life and health. Nuclear power is unique in so far as radioactive substances are formed in the fuel during operation. Some of these radionuclides can be released to the environment in case of an accident. To prevent this from occurring is the prime purpose of reactor safety. Radiation hazards are also associated with the nuclear fuel cycle, especially the management and storage of spent fuel. Another kind of risk concerns the relationship between nuclear power and nuclear weapons.

Nuclear power has provoked debate because of divided opinions on the risks involved. There are two aspects to this issue. Firstly, there is the matter of estimating the risk, which is a scientific and technical task. Secondly, there is the question of deciding whether the estimated risk is acceptable or not, which is a socio-political issue. This book considers only the first aspect, addressing nuclear power plants with light water reactors, or more precisely safety during the design, construction and operation of these plants. Safety issues related to the nuclear fuel cycle are not treated.

There are two main types of light water reactors: the pressurized water reactor (PWR) and the boiling water reactor (BWR). By the end of 1987 there were 225 PWRs and 82 BWRs in operation in the world, representing an installed net capacity of 251,652 MW of electricity in 21 countries. At the same time, 85 PWRs and 10 BWRs were under construction, bringing the total capacity to 336,989 MWel. Light water reactors represented 84% of the total nuclear power capacity in operation or under construction and generated 14% of the electricity in the world during 1987.

The risks associated with nuclear reactor operation arise from the uncontrolled release of radionuclides and not from the uncontrolled release of energy. It is physically impossible for a light water reactor “to explode like an atom bomb” *The basic aim of reactor safety is to prevent the release of radionuclides.* This applies to normal operation as well as to accident conditions. In practice, there is no absolute safety in the sense that radionuclide release can be completely avoided. Releases during normal operation are kept as low as reasonably practical. They are continually monitored and are usually way below limiting reference levels. Uncon-

2 Light Water Reactor Safety

trolled releases in the event of accidents can be large, but have little likelihood of occurring.

The major part of the radioactive material remains trapped in the reactor fuel where it is produced. A necessary condition for the release of this material is that the fuel be damaged. Large releases can only occur when the fuel is overheated and melts or disintegrates. *The basic strategy of reactor safety is to prevent fuel overheating.* This is achieved by designing and operating the reactor so that the power is always controlled and the core well cooled.

During normal operation an equilibrium is maintained between the heat produced by the nuclear reactions in the fuel and the heat removed by the reactor coolant. The equilibrium is stable so that balance is restored if the normal operating conditions are perturbed. In certain fault conditions, the capacity of the reactor's main operating and control systems may be insufficient. The reactor is therefore equipped with special *safety systems* which are initiated when needed to prevent the fault conditions from resulting in fuel overheating. The reactor scram system which automatically reduces the power in abnormal situations, and the emergency core cooling system which is activated if the main cooling system is unavailable, are examples of safety systems.

The safety systems are designed with high requirements on availability. If they do not operate effectively, the fuel can overheat and in severe instances melt partially or completely, resulting in large releases of radioactive material from the fuel. However, in most cases the consequences for the environment will be minimal even if the whole core or parts of it should melt. This is because the central part of the plant is surrounded by a leaktight building, the *reactor containment*. Large offsite releases of radioactive material will only occur if there is a breach or a leak in the reactor containment.

Normal reactor operation includes planned changes of the operating conditions, such as start-up and shutdown, as well as disturbances which are controlled by the reactor's main operating and control systems without suspension of operations.

Abnormal events refer to all fault conditions which lead to unplanned outage. Abnormal events relevant to safety may be classified into:

- incidents*, when the reactor's safety systems are actuated but allowing more or less immediate return to normal operation;
- accidents within (the) design (basis)*, which are brought under control by the reactor's safety systems with insignificant offsite consequences, but which may require long shutdown for correction or repair;
- accidents beyond design*, including all events that the safety systems fail to control or that the safety systems are not designed to control, and which may lead to large offsite releases.

With these definitions, only accidents beyond the design basis are hazardous to the general public. No large-release event has yet occurred in the some 3000 operating years so far achieved (1987) by the world's light water reactors. The much-discussed event at Three Mile Island in Harrisburg, Pennsylvania, on 28 March 1979 resulted in severe core damage but only small radioactive releases to the environment.

On 26 April 1986 a severe accident occurred in a reactor at Chernobyl, Ukraine. The destroyed reactor was of a different type to those treated in this book, and the accident was of a different nature to those most thoroughly analysed for light water reactors. However, since the accident highlighted some safety issues of general significance, a review of the accident and its implications for light water reactor safety is included in this book.

The goal of reactor safety is to ensure that the operation of nuclear power plants does not contribute significantly to individual and societal health risks. Therefore, large efforts are required during all stages of reactor design, construction, operation, inspection and maintenance. Experience has shown that a high level of safety was already reached in the first generation of light water power reactors. Nevertheless, safety levels have successively been raised for new plants, and measures have been taken to improve safety in plants already in operation.

Several parties are involved in reactor safety activities. Governments maintain an overall responsibility through legislation and licensing. Safety authorities regulate and supervise the construction and operation of the plants. The licensee, the electric power utility, is directly responsible for fulfilling the safety requirements. The reactor supplier plays an important role in the design and manufacture of safe reactors. Manufacture and construction are carefully controlled to ensure a high quality of components and systems. Rules and regulations are established to maintain safety during normal and faulty operating conditions.

As experience is accumulated from the operation of nuclear power plants, the systematic analysis of incidents and accidents and the feedback of information to reactor design and operation are perhaps the most important means of improving safety and maintaining a high level of safety. In many cases, non-technical factors such as the behaviour of individuals and organizations have had a decisive influence on the causes and progression of accidents. The administration of reactor safety, the analysis of human behaviour and man-machine interaction, training, etc., have become increasingly important.

Reactor safety is influenced by many factors: reactor design, licensing requirements, operating experience and public debate. This book begins with an overview of milestones in the history of reactor safety. The technical bases for reactor safety and the design of light water reactors are then described. Chapter 6 deals with radioactivity and radiation protection dur-

4 Light Water Reactor Safety

ing normal operation. The following chapters are devoted to principles and practices for reactor safety under fault and accident conditions. Methods for analysing plant safety, containment behaviour, offsite releases and health effects are described. The results of safety studies are reviewed and compared. Chapter 13 analyses operating experience and significant events, including the accidents at Three Mile Island and Chernobyl. The following chapter discusses some generic safety issues and their resolution. Chapter 15 is an overview of reactor safety research. The book concludes with a description of the safety design of the SECURE reactor.

2

Historical Review

The development of light water reactors began in the USA after the Second World War. Principles of reactor safety were elaborated hand in hand with the development work. The U.S. activities were soon followed by similar efforts in other countries. In Sweden, serious interest in light water reactors started in the early sixties. This chapter reviews some of the developments in the USA and Sweden pertinent to reactor safety.

2.1 Developments in the USA

Wartime research showed that nuclear energy could be released through the nuclear fission of uranium and plutonium—both in the violent blast of the bomb and in the controlled chain reaction of the reactor. The possibility of fast power excursions and the hazards of radioactive fission products placed safety issues at the centre of reactor development right from the start. The first experimental reactor, built in 1942 at the University of Chicago under the leadership of Enrico Fermi, was designed so that an uncontrollable chain reaction could not occur (201). As an extra safety precaution, the reactor was equipped with a rod, containing highly neutron-absorbing material which could be quickly inserted into the reactor and thereby interrupt the chain reaction: a rudimentary predecessor of today's reactor scram system.

The Fermi reactor had a thermal power output of only 200 watts and did not require any special cooling. The first reactors for the production of military plutonium were built between 1943–5 in Hanford, Washington, and had a thermal power output of several hundred million watts (MWth). These reactors used natural uranium as fuel, graphite as moderator and water as coolant. They were located in an isolated area with an abundant supply of cooling water and were the first examples of remote siting for public safety.

After 1945, studies of the possibility of generating electricity with a reactor as the power source were started. Several design proposals were made. They all had one feature in common: large safety margins to compensate for the lack of detailed knowledge. The U.S. Atomic Energy Commission established a reactor safety committee to evaluate the design proposals. The

first meeting in 1947 discussed a proposal for a reactor surrounded by a leaktight containment which would prevent the release of radioactive substances into the environment in case of an accident (202). Reactor containment has been a cornerstone of reactor safety ever since.

The idea of using ordinary water as a moderator-coolant and enriched uranium as fuel in a pressurized reactor originated in the U.S. Marines during the war (203). Under the direction of Hyman G Rickover, the first electricity generating plant using a pressurized water reactor as the heat source was built. It was put into operation in 1953 as a prototype for the reactor to be installed in the Nautilus submarine, first launched in 1955. A key to the success of the reactor was the discovery and development of zirconium alloys for fuel cladding. Based on the operating experience from the submarine reactor, the first full-scale reactor for civilian use was commissioned in 1957 in Shippingport, Pennsylvania. At that time the reactor had an electrical power capacity of 60 megawatts (MWel) and was successfully run with various advanced cores until 1982. The first electricity-producing reactors were characterized by the high requirements on the quality of reactor components and systems, which has ever since been a hallmark of reactor technology

The Shippingport reactor was the first step towards the development of pressurized water reactors by the Westinghouse Electric Corporation. The first reactor for the commercial production of nuclear power was commissioned in 1960 by Yankee Atomic Electric Company in Rowe, Massachusetts. The Yankee reactor started off with a capacity of 110 MWel, which was later raised to 185 MWel. It was the prototype for a series of reactors with successively increased output capacity which largely established the design principles for the pressurized water reactor and the main data for the thermodynamic process.

It is a well-known fact that water removes heat more efficiently when it is boiling. At first it was believed that the generation of steam in a reactor would lead to instability. However, in a series of experiments between 1953 and 1955 at the Atomic Energy Commission's research station in Arco, Idaho, it was demonstrated that a light water reactor of suitable design could be operated in a stable manner even if the water in the core was allowed to boil. It was also demonstrated that the reactor would shut itself down if the thermal output and the steam generation increased (204). These results paved the way for the development of the boiling water reactor.

The first experimental boiling water reactor was built at the Argonne National Laboratory in Chicago, Illinois. The reactor was completed at the end of 1956 and produced 5 MW of electricity. Nearly a year later, the first privately financed electric power plant was completed at Vallecitos, California, with a 10 MWel boiling water reactor designed by the General Electric Company. The first commercial demonstration plant was commissioned in 1960 at Dresden, Illinois. The reactor had a capacity of

180 MW_e (later raised to 215 MW_e). The plant had a dual steam cycle so that steam from the reactor could be carried either directly to the turbine or to a special steam generator where secondary steam was produced for the turbine. Any misgivings that the turbine would be contaminated with radioactive materials carried by the steam from the reactor were shown to be unjustified. As a result, the boiling water reactors developed afterwards are designed for direct cycle operation.

The first Geneva Conference in 1955 on the peaceful uses of atomic energy was partly devoted to reactor safety. Papers presented and published in the conference proceedings gave a clear picture of the basic safety principles for reactor design, containment and siting. A U.S. contribution (205) assessed the environmental consequences of a hypothetical reactor accident. By way of example it was shown that if the total radioactive inventory from a 1000 MW_{th} reactor were released in an area with a population density of 500–1300 people per square kilometre, between 200 and 500 people would die and possibly 3000–5000 would be exposed to dangerous levels of radioactivity, even if evacuation were to take place fairly quickly.

In March 1957 the Atomic Energy Commission published a report on the possible consequences of a theoretically feasible but very unlikely reactor accident (206). The investigation, which became known as WASH-740, was carried out by a study group from the Brookhaven National Laboratory. The objective was to provide a basis for decision on the liability for damage in the event of a reactor accident. No large nuclear power reactor had yet been commissioned at the time the report was published.

The investigation attempted to estimate the damage to life, health and property resulting from radioactive releases following an accident. With the stated aim of arriving at the maximum consequences of such an accident, it was assumed that 50% of the inventory of radioactive substances in a 500 MW_{th} reactor would be released into the atmosphere, and that the release would occur under unfavourable weather conditions. The number of fatalities was estimated at between 0 and 34,000, and the number of injuries at between 0 and 43,000. Up to 240,000 km² of land would have to be placed under some form of restriction. The upper limit values referred to conditions estimated to occur during less than 10% of the time and were, according to the investigators, probably overestimated due to the conservative assumptions used.

In 1950 the Atomic Energy Commission's Reactor Safeguards Committee had already proposed regulations for reactor siting (207). An exclusion zone was defined around the reactor with an area proportional to the reactor output. No buildings should be allowed within the exclusion zone. Outside the zone, a limit value for the calculated radiation dose should not be exceeded. Actual siting criteria based on these principles, and defining the physical conditions of a proposed site, were prepared by the Atomic Energy Commission in 1959 and enacted in 1962.

The first organized anti-nuclear movement began around 1962. It was mainly provoked by an application for permission to construct a 1000 MWe nuclear power plant in Ravenswood, New York City, and by the planned location of two plants in California. The Ravenswood case was a matter of principle, namely whether or not the siting of a nuclear power plant in a densely populated area could be permitted. The case was not carried through, as the application was withdrawn by the utility. Plans to build the Californian plants were also abandoned. The critics were in favour of using the site for recreational purposes and against the location of the reactor in a potential earthquake zone. After lengthy public hearings, the Atomic Energy Commission found that there was insufficient basis for a decision (208).

The commercial breakthrough of light water reactors came in the mid-sixties when large plants with pressurized and boiling water reactors were ordered. Even though only a few demonstration plants had been commissioned, there was a rapid increase in the power capacity of the reactors tendered and ordered. The Atomic Energy Commission then appointed a study group to review the emergency core cooling systems, i.e. the reserve systems for preventing core overheating if the main coolant system failed. The study group published its results in a report (209) which became a turning point in the attitude towards the emergency core cooling systems and their function.

When the first reactors were designed, it was assumed that the emergency core cooling systems would operate as intended when required, for example, in the event of a pipe break in the reactor's primary cooling system. The reactor containment was designed to withstand any increase in pressure which resulted from the pipe break due to flashing steam and hot water, provided the emergency core cooling was effective. The study group analysed cases under the assumption that the emergency core cooling system did *not* function efficiently. It was shown that this could lead to the meltdown of large parts of the core. It could not be assured that the containment would remain intact if the entire core or parts of it melted.

The study provoked considerable activity in the field of reactor safety. The conditions for licensing were tightened as from 1966. The emergency core cooling systems were improved in new reactors: greater capacity, assured electricity supply and better instrumentation. The performance and safe operation of the systems increased considerably. The Atomic Energy Commission also passed a regulation that older reactors should be modified to improve their emergency core cooling systems. An extensive research programme was launched in order to determine the progression of a loss of coolant accident and to demonstrate the conditions for effective emergency core cooling.

Unexpected results were obtained during some experiments in a thermo-

hydraulic loop at the Atomic Energy Commission's research station in Idaho. These small-scale experiments used electrically heated rods which simulated nuclear fuel. The injected emergency cooling water did not behave as anticipated and did not reach the rods. Later it was shown that the results were specific to the experimental set-up and therefore not representative of the real behaviour in a reactor.

The lack of agreement between calculations and experiments caused the Atomic Energy Commission to tighten the requirements on the calculational models used for analysing loss of coolant accidents. Provisional criteria issued in 1971 specified fairly detailed assumptions for this type of analysis. It was expected that the criteria would be modified as new information became available from the research programme. Expectations that the new criteria would quell the debate were not met, however. Instead, the debate intensified. At that point, the Atomic Energy Commission decided to hold public hearings on the emergency cooling criteria. The hearings took place from January 1972 to July 1973 and produced more than 22,000 pages of documentation (210).

The debate on emergency core cooling led to a series of measures. The interim criteria were revised on several points. The programme for reactor safety research was expanded and several new full-scale projects launched. Up until 1976, the new criteria resulted in a temporary reduction in output of 5%, on average, for all nuclear power units in operation or under construction in the USA. Modified reactor designs were produced by all four U.S. light water reactor suppliers.

A critical analysis of the emergency core cooling criteria and research programme was published in 1975 by a study group from the American Physical Society (211). The group arrived at the conclusion that the quantitative evaluation of all aspects of reactor safety was hardly possible on the basis of the information available at the time. The group considered that intensive research conducted over a period of 10 years could result in considerably improved knowledge.

In particular, the group recommended increased efforts to reduce the possibility of operator error in the management of abnormal events, as well as increased efforts to meet the high standards of quality for reactor system design and construction. The safety margins for emergency core cooling should be better quantified and, if necessary, increased. Problems relating to the behaviour of the reactor containment in accident situations should be further studied.

The Atomic Energy Commission had begun to prepare formal safety regulations as early as in the mid-sixties. The regulations defined the basic safety requirements for the design, construction and operation of a reactor. They set standards for radioactive releases and established design criteria and operating rules. A very comprehensive code of rules and regulations

has developed with time. While this code has been instrumental to reactor safety activities worldwide, it has also to some extent been counterproductive in obstructing and delaying the licensing process in the USA.

Opposition to nuclear power began to intensify in the late sixties, when some books and articles, hostile to the idea of nuclear power, were published. The Atomic Energy Commission, which had previously opted against participating in the debate on nuclear power, now decided to face the critics. This started a period of confrontation which culminated with the public hearings on emergency core cooling mentioned earlier.

Another event in the early seventies which was of future importance was the case of Calvert Cliffs. It concerned the application of the new Environmental Protection Act to nuclear power plant siting. Through a court ruling, the Atomic Energy Commission was enjoined to not only carry out a complete analysis of the effects of the particular nuclear power plant on the environment but also to provide evidence in support of the need for energy as well as to investigate the environmental effects of fulfilling the energy need by alternative means.

The general design criteria (212) promulgated in 1971 are basic to the design of the safety systems in current nuclear power plants with light water reactors. The criteria involved the postulation of limiting accidents which were to be accommodated by design without significant radioactive releases to the environment. For example, a loss of coolant accident as a result of a sudden rupture of the largest pipe in the reactor's main cooling system is the design basis accident for the emergency core cooling systems and the reactor containment.

The principle of design basis accidents reflects a deterministic safety philosophy. The probability of the postulated accident is not explicitly taken into account, nor is the possibility of more extreme accidents. Critics pointed out that there was a risk of concentrating safety efforts on fulfilling the criteria rather than on improving safety. On the other hand, it was necessary to have very detailed rules and regulations in order to ensure a high and uniform level of safety in reactor design within the rapidly expanding reactor industry.

Remote siting, reactor containment and design basis accidents were the cornerstones of the approach to reactor safety during the years of expansion. In the mid-sixties attempts were made to locate nuclear power plants near population centres. The need for a quantitative measure of safety then arose. In 1967 the Englishman F R Farmer proposed a simple criterion based on the concept of risk (213). A risk value was defined as the product of the probability of a radioactive release and the magnitude of the release.

Due to the complexity of a reactor plant, it was not yet possible to calculate the probability of accidents that could lead to large releases, much less the magnitude of the release. It was not until the mid-seventies that it became feasible to conduct a broad study of both the probability and the

consequences of conceivable reactor accidents. This study, which was carried out under the leadership of Norman F Rasmussen, at the request of the Atomic Energy Commission, is known as the Reactor Safety Study and represents a milestone in reactor safety. The study was published in 1975 (214) by the Nuclear Regulatory Commission (NRC), the Atomic Energy Commission's successor as regulatory and supervisory body.

The Reactor Safety Study drew attention to the importance of core melt-down as a condition for large radioactive releases. More than a thousand event sequences were analysed in detail. The core melt probability, the radioactive release and the offsite consequences were estimated. It was found that other types of events than the design basis accidents dominated the overall risk. According to the study, the largest contributions were obtained for accidents with core melting and containment failure. The importance of the reactor operator as a source of error as well as an agent for steering an accident sequence in a favourable direction was demonstrated.

At the request of the NRC, a critical evaluation of the Reactor Safety Study was performed by a group of scientists with different opinions on reactor safety (215). The group found probabilistic risk analysis to be sound method and an important step forward compared to previous methods of safety analysis. The group recommended that the method be further developed and used more often for safety assessment. However, it was considered difficult to balance the degree of optimism and pessimism in the probabilistic estimates. The conclusion was therefore that it was impossible to determine whether the probability of core melt had been overestimated or underestimated, but that the uncertainties had been definitely underestimated.

Systematic reliability analysis using probabilistic methods has become a powerful tool for identifying safety issues and selecting and evaluating measures to improve safety. The Reactor Safety Study was intended to be generic, i.e. specific to pressurized water reactors and boiling water reactors. No significant difference in the overall risk could be noted for these two reactor types. A similar study was conducted in West Germany for a pressurized water reactor of German design, which essentially yielded the same results as the Reactor Safety Study (216).

In March 1979 an accident occurred at Three Mile Island, Unit 2, near Harrisburg, Pennsylvania, which dramatically confirmed some of the predictions of the Reactor Safety Study. This event was to play an important part in the future development of reactor safety. Perhaps the most important lesson was, as subsequent investigations demonstrated (217), that there were shortcomings in the non-technical area of safety, regarding

- organization and management,
- routines and instructions,
- operator training,

- emergency preparedness,
- communication with the mass media.

The accident at TMI-2 led to intensified studies of accidents beyond the current design bases, e.g. the analysis of reactor and containment behaviour during core meltdown. Measures are being introduced for mitigating the consequences of such improbable events. However, accident prevention remains the focal point of reactor safety efforts. This is where the accumulating experience of reactor operation provides the basis for risk reduction. Comprehensive information systems for the feedback of operating experience are being used by nuclear utilities and safety authorities worldwide.

2.2 Developments in Sweden

The Swedish nuclear power programme was initiated immediately after the Geneva Conference in 1955. In the beginning, development work was focused on heavy water reactors with natural uranium as fuel, which led to the Ågesta and Marviken projects. The Ågesta reactor was successfully operated from 1964 to 1973 for the production of 55 MW district heating for Farsta, a suburb south of Stockholm, and 10 MWel with a back pressure turbine. The Marviken project for a 200 MWel boiling heavy water reactor with the possibility of nuclear superheat was abandoned in 1970 for technical and economic reasons (219).

During the sixties the utility industry became more interested in light water reactors, partly because of the commercial breakthrough in the USA, and partly because of the possibility of securing the supply of enriched uranium through long-term contracts. In 1965 Oskarshamnsverkets Kraftgrupp AB (now OKG AB) ordered a 400 MWel plant with a boiling water reactor of Swedish design from the ASEA company. This was followed by a contract from the Swedish State Power Board for two units for the Ringhals power station: a boiling water reactor from ASEA and a pressurized water reactor from Westinghouse Electric Corporation. In 1969 two additional boiling water reactors were purchased from ASEA by OKG and the Sydskraft utility.

Plans for the expansion of nuclear power were presented by the utilities in the early seventies. The extent and rate of the expansion was the object of intensive political debate. The policy decision of 1975 forecasted the need for thirteen units by 1985. Nuclear power became an important issue in the 1976 election campaign. The new Government appointed a commission to prepare a proposal for the future energy policy. The Energy Commission's recommendation (220) for a nuclear power programme with twelve units became the Government's proposal in the energy policy bill submitted in March 1979.

In the same month the accident occurred at Three Mile Island 2. The

accident had an immediate effect on the political situation in Sweden and led to an agreement for a referendum on nuclear power. The referendum took place in March 1980. The results caused the 1980 Parliament to rule in favour of carrying on the reactor programme but to limit the use of nuclear power to the technical lifetime of no more than twelve units. As a result, safety aspects will determine the order in which the units are to be decommissioned. The last reactor in Sweden will be shut down in the year 2010.

Safety aspects were considered at an early stage in the development of nuclear power in Sweden. The governmental Atomic Energy Investigation of 1955 (221) pointed out that radioactive substances could be dispersed over populated areas during an accident involving fuel overheating, and that the reactor should be placed in a tight building with walls strong enough to withstand any increase in pressure following an accident. Since the building could not be made completely leaktight, large reactors should be located as far away as possible from residential areas to ensure that the consequences of accidents were limited. According to the investigation, nuclear installations should preferably be located underground in rock caverns.

The investigation led to the 1956 Atomic Energy Act. The Act stipulated that a government licence was required for the erection, ownership or operation of installations for activities relating to nuclear technology and for the acquisition, ownership, transfer, processing of, or any other activity involving nuclear materials. An Atomic Energy Delegation was appointed

TABLE 2.1. *Main data for Swedish nuclear power plants (July 1987)*

Unit	Type ^a	Capacity, MWel gross/net	Commercial operation	Operator ^b	Contractor
Barsebäck 1	BWR	615/600	1975	SK	Asca-Atom ^c
Barsebäck 2	BWR	600/585	1977	SK	Asca-Atom
Forsmark 1	BWR	1008/970	1981	SV	Asca-Atom
Forsmark 2	BWR	1008/970	1981	SV	Asca-Atom
Forsmark 3	BWR	1101/1063	1985	SV	Asca-Atom
Oskarshamn I	BWR	460/440	1972	OKG	Asca-Atom
Oskarshamn II	BWR	615/595	1975	OKG	Asca-Atom
Oskarshamn III	BWR	1110/1070	1985	OKG	Asca-Atom
Ringhals 1	BWR	780/750	1976	SV	Asca-Atom
Ringhals 2	PWR	840/800	1975	SV	Westinghouse
Ringhals 3	PWR	960/915	1981	SV	Westinghouse
Ringhals 4	PWR	960/915	1983	SV	Westinghouse

^a BWR = boiling water reactor

PWR = pressurized water reactor

^b OKG = OKG AB

SK = Sydskraft AB

SV = Swedish State Power Board

^c From 1 January 1988 ABB Atom (ABB = Asea Brown Boveri)

Source: Swedish State Nuclear Power Inspectorate, Quarterly Report, Second Quarter 1987

as an advisory body to the Government. The Delegation was charged with policy-making for activities relating to atomic energy and with advising the Government on licensing issues, legislation and confidential matters in the area of atomic energy as well as with inspecting nuclear installations. The task of reviewing and supervising safety-related activities was handled by the Delegation's Reactor Siting Committee. In 1975 these tasks were transferred to the newly appointed Swedish Nuclear Power Inspectorate.

In September 1956 AB Atomenergi, the national nuclear research establishment, submitted an application for permission to install a materials testing reactor, called R2. The safety-related considerations on the siting and design of the reactor were principally based on information provided at the 1955 Geneva Conference. For the first time in Sweden, an assessment was made of the risks involved in the dispersion of radioactive materials following a reactor accident. In April 1958 the Government issued a licence to AB Atomenergi to construct, own and operate the R2 reactor at Studsvik with the provision that the reactor could not be commissioned until it had received final approval from the Atomic Energy Delegation. In May 1960 the Delegation issued a licence for test operation at low power. In 1961, after supplementary reports had been submitted, the Delegation issued final approval and R2 was commissioned for operation at full capacity, 30 MWth (later raised to 50 MWth).

Sweden has had a long tradition of radiation protection work. The first Act on the utilization of radioactive sources dates from 1941. A review of existing legislation in the area of radiation protection was carried out parallel to the Atomic Energy Investigation. This led to the Radiation Protection Act of 1958. According to this Act, a licence must be obtained from the relevant authority in order to pursue radiological work. However, no licence is necessary for activities covered by the Atomic Energy Act. The supervisory body appointed in accordance with the Radiation Protection Act is the National Swedish Institute of Radiation Protection.

In January 1957 AB Atomenergi submitted an application for permission to construct the first Swedish power reactor, a pressurized heavy water reactor, known as R3, in Ågesta. The assessment of the accident risks was mainly based on documentation published at the 1955 Geneva Conference. In October 1957, partly on the recommendation of the Atomic Energy Delegation and its Reactor Siting Committee, the Government granted a licence for the construction, ownership and operation of the reactor.

In 1959 two official investigations which would play an important role in reactor safety, were published. One of them proposed provisional legislation on liability and insurance for nuclear reactor operation (222). In the report, the accident risks were described in general terms. The proposal limited the owner's liability to 25 million Swedish Kronor and prescribed the owner's insurance liability. If the liability amount did not suffice, compensation would be granted by the State. A Nuclear Liability Act based on

international conventions was passed in 1968. The liability amount has since then been increased.

The other investigation concerned emergency preparedness. In the report (223) there is a chapter on accidents and other disturbances in nuclear installations. Possible types of accidents, their progression and effects are briefly described. An example of a severe accident is presented, based on the USAEC report WASH-740.

In 1960 the Swedish State Power Board submitted an application for permission to construct R4/Eva, a heavy water reactor of the pressure vessel type to be located at Marviken. The Government granted preliminary permission in January 1962, but left the question open as regards the detailed design. The reference design of Marviken as a direct boiling heavy water reactor with the possibility of internal superheat was supported by the Atomic Energy Delegation and approved by the Government in 1963.

The review by the Reactor Siting Committee only treated the version with saturated steam since documentation on nuclear superheat was considered insufficient for evaluating the safety. For the first time in Sweden the licence application provided a realistic account of accidents which could result in large releases of radioactive substances. The basic design was aimed at avoiding such accidents. The Reactor Siting Committee prescribed a series of conditions and proposed certain design modifications. The detailed design of Marviken was finally approved by Parliament in 1965 (219).

During the mid-sixties, general design criteria were established in the USA. These criteria were not available when Marviken was designed. When they were published in 1967, it was evident that the reactor could not comply with them without thorough modification. This fact contributed to the abandonment of the Marviken project in 1970. Instead, the Swedish reactor programme concentrated on light water reactors. A Swedish design of a boiling water reactor was prepared by ASEA, based on the experience from the Ågesta and Marviken projects as well as the U.S. design criteria.

In 1968 a series of applications were submitted for permission to construct nuclear power plants with light water reactors. Several of these concerned power plants located in sparsely populated areas. The sites were approved by the Government after appropriate evaluation by the authorities. The decisions did not provoke any objections, even though the plant at Barsebäck was only some 25 km from Malmö and Copenhagen. One case concerned a cogeneration plant, known as the Värtaverk project, only a few kilometres from the centre of Stockholm. This project concerned the underground siting of a boiling water reactor for 1550 MWth, of which 360 MW would be delivered as electricity and 1100 MW as district heating.

The evaluation by the Reactor Siting Committee led to the conclusion that additional information was required before large nuclear power stations could be sited in close proximity to a densely populated residential area. In June 1969 the Atomic Energy Delegation therefore decided to postpone

the case. In March 1970 the Government launched an investigation of nuclear power plant siting near densely populated areas. In 1971 the assignment was expanded to include the establishment of general guidelines for the distance of a nuclear power plant from an urban area. The final report was submitted in June 1974 (224).

For the first time in Sweden, the Urban Siting Investigation used probabilistic methods for risk assessment. The quantitative analysis was limited to the acute health effects of releases during normal operation and in accident situations from a nuclear power plant, situated between 5–100 km from the centre of a model city with a population of about a million within a radius of 25 km. The evaluation was based on a 100% release of the core inventory of noble gases and a 3–30% release of iodine. The dispersion of these materials in the atmosphere during an accident was calculated using real meteorological data. The general conclusion was that the worst possible effects did not differ in extent as regards acute personal injury from the risks already accepted by society.

The Energy Commission which was appointed in 1976 for the first time comprised members who were known critics of nuclear power. Its group of experts on safety and the environment had an independent risk study of the Barsebäck plant carried out (225) to complement a similar study which had been started earlier by the Nuclear Power Inspectorate. While the probability for core melt was estimated to be about the same in the two studies, the results of the consequence analysis differed considerably, especially for the ground deposit of radioactive substances. A separate study (226) on the core melt probability for a modern Swedish boiling water reactor showed substantially lower values than those of the older American reactors in the Reactor Safety Study. The general conclusion was that the risks arising from nuclear power were acceptable, taking into consideration the alternatives available and the social benefits of electric power.

Already a week after the Three Mile Island accident, the Nuclear Power Inspectorate prescribed certain corrective measures for the Ringhals 2 reactor, which was the only pressurized water reactor in operation in Sweden at that time. The Government requested the Inspectorate to submit, within a month, a report on the sequence of events during the TMI accident and on the measures that had been taken to prevent a similar accident in Swedish reactors. Two months later a committee was appointed to study whether the risks from nuclear power should be re-assessed in the light of the accident, and to investigate which measures should be taken to increase the level of safety in the Swedish nuclear power plants. The committee submitted its report in November 1979 (227).

The Reactor Safety Investigation noted that the real level of safety in Swedish power plants was probably higher after TMI than before, due to the safety issues brought to light by the accident and the measures for resolving them that had been undertaken. The investigation found no tech-

nical reason to re-assess the risks from nuclear power as compared to those previously estimated by the Energy's Commission's expert group on safety and the environment. However, these risks as well as the TMI accident showed, in the opinion of the investigators, that more stringent requirements should be placed on safety. This applied to all stages from the design of reactors and their safety systems via the activities of the supervisory bodies to the daily routines during the operation and maintenance of nuclear power plants.

The Reactor Safety Investigation proposed a number of measures to improve safety within the following areas:

- rules and responsibilities,
- design and construction,
- limitation of radioactive releases,
- man-machine interaction,
- recruiting and training,
- rules for normal operation,
- emergency preparedness,
- feedback of experience,
- reactor safety research.

The feedback of operating experience was considered particularly important for the prevention of accidents. However, since severe accidents could not be ruled out, increased efforts were considered necessary also for limiting radioactive releases.

In a parallel investigation, the Institute for Radiation Protection studied the matter of emergency preparedness (228). While the existing emergency preparedness planning was based on information available at the end of the sixties, the investigators pointed out that the consequences of severe accidents could be larger, especially due to the deposition of radioactive materials on the ground. The investigation used information from the U.S. Reactor Safety Study for accidents involving steam explosion in the reactor vessel or containment. Using unfavourable weather scenarios, worst consequences were calculated as a basis of proposed measures for emergency planning.

Since the worst consequences in the Reactor Safety Study were subject to debate and new experimental information had been brought to light, the Government appointed a committee to review the facts on steam explosions. The committee found that although limited steam explosions could occur in connection with severe core damage, they would not be strong enough to cause the reactor vessel and containment to rupture (229). The committee therefore came to the conclusion that steam explosions did not need to be considered in the design of the safety systems and for emergency planning.

The changed attitude towards nuclear power in Sweden at the end of the seventies resulted in a number of special acts. In 1983 the special Acts and the Atomic Energy Act were combined in the Nuclear Energy Act.

As the Swedish nuclear power programme has been implemented and reactors successively placed into operation, the focus of safety activities has shifted from the design of safety systems and the verification of safety criteria to the analysis and feedback of operating experience and modifications to improve safety in the plants commissioned. Traditional safety requirements for design basis accidents have been supplemented with requirements for limiting radioactive releases in the event of severe accidents (230).

After the Chernobyl accident a new investigation was undertaken to study the basic reactor safety issues and to evaluate possible consequences for the Swedish reactor programme. The conclusion of the investigation was (231) that because of the technical differences between the Chernobyl reactor and the light water reactors there was no reason to reassess the accident risks of the Swedish reactors.

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3

Elements of Reactor Technology

This chapter begins with a description of how the light water reactor works. Fuel design and fuel behaviour during operation are discussed. The prevention of fuel overheating is fundamental to reactor safety. For this reason, the reactor power must be kept under control and the fuel well cooled. Sections 3.3 and 3.4 review the principles of power generation in the core and heat transport from the core to the coolant. Finally, some basic facts for the design of the reactor vessel and coolant system pressure boundary are presented.

3.1 Basic Principles

A nuclear power plant, like any thermal power plant, generates electricity through the medium of steam. A thermal power plant basically consists of a steam supply system and a turbo-generator. Part of the energy in the steam is converted to mechanical work in the turbine which drives the generator. In this process the steam expands and cools, condensing into water which is then returned as feedwater to the steam system in a closed cycle.

The efficiency is a measure of how much of the thermal energy is converted into electricity. In a closed cycle the efficiency of the conversion of heat to mechanical work cannot exceed a certain value determined by the ratio of the absolute temperatures at which heat is removed and supplied. The lower the ratio, the higher the ideal efficiency. Modern nuclear power plants with light water reactors have an efficiency of about 35%. This means that 65% of the primary thermal energy goes to waste, mainly as warm cooling water when steam from the turbine is condensed.

The main difference between a nuclear power plant and a conventional boiler plant is the heat source used in the steam supply system. The primary energy in a nuclear power plant is generated by nuclear reactions—*fissions*—which take place in the core of the reactor. In coal- or oil-fired plants, chemical energy is released through the combustion of organic fuel in the boiler. The reactor core is equivalent to the furnace of the boiler. In both instances the primary energy appears as heat which is transferred to water. The heated water is brought to boiling at high temperature and high pressure. In a boiling water reactor steam is raised directly in the core. In a

pressurized water reactor the steam is produced indirectly via heat exchange in the steam generator.

The design principle of a nuclear power plant using a *boiling water reactor* is shown in Fig. 3.1. The nuclear steam supply system basically consists of a reactor pressure vessel and internals. The pressure vessel houses the core with the uranium fuel. Steam is raised in the core, separated and dried in the upper part of the vessel and then led to the turbine. In order to improve heat transfer in the core, the water which has not turned to steam is recirculated. The fission power and thus the thermal heat output is controlled by inserting or withdrawing control rods or by varying the recirculation flow.

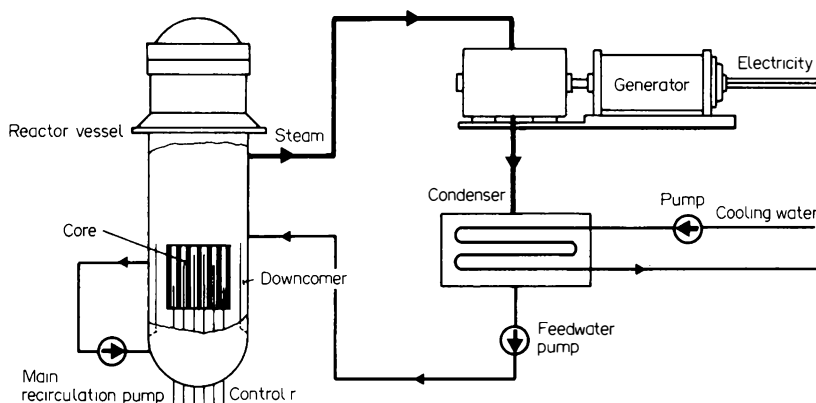


FIG. 3.1. Boiling water reactor schematic

Figure 3.2 shows the basic scheme of a *pressurized water reactor* plant. The reactor pressure vessel is completely filled with pressurized water so as to prevent bulk boiling. The pressure is controlled by means of a pressurizer connected to a main coolant line.

Steam production takes place in separate steam generators. Thus, there are two separate circuits—the primary circuit, including the reactor and the tube bundles of the steam generators, and the secondary circuit, which comprises the shell side of the steam generators, the turbine and condenser. The power in the core is regulated by control rods or by varying the concentration of boron (a strong neutron absorber) in the coolant.

In both types of reactors steam is delivered to the turbine at a temperature of about 286°C and a pressure of about 7 MPa. Because of the temperature difference between the tube and shell sides of the steam generator, the dual cycle of the pressurized water reactor involves a higher primary coolant temperature and pressure than the direct cycle of the boiling water reactor. In practice the core outlet water temperature is about 320°C and the operating pressure about 15 MPa in a typical pressurized water reactor.

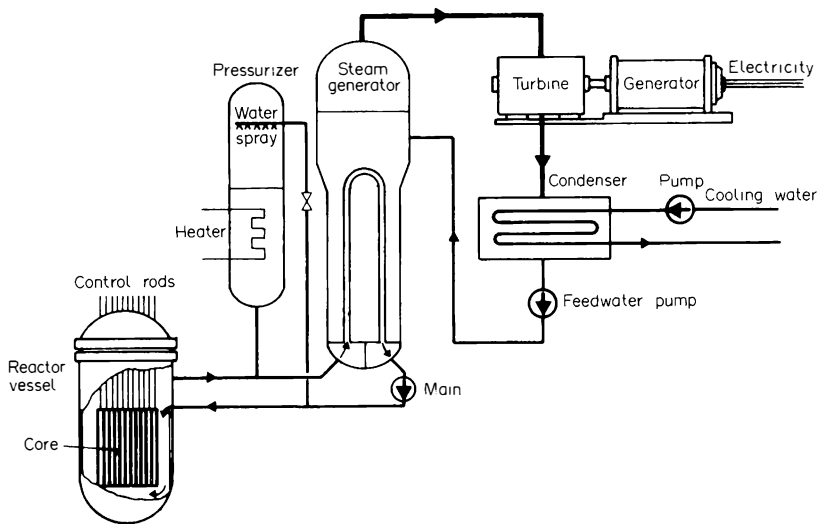


FIG. 3.2. Pressurized water reactor schematic

3.2 Reactor Fuel

The fuel consists of small cylindrical pellets, made of uranium dioxide, UO_2 , a ceramic material with a high melting point. The pellets are stacked in long metal tubes made of a zirconium alloy, Zircaloy, which has low thermal neutron absorption, high strength and good corrosion resistance. The fuel rods are grouped in bundles to form fuel assemblies.

A fuel assembly for a boiling water reactor is shown in Fig. 3.3. This fuel assembly contains 8×8 rods and is about 4 m long. The outer diameter of the rods is about 12 mm. The fuel assembly is enclosed by a square fuel box, made of Zircaloy, through which the coolant flows. There are about 400–700 fuel assemblies in the core of a boiling water reactor, depending on the total power output.

A fuel assembly for a pressurized water reactor has the same basic design. It normally contains 17×17 rod positions but has no fuel box. The rod diameter is about 10 mm. A typical pressurized water reactor (Ringshals 2, 800 MWe) contains 157 fuel assemblies.

Figure 3.4 illustrates a fuel rod. The ends of the pellets are slightly dished to compensate for the axial thermal expansion during operation. The temperature and linear expansion increase from the surface towards the centre of the pellet. Between the fuel stack and the cladding there is a gap filled with pressurized helium. Unirradiated fuel has a diametral gap width of about 0.2 mm. During reactor operation the gap decreases, since the pellets expand more than the cladding. The gas composition in the gap changes as

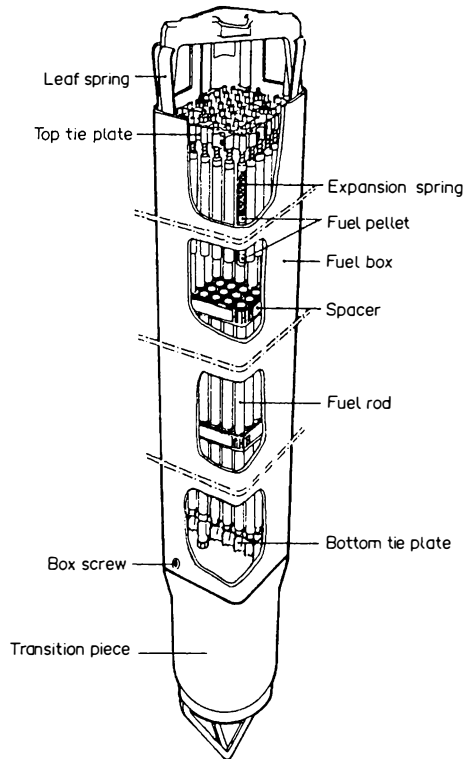


FIG. 3.3. Fuel assembly of a boiling water reactor. Courtesy AB Asea-Atom

gaseous and volatile fission products are released. In order to prevent internal overpressure, a plenum is provided at the end of the fuel rod.

During normal operation there is equilibrium between the heat produced in the fuel and that removed by the coolant. The radioactive fission products remain trapped in the fuel and are prevented by the cladding from contact with the coolant. The fuel and the cladding form a first barrier against the release of radioactive substances.

Mismatch of heat generation and removal can result in fuel overheating and cladding failure. In extreme cases the fuel will melt. The cladding may also be damaged in fabrication or during operation through mechanical interaction with the uranium pellets.

For reasons of economy, the core designer aims at achieving as high an average fuel heat rating as possible without overheating the fuel. For sufficient cooling of the hottest fuel rod, the maximum surface heat flux must be limited. In practice the maximum linear heat rating is set at about 400 watts per centimetre (W/cm) of rod length. This gives a maximum surface

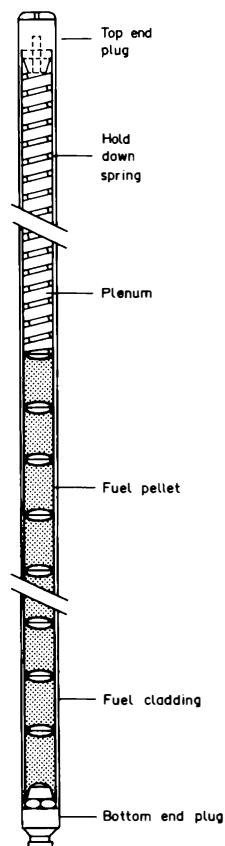


FIG. 3.4. Cutaway of fuel rod (schematic)

heat flux of about 110 W/cm^2 in the boiling water reactor and about 140 W/cm^2 in the pressurized water reactor.

The behaviour of the fuel rod during operation depends on a complexity of metallurgical, mechanical, thermal and chemical factors. The composition of the fuel changes with time, since the fissile material is depleted and fission products build up as energy is released. A measure of the cumulative energy release is the fuel *burn-up*, i.e. the product of the thermal power per unit of weight of uranium and the operating time in full power days. A commonly used unit for the burn-up is “megawattdays per kilogram uranium” (MWd/kg U). The burn-up is mainly determined by conditions of reactor physics and metallurgy.

The operating cycle is normally 1 year. During refuelling at the end of the operating period, about one-third of the fuel in the core of a pressurized water reactor and about one-quarter of that in a boiling water reactor is

changed. The relation between burn-up, E (MWd/kg), specific thermal power, P (MWth/ton), and the number of full power hours, T , is

$$E = PT/24,000n$$

where n is the fraction of the core fuel charged and discharged. Typical values for burn-up and fuel throughput are shown in Table 3.1.

For economic reasons, it is desirable to extend the burn-up as much as possible without increasing the number of fuel failures. Fuel failures may be systematic and result from faulty design, fabrication or operation. Or they may be stochastic, as a result of variations in material properties or of defects. Systematic failure can be prevented by modifying fuel design and fabrication and by establishing detailed operating rules. The probability of stochastic failure can be minimized by thorough quality control of materials and fabrication and by adequate safety margins in fuel design.

TABLE 3.1. *Typical fuel throughput and composition in light water reactors*

	Unit	Pressurized water reactor	Boiling water reactor
Electrical power	MWeI	1000	1000
Thermal power	MWth	3077	3067
Specific power	MWth/ton	37.5	23.8
Burn-up (average)	MWd/kg	33	27.5
Time between refuelling	days	365	365
Full power hours per operating cycle	hours	6000	6000
<i>Fresh fuel</i>			
Uranium-235	kg/year	900	840
Uranium-238	kg/year	26,450	31,320
Total	kg/year	27,440	32,260
<i>Spent fuel</i>			
Uranium-235	kg/year	220	233
Total uranium	kg/year	26,150	31,100
Fissile plutonium	kg/year	170	200
Total plutonium	kg/year	250	280

3.3 Fission Power

3.3.1 Neutron balance

The energy in a nuclear reactor is generated by the *fission* of heavy nuclei with neutrons. Most of the energy is released as kinetic energy in the fission products. Due to the slowing down of the fission products in the fuel, which occurs in a hundredth of a millimetre, their kinetic energy is converted into heat. It is this “friction heat” which is transferred to the coolant and utilized to raise steam.

On average, two to three new neutrons are emitted during fission. If at

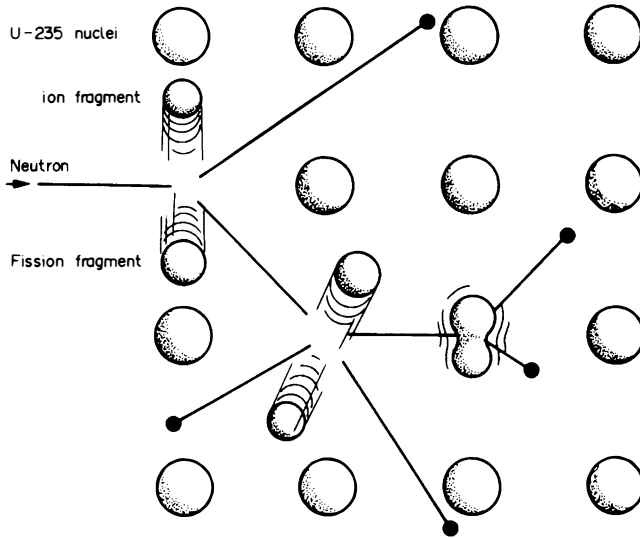
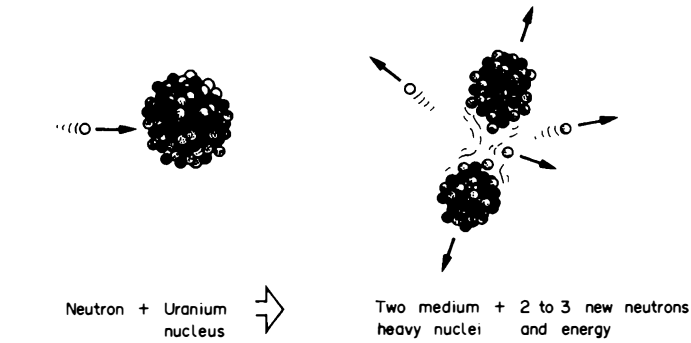


FIG. 3.5. Above: The fission process. Below: Three steps in a chain reaction

least one of these neutrons can be made to undergo another fission, a *nuclear chain reaction* results (Fig. 3.5). This is no simple condition to satisfy, since neutrons are easily absorbed by non-fissionable nuclei or escape from the system by leakage.

The emitted neutrons have a high velocity, typical of *fast neutrons*. If their speed is reduced, the probability of new fissions increases. Neutrons are slowed down if they are made to collide with light nuclei in a *moderator*. In a light water reactor ordinary water with its light hydrogen nuclei acts as

the moderator. The energy of the neutrons is reduced to become almost in balance with the thermokinetic energy of the moderator atoms. This occurs with *thermal neutrons* in a thermal reactor.

The only naturally occurring *fissile* (fissionable with thermal neutrons) nuclide is uranium-235, of which 0.71% is present in natural uranium. Raising the uranium-235 content of the uranium increases the possibility of fission. Such *enriched uranium* is produced in special enrichment plants in several countries around the world. Light water reactors use uranium with 2–4% uranium-235.

The rest of the uranium is uranium-238. This nucleus can undergo fission with fast neutrons but not with thermal neutrons. If a neutron is absorbed by uranium-238, the nucleus is converted into *plutonium-239*, which is also fissile. As a matter of fact, the fission of self-generated plutonium accounts for about half the energy generated in a typical light water reactor. Nevertheless, substantial quantities of plutonium remain in the spent fuel removed from the reactor (Table 3.1). This plutonium may be recovered by chemical reprocessing of the spent fuel, and may be re-used after mixing with uranium. Large-scale reprocessing plants for light water reactor fuel exist in France, Great Britain and elsewhere.

The best conditions for a chain reaction are obtained when fuel and moderator are separate. The fission neutrons escape from the fuel into the moderator where they are slowed down. They then return to the fuel where new fission neutrons are produced (Fig. 3.5). In equilibrium, the number of neutrons and therefore the fission rate and heat generation are constant. The level of equilibrium is determined by the efficiency of heat removal.

The neutron population in a reactor bears some resemblance to a very thin gas, filling the core. In order to minimize neutron leakage, the core is surrounded by a *reflector* which scatters the neutrons back into the core, acting as a kind of wall for the neutrons. The reflector of a light water reactor is a layer of water around the core.

Neutron balance is achieved when the number of neutrons produced is exactly equal to the number lost by absorption in the core and by leakage out of the reactor. The ratio of the number produced to the number lost is called the (effective) *multiplication factor*, k . At criticality, $k = 1$. Depending on whether k is greater or less than 1, the neutron population and thus the reactor power increases or decreases. The relative deviation from 1 is called *reactivity*, and is denoted ρ . By definition

$$\rho = (k - 1)/k$$

Reactivity is normally measured in percent. Positive (negative) reactivity is known as excess (deficit) reactivity. Correspondingly, the reactor is said to be supercritical or subcritical.

The product of neutron density (n/cm^3) and neutron velocity (cm/s) is called the *neutron flux* ($\text{n/cm}^2 \text{ s}$). There is a simple (approximate) relation

between the neutron flux, ϕ , and the thermal power generated in the fuel

$$\phi = 2.2 \times 10^{12} P/e$$

where P is the specific thermal power in megawatts per ton of fuel and e the enrichment in weight percent. If, for example, the specific power is 25 MW/ton and the enrichment 2.5 w/o, the neutron flux is 2.2×10^{13} n/cm²s.

3.3.2 Power distribution

The neutron flux and thus the power density is not uniform within the reactor, but varies both radially and axially. It decreases towards the boundary between the core and the reflector. The flux distribution also changes slowly with operating time since the composition of the fuel and therefore its reactivity changes. The ratio between maximum and average power density is called the *form factor*. For economic and safety-related reasons, the reactor is designed and operated so that the form factor is kept as low as possible. In practice, the total form factor is between 1.5 and 2.5 in a light water reactor.

Figure 3.6 shows a measured axial power distribution in a boiling water

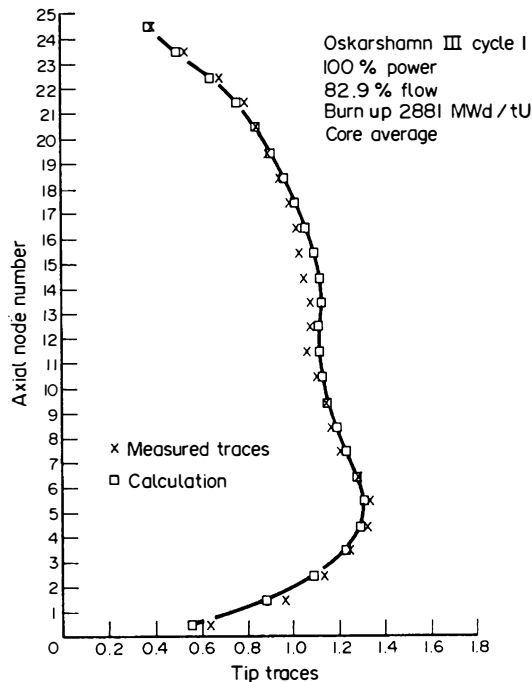


FIG. 3.6. Axial power distribution in a boiling water reactor (Oskarshamn III, cycle 1, 120 full power days). From S Lundberg, CASMO-3/SIMULATE-3 Core Follow Calculations, VTT Symposium 79. Status of Reactor Calculations in the Nordic Countries, Technical Research Centre of Finland, 1987

reactor. In this case the power distribution is displaced to the bottom of the reactor and the axial form factor is about 1.35. The pear-shaped axial power distribution is typical of a BWR and is due to the effect of coolant density on reactivity.

Figure 3.7 gives an example of the radial power distribution in a pressurized water reactor (Ringhals 2). The numerical values in the two-dimen-

Ringhals 4 Cycle 4

Assembly power

Burnup 5810 MWd / tU

Calculation

XXXX

Measurement

YYYY

Calc - meas

ZZ

	H	G	F	E	D	C	B	A
8	1203 1188 15	1092 1073	1064 1051	1018 1010	1261 1257	1012 1007	998 995	884 881
9		1100 1080 20	1254 1234 21	1067 1057 10	864 864 0	984 986 -2	1185 1185 1	637 641 -3
10		1253 1245 8	1225 1218 8	916 914 2	1163 1168 -5	857 862 -5	1041 1047 -5	
		1064 1068 -4	915 922 -7	1290 1293 -3	887 895 -8	1176 1186 -10	692 697 5	
12		864 866 2	1163 1173 -10	886 896 -10	1194 1209 -15	680 690 -9		
13		985 983 2	858 865 -7	1176 1192 -15	681 688 -8			
14		1187 1183 4	1043 1043 -1	693 691 1				
15		638 638 0						

FIG. 3.7 Radial power distribution in a pressurized water reactor (Ringhals 4, cycle 4, 151 full power days). From E B Jonsson et al, *CASMO-3/MBS Benchmark Calculations on Ringhals PWR*, Paper at The International Nuclear Simulation Symposium and Mathematical Modelling Workshop, 13-15 October 1987, Schliersee, West Germany

sional table represent the normalized calculated and measured (symmetrized) power per fuel assembly in a quadrant of the reactor core. The variation in the assembly power is mainly due to the different burn-up levels of the assemblies. The radial form factor is 1.27 in this case.

The power distribution is affected when highly neutron-absorbing material is inserted into or withdrawn from the core. Boron is a strong neutron absorber contained in the *control rods* of the boiling water reactor. The control rods are partially inserted into the core (from below) at the start of the operating cycle to compensate for the excess reactivity of the unirradiated fuel. They are then withdrawn (downwards) as burn-up increases and reactivity decreases. In pressurized water reactors, soluble boron is used in the moderator-coolant to control the long-term variation of the reactivity.

An important task of the control rods in both reactor types is to quickly reduce the reactor power when the need arises. This is called *scram*. The rods are then rapidly pushed into the core, thereby interrupting the nuclear chain reaction and bringing the reactor to a subcritical state.

Another way of controlling the neutron flux is to change the water density which affects the efficiency of the water as a moderator. In boiling water reactors, this can be done by regulating the speed of the main recirculation pumps which determines the *coolant flow* through the core. Any decrease in the coolant flow causes an increase in steam generation, i.e. the density of the moderator decreases, which means that the neutron flux and the power decreases. Correspondingly, a power increase is achieved by increasing the speed of the recirculation pumps.

If the moderator is suddenly lost, for example during a sharp power increase causing steam flashing and expulsion of water from the core, the nuclear chain reaction will immediately cease. For this reason, it is physically impossible for a reactor to “explode like an atom bomb” This inherent characteristic of the light water reactor was demonstrated in reactor experiments in the early 1950s.

3.3.3 Reactor kinetics

The chain reaction is maintained when on average one of the neutrons emitted during fission is made to strike another fissionable nucleus and cause it to fission and emit a new generation of neutrons. The time between two generations depends on the number of collisions with the moderator and the time between the collisions. The generation time is less than 0.0001 second (100 microseconds) in a light water reactor.

If the generation time alone were the determining factor, the neutron flux and thus the fission rate and the nuclear power would change very quickly at the slightest deviation from criticality. It would be impossible to control the chain reaction by mechanical devices, such as control rods. Fortunately,

the processes do not occur so quickly, because of the decisive role played by the *delayed* neutrons. These neutrons are emitted by particular fission products and appear from a fraction of a second to a few minutes after the fission event itself. The delayed neutrons have in effect a much longer “lifetime” than the *prompt* neutrons which are emitted directly during fission.

The number of delayed neutrons relative to the number of prompt neutrons is a nuclide-specific parameter. The delayed neutron fraction is 0.65% for uranium-235, 1.48% for uranium-238 and 0.21% for plutonium-239. For small deviations from criticality, the “effective” neutron lifetime, taking into account the delayed neutrons, is about 80 milliseconds in a uranium-235 system, i.e. about three orders of magnitude larger than the prompt neutron lifetime. This has a profound effect on the response of the reactor to reactivity disturbances as illustrated in Figs. 3.8 and 3.9.

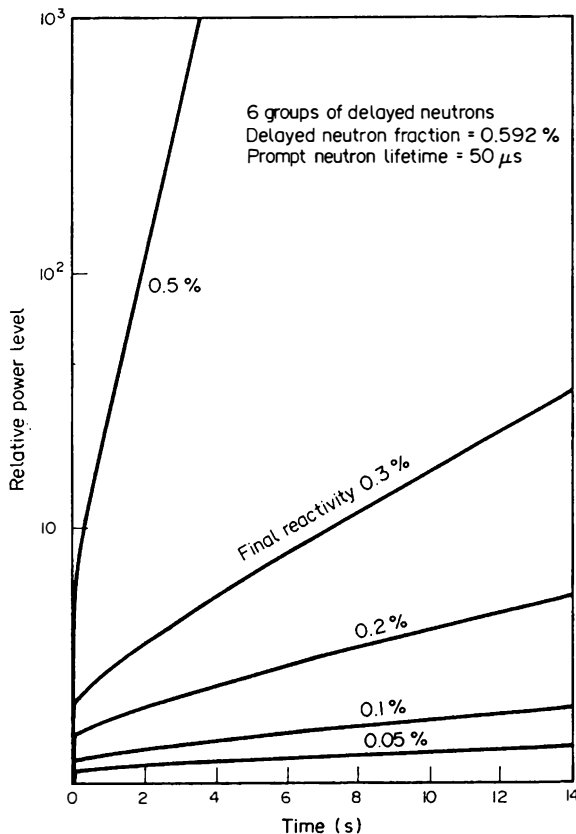


FIG. 3.8. Relative fission power level following a positive step change of reactivity. From N-G Sjöstrand, private communication, Chalmers Institute of Technology, 1987

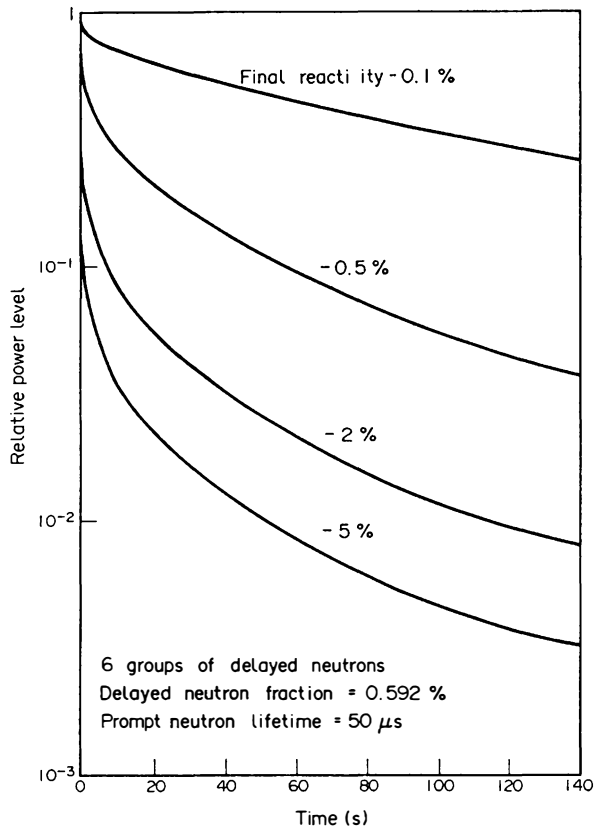


FIG. 3.9. Relative fission power level following a negative step change of reactivity. From N-G Sjöstrand, private communication, Chalmers Institute of Technology, 1987

Figure 3.8 shows that the power increase is relatively slow for moderately positive reactivity, so that power control with movable rods presents no problem. Figure 3.9 illustrates that the power decreases rapidly when the control rods are inserted into the core making the reactor subcritical.

In reactor kinetics, reactivity is often expressed in terms of “dollars” i.e. the ratio of the reactivity to the delayed neutron fraction. One dollar corresponds to a reactivity of 0.65% in a uranium-235 system. With a reactivity of 1 dollar, the reactor is said to be *prompt critical*, since criticality is attained when considering only the prompt neutrons. In practice, the reactivity involved during “normal” reactor transients is usually of the order of cents or less. The corresponding power response is rather slow as can be inferred from Figs. 3.8 and 3.9.

A reactivity of 1 dollar or more induces a fast *power excursion*, since the delayed neutrons are more or less ineffective. However, the power excursion

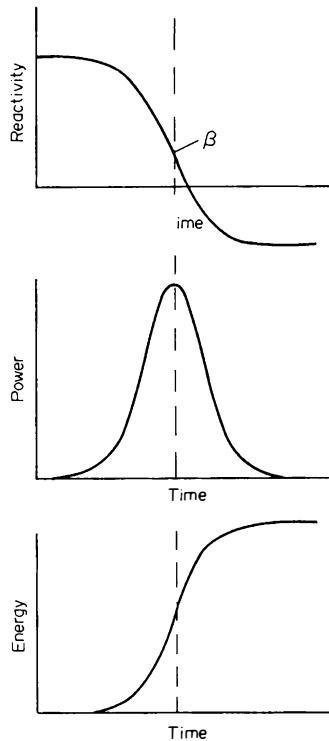


FIG. 3.10. Model calculation of reactivity, power, and energy in a superprompt self-limited excursion

sion will be mitigated and terminated by the introduction of negative reactivity. The result will be a power burst as illustrated in Fig. 3.10. The negative reactivity is obtained by inherent feedback effects, acting promptly, and by rapid insertion of the control rods (scram). The action of the control rods will be delayed a few seconds due to the actuation time and mechanical inertia. While a sudden reactivity insertion of 1 dollar or more is difficult to envisage in a light water reactor during power operation, prompt and even superprompt criticality can occur during the start-up procedure at essentially zero power.

3.3.4 Reactivity coefficients

The reactivity of a reactor depends on the physical state, e.g. the temperature and density of the moderator-coolant and the temperature and composition of the fuel. The reactivity change associated with a small change of a state variable is called the *reactivity coefficient* (of the state variable). The most important reactivity coefficients from the point of view of safety are:

- the fuel temperature coefficient,
- the moderator temperature coefficient,
- the coolant void coefficient.

The main component of the *fuel temperature coefficient* is known as the *Doppler coefficient*. The Doppler effect arises when the neutron absorption in uranium-238 changes in response to a change in temperature. It is negative, i.e. the reactivity decreases when the fuel temperature increases. The Doppler effect is of great importance for the stable operation of the reactor. Power variations due to small perturbations of the normal operating state will be slow and damped. The Doppler effect also plays a vital role for the limitation of fast power excursions.

The magnitude of the fuel temperature coefficient depends on the state of the fuel and the reactor. The Doppler coefficient becomes less negative with increasing fuel temperature. In oxide fuel there will be a positive contribution to the fuel temperature coefficient as burn-up proceeds due to the build-up of plutonium-239, and a negative contribution from plutonium-240. The net effect is small in light water reactors. In boiling water reactors the formation of steam, which reduces moderator density, will make the coefficient more negative.

The *moderator temperature coefficient* in a light water reactor is, in general, strongly negative at operating temperature. In boiling water reactors, the withdrawal of the control rods compensates for the decrease in reactivity during start-up. In pressurized water reactors, the corresponding reactivity compensation is achieved by reducing the boron concentration in the moderator. The moderator temperature coefficient is affected by the boron concentration so that a high boron concentration and low temperature (room temperature) leads to a slightly positive temperature coefficient.

The density coefficient of the coolant, or the *void coefficient*, is of importance primarily in the boiling water reactor. An increase in the relative steam volume or the void fraction in the core leads to a decrease in reactivity, i.e. the void coefficient is negative. The negative void coefficient has a stabilizing effect on the reactor power in the boiling water reactor, like the negative moderator temperature coefficient in the pressurized water reactor. The reactivity feedback effect is delayed by the time it takes for the heat to redistribute in the fuel and transfer to the coolant (cf 3.4.2).

The negative void coefficient is an inherent characteristic of normal light water reactors. In contrast, the Chernobyl type of graphite moderated, boiling water cooled reactor normally has a positive void coefficient. This has a destabilizing effect on reactor power which must be counteracted by a fast control system. The positive void coefficient was a contributing factor to the Chernobyl accident (see 13.7.4).

In a boiling water reactor, an increase of the system pressure leads to a reduction of the void fraction and thus to a reactivity increase. This means

that the reactor has a positive *pressure coefficient* of reactivity. The pressure must therefore be carefully controlled and sudden large increases avoided. In fact, the first commercial demonstration BWR, the Dresden reactor in the USA, was designed with a two-stage steam pressure system in order to decouple changes in turbine demand from the positive pressure coefficient of the boiling core.

Table 3.2 provides typical values for reactivity coefficients in light water reactors. The reactivity is expressed in pcm (“pour cent milles” $1 \text{ pcm} = 10^{-5}$), which is a common unit for small reactivity contributions.

TABLE 3.2. *Typical reactivity coefficients*

Reactivity coefficient	Unit	Boiling water reactor	Pressurized water reactor
Fuel temperature (Doppler coefficient)	pcm/°C	2	2.5
Moderator temperature (operating temperature)	pcm/°C	30	15
Moderator temperature (room temperature)	pcm/°C	5	+ 2
Coolant density (void coefficient)	pcm/vol% steam	160	not applicable
Boron content (operating temperature)	pcm/ppm B	not applicable	12

3.3.5 Reactor stability

The physical state of a reactor varies with the power level through changes in temperature, density, etc. As a consequence, power variations cause reactivity changes, which cause power changes. This phenomenon is known as reactivity *feedback*. If the feedback is positive, a power increase results in a reactivity rise and the reactor is unstable. Negative feedback is required for stability.

Feedback effects can be inherent (passive), such as those due to heating, or engineered (active) in the form of control systems. An inherently unstable reactor can be stabilized by means of a control system. The normal power control system uses signals from neutron flux detectors in the core to operate control rods. In the boiling water reactor, power control is also achieved by varying the coolant mass flow and thereby the moderator density in the core by regulating the speed of the main recirculation pumps.

A linear feedback effect is characterized by the magnitude of the reactivity change and the time delay relative to the power change. The magnitude is expressed by a reactivity coefficient, and the delay by a *time constant*. The time constant is a measure of the rate of change of the state variable affected by the power change.

Feedback due to heating is small at low power and grows with the power

level. In order to adequately describe the feedback, it is necessary to consider the temperatures of the fuel, cladding and coolant separately with reference to the effect of power on the temperatures and the effect of the temperatures on reactivity. The time constants of the various parts depend on heat capacities, heat transfer coefficients and mass flow rates. The overall time constant for heat transfer from fuel to coolant is typically about 5 seconds for LWR fuel.

Two extreme cases are of interest. In very large, fast power excursions the time scale of the power change is much smaller than the time constant for heat transfer from fuel to coolant. The heat loss can then be neglected and the rate of fuel temperature rise is directly proportional to the instantaneous power density. This means that the negative reactivity feedback due to the Doppler effect acts promptly in response to the deposited energy. Model calculations (301) for a superprompt, Doppler-limited power excursion are illustrated in Fig. 3.10.

At the opposite extreme when power changes very slowly, the temperatures will at any time be in equilibrium with the power level at that time. It is then possible to define an overall *power coefficient* as a weighted sum of the individual reactivity coefficients (302). A positive power coefficient is autocatalytic, i.e. it causes a monotonic build-up of any small deviation of the power from its equilibrium level. A negative power coefficient may cause an aperiodic damping or a periodic oscillation of the power deviation. In the latter case, the amplitude may decrease, stay constant or increase with time, depending on the reactivity coefficients and time constants involved. Thus, a negative power coefficient alone does not ensure stability.

In order to investigate the stability of a reactor system, it is necessary to set up a complete model of the neutron kinetics and thermal hydraulics of the system, including all inherent and engineered feedback effects. The natural periods of oscillation are found by assuming small perturbations of the state variables from their equilibrium values. The reactor is stable if all natural oscillations are damped.

While the neutron kinetics equations are basically non-linear, it has been shown experimentally that LWRs behave as linear systems under normal operating conditions. When, however, unstable conditions are reached, small oscillations may grow large enough for non-linear effects to become important. This may result in *limit cycles* where the oscillation amplitudes are bounded.

An example of a case where the reactor may be unstable in spite of the power coefficient being negative, is the *void-induced feedback instability* in a boiling water reactor. If the void coefficient is sufficiently large and delayed in such a manner that the phase lag is greater than 90 degrees with respect to the power, divergent oscillations of the power level may arise. Early BWRs with natural circulation showed tendencies to this type of instability. Bounded power oscillations have also been observed in modern, forced-circulation BWRs at low-flow conditions.

Whilst the void-induced feedback instability affects the overall power level of the reactor, there may also be local instability in a fuel channel, known as *hydrodynamic instability* or channel instability. This type of instability can be thought of as an oscillation of the location of the boundary between the boiling and non-boiling part of the fuel channel. If the coolant flow and steam content is perturbed, the corresponding change in the two-phase pressure drop gives rise to a change in the single-phase pressure drop with the opposite sign, since the total pressure drop over the channels is kept constant. The heat balance then results in feedback effects on the coolant flow and steam content which may dampen or amplify the perturbation. Hydrodynamic instability is avoided by suitable orificing at the channel inlet, which reduces the relative effect of the pressure drop in the channel.

3.3.6 Excess reactivity

The reactivity which must be provided to achieve criticality at various operating conditions is known as excess reactivity. During start-up, the reactor is slowly heated from room temperature to operating temperature. Then the isothermal temperature coefficient of reactivity, i.e. the sum of the fuel and moderator temperature coefficients, is of interest. It varies with temperature and burn-up, but is on average strongly negative in the boiling water reactor, and about zero in the pressurized water reactor. This means that the difference in reactivity between room temperature and operating temperature, the *temperature defect*, is usually considerably larger in the boiling water reactor than in the pressurized water reactor. Because of the positive temperature coefficient at low temperatures, nuclear heat-up from room temperature is not allowed in the PWR, i.e. the reactor must not be made critical until operating temperature is reached.

Once the boiling water reactor has been brought to operating temperature, the power is increased by withdrawing the control rods. In the pressurized water reactor, this is achieved by reducing the concentration of boron in the moderator. Reactivity is then bound in the fuel temperature due to the negative Doppler coefficient, and, for the boiling water reactor, in the moderator as a result of the negative void coefficient. Table 3.3 shows

TABLE 3.3. Typical reactivity contributions

Contributions	Reactivity investment (percent)	
	BWR	PWR
Isothermal temperature defect.		
Cold to hot reactor	4.0	0–4.0
Fuel temperature hot reactor. Zero power to full power	1.0	
Moderator density hot reactor. Zero power to full power	3.0–4.5	not applicable

examples of the excess reactivities with respect to the cold critical reactor which have to be “inserted” in order to attain the operating state.

Conversely, during shutdown, compensation for items 2 and 3 in the table must be provided relatively quickly. This is achieved by inserting the control rods. There is more time at hand to balance the reactivity increase from hot to cold reactor.

3.3.7 Xenon poisoning

One of the nuclides formed during fission is iodine-135. This nuclide is unstable and forms xenon-135 which is also unstable and decays with 9.1 hours half-life. Xenon-135 is a very strong neutron absorber which steals neutrons from the chain reaction. This *xenon poisoning* is equivalent to a reactivity loss of about 3% during normal operation. Xenon poisoning also results in reactivity transients during start-up, shutdown and at power changes.

During the start-up of a xenon-free core, iodine-135 is first produced, as the fission processes get started. The iodine then decays to xenon and the reactivity decreases, which is compensated for by withdrawing the control rods (BWR) or reducing the moderator boron concentration (PWR). Since xenon is lost through neutron absorption or radioactive decay, equilibrium is reached after about 10 hours, when the amount of xenon produced is the same as the amount lost by absorption and decay.

At shutdown, iodine production ceases and xenon is no longer lost through neutron absorption (since the neutron flux disappears). The iodine left in the reactor after shutdown decays to xenon at the same time as the xenon loss decreases. The xenon concentration therefore *increases* after shutdown and reaches a maximum after about 10 hours (Fig. 3.11). As a result, it may be impossible to start the reactor from 5 to 10 hours after shutdown if there is not enough excess reactivity available to counteract the xenon poisoning. Figure 3.11 also shows a reactivity transient during a gradual power increase, starting 15 hours after shutdown.

Xenon poisoning represents an interesting case of positive reactivity feedback. An increase in neutron flux causes a drop in the xenon concentration due to increased neutron absorption. As a result, the reactivity increases and the feedback is positive. The counteracting increase in the xenon concentration is delayed for about 10 hours, since xenon is formed via iodine. The resulting instability of the power level is easily controlled because of the long time constants involved.

In geometrically large reactors, xenon poisoning can lead to instabilities in the power distribution over the reactor space, known as *xenon oscillations*. These oscillations arise because the xenon equilibrium is fundamentally unstable. If the reactor is large enough, one half of the reactor can act as a critical unit. The power distribution oscillates from one half to the other

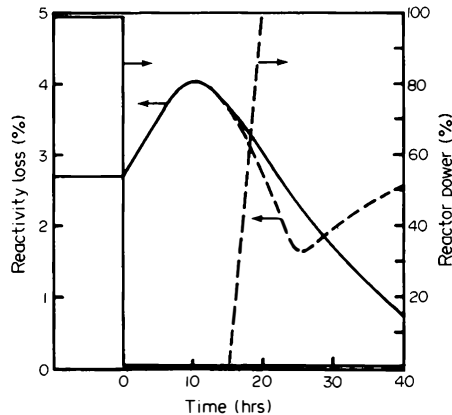


FIG. 3.11. Xenon transient after scram in a boiling water reactor. From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

with a period determined by the characteristic time for iodine and xenon decay. Controlling the xenon oscillations does not normally pose a problem, since the period is long.

3.3.8 Burnable absorbers

Excess reactivity must be built into the reactor core, not only to offset the negative reactivity coefficients and xenon poisoning but also to compensate for the reactivity decrease due to burn-up. This decrease occurs because of changes in the isotopic composition of the fuel (Table 3.1), and because neutrons are lost by absorption in the fission products. Roughly, the reactivity decreases linearly with burn-up. Increasing the enrichment of the fresh fuel is the means of increasing the level of reactivity.

For various reasons, *burnable absorbers* are usually added to fresh fuel. With boiling water reactors, the aim is to limit the amount of excess reactivity that has to be compensated by control rods. With pressurized water reactors, the prime purpose is to prevent the temperature coefficient of the moderator from becoming positive at operating temperature and full power, by decreasing the required boron concentration in the moderator.

The burnable absorber is a material with a high neutron absorption, such as gadolinium or boron. It reduces the reactivity of the fuel and is more or less completely depleted during the first operating cycle.

Boiling water reactors use gadolinia, Gd_2O_3 , which is either homogeneously mixed in 2–4% by weight with uranium dioxide or inserted between the fuel pellets in the form of thin plates. Figure 3.12 shows reactivity

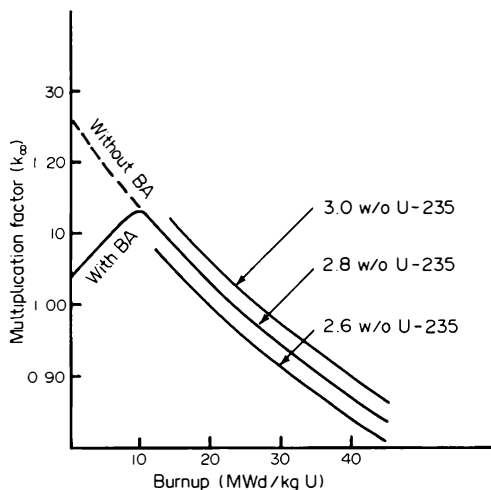


FIG. 3.12. Typical multiplication factor for a boiling water reactor fuel assembly. BA = burnable absorber. From E B Jonsson, private communication, Studsvik Nuclear, 1987

versus burn-up for a fuel assembly with various degrees of enrichment. The reactivity without burnable absorber is indicated by the dashed line.

Pressurized water reactors have mainly used boron as a burnable absorber in the form of boron glass rods in fuel rod positions. Figure 3.13 shows reactivity as a function of burn-up for a typical pressurized water reactor fuel assembly with the number of boron glass rods per assembly as

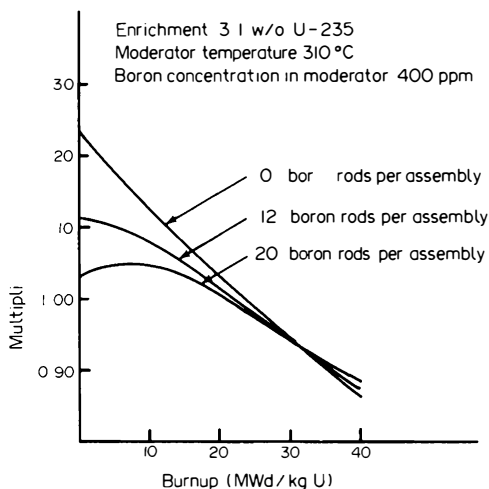


FIG. 3.13. Typical multiplication factor for a pressurized water reactor fuel assembly. From E B Jonsson, private communication, Studsvik Nuclear, 1987

parameter. Boron is not as strong an absorber as gadolinium and does not burn out completely during the operating cycle.

In practice, the reactor core contains fuel assemblies at different degrees of burn-up. At the beginning of an operating cycle, one-third to one-quarter of the core consists of fresh fuel. The remaining fuel assemblies will have been in the reactor for 1–3 operating cycles. The reactivity decrease during operation is compensated for in boiling water reactors by withdrawing the control rods, and in pressurized water reactors by reducing the boron concentration in the moderator. Ideally, at the end of the operating period, all control rods are withdrawn and all boron is removed.

3.3.9 Reactivity control

As explained above, reactivity control in boiling water reactors is mainly achieved by fixed (burnable) and movable absorbers. Reactivity can also be controlled to a certain extent by regulating the speed of the main recirculation pumps and thereby varying the coolant flow and steam generation in the core. The control rods are normally used for reactivity control during

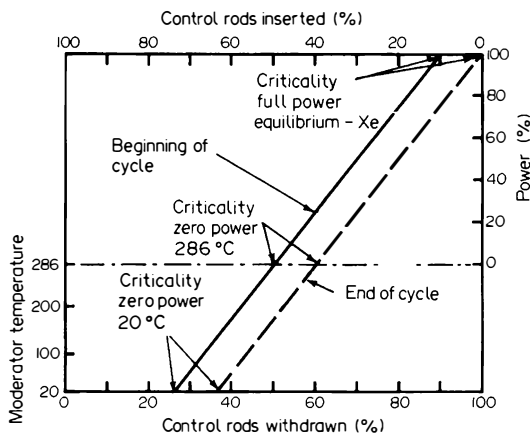


FIG. 3.14. Reactivity control during start-up and shutdown in a boiling water reactor (schematic). From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

start-up and shutdown, as shown in Fig. 3.14. The required number of control rods for criticality in different situations is also illustrated. For example, 50% of the control rods is sufficient to shut the reactor down at full power and to keep it sub-critical at operating temperature. About 75% of the control rods is required to cool down the core to room temperature in the most reactive condition at the beginning of an operating cycle.

Chemical shimming with boric acid in the moderator as well as with fixed

and movable absorbers are used for reactivity control in the pressurized water reactor. The control rods are used for fast power and moderator temperature changes during operation and for shutdown. Figure 3.15 shows an example of reactivity control during the first operating cycle. The initial decrease of the critical boron concentration corresponds to the xenon poisoning which reaches equilibrium at full power after a burn-up of about 0.15 MWd/kg U. When the burnable absorber is depleted, the critical boron concentration decreases linearly with burn-up. In order to keep the control-rod-free core safely subcritical, a boron concentration of 1233 ppm (parts per million) at operating temperature and 1235 ppm at room temperature is required in the example. The temperature defect is thus almost zero in this case.

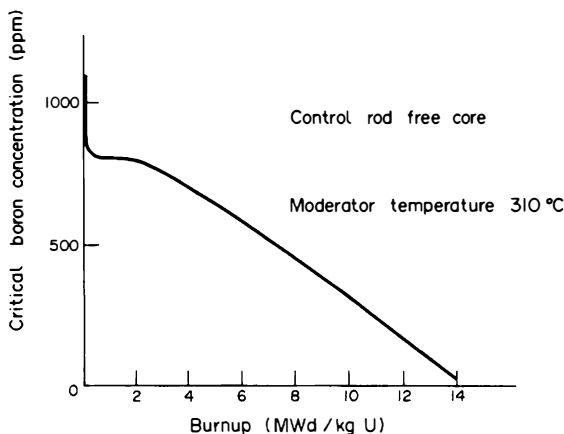


FIG. 3.15. Reactivity control during an operating cycle in a pressurized water reactor. From E B Jonsson, private communication, Studsvik Nuclear, 1987

3.4 Heat Transfer

3.4.1 Heat balance

Steady-state reactor operation is determined by two equilibrium conditions:

- neutron balance*, which means that the number of neutrons produced is equal to the number of neutrons lost so that the fission rate and thus the nuclear power is kept constant;
- heat balance*, which means that the heat produced in the core is equal to that removed by the coolant so that the fuel temperature is kept constant.

If heat balance is not maintained, for example due to lack of neutron balance, the fuel may overheat and melt or disrupt leading to the release of large quantities of radioactive substances.

Heat is transported by conduction in the fuel and transferred by convection to the coolant. In light water reactors, the water acts as coolant and moderator. Water has good heat transfer properties but requires high temperature and pressure for the efficient conversion of thermal energy to mechanical work.

The operating conditions are different in boiling water reactors and pressurized water reactors. This is shown by the vapour pressure curve which defines the temperature at which water turns into steam (Fig. 3.16). The curve represents corresponding values of saturation pressure and saturation temperature. For example, the saturation temperature is 100°C at atmospheric pressure (0.1 MPa) and 286°C at 7 MPa, which is the operating temperature in boiling water reactors. The operating pressure in a typical pressurized water reactor is 15.5 MPa, which corresponds to a saturation temperature of 345°C .

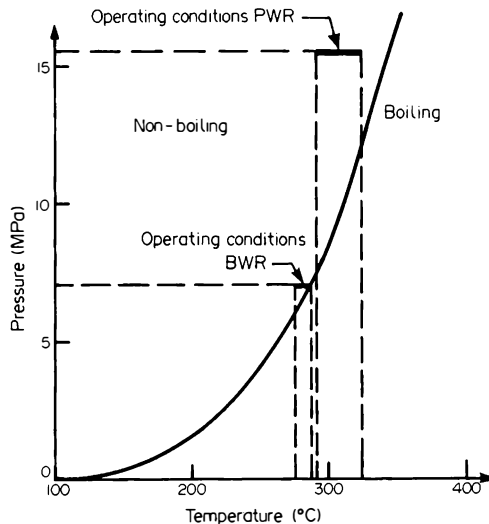


FIG. 3.16. Water temperature and pressure at saturation

In the boiling water reactor, steam is generated as the coolant flows upwards through the core. The average steam fraction at the core outlet is 6–15% by weight. No bulk boiling is permitted in pressurized water reactors. The average temperature of the water leaving the core is $20\text{--}30^{\circ}\text{C}$ lower than the saturation temperature at operating pressure (see Table 3.4).

TABLE 3.4. *Coolant data for a boiling water reactor (Forsmark 1) and a pressurized water reactor (Ringsbals 3)*

	Unit	Forsmark 1	Ringsbals 3
Electric output, net	MWeI	890*	915
Thermal output	MWth	2700	2783
Operating pressure	MPa	7	15.5
Saturation temperature	°C	286	345
Coolant flow rate	kg/s	10,400	12,860
Coolant temperature core inlet	°C	272	284
Coolant temperature core outlet	°C	286	323
Steam quality at core outlet	wt %	13	0

* Now 970 MWeI.

The heat balance means that the power generated in the fuel is equal to that transferred to the coolant. This can be expressed as

$$P = q (h_{\text{out}} - h_{\text{in}}) \quad (3.1)$$

where P = fuel heat generation (watt),

q = coolant mass flow (kg/s),

$h_{\text{out}} - h_{\text{in}}$ = increase in coolant enthalpy (joule/kg).

3.4.2 Heat conduction in the fuel

Uranium dioxide has a low thermal conductivity, which leads to a large temperature difference between the centre and the surface of the uranium pellet. A common criterion is that the centreline temperature should not exceed the melting point, about 2800°C. Typically, the peak centreline temperature at 100% power is about 1800°C. The gap between the pellet and the cladding represents a heat resistance and therefore a temperature drop. Similarly, the temperature drops in the cladding and the layers of oxide and corrosion products which build up on the clad wall during operation. The temperature distribution in a fuel rod is shown in Fig. 3.17

The thermal conductivity of uranium dioxide varies with temperature. In order to calculate the temperature drop, ΔT_k , from the centre of the pellet to the surface, it is convenient to use a mean value of the thermal conductivity. Then

$$\Delta T_k = P_l / 4\pi\lambda \quad (3.2)$$

where P_l = linear heat rate (W/m),

λ = mean thermal conductivity (W/m K).

The surface heat flux can be written:

$$\phi = P_l / 2\pi r \quad (3.3)$$

where ϕ = surface heat flux (W/m²),

r = pellet radius (m).

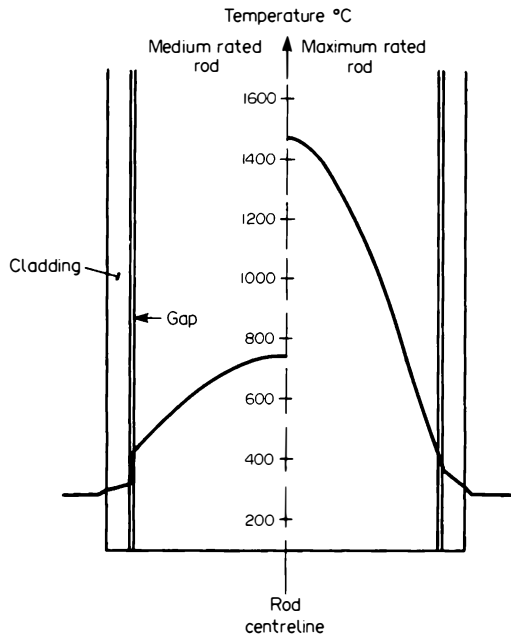


FIG. 3.17 Typical temperature profile of a fuel rod at the end of an operating cycle

Equations (3.2) and (3.3) show that for a given linear heat rate, the centre-line temperature is independent of the pellet radius and that, for the same surface heat flux, a reduced rod diameter results in a lower centreline temperature.

The temperature drop, ΔT_s , over the pellet-to-clad gap is difficult to calculate due to the irregular variation of the gap width and composition during operation. It may amount to a few hundred degrees. Formally:

$$\Delta T_s = \phi/k_g \quad (3.4)$$

where k_g = the gap heat conductance ($\text{W/m}^2 \text{K}$).

There are special calculational programmes and certain experimental data for estimating the gap conductance.

The temperature drop over the clad wall including surface deposits is typically about 100°C .

During steady-state operation, a large amount of sensible heat is stored in the hot fuel. When the operating conditions change, the heat is redistributed (Fig. 3.18). A sudden deterioration of the cooling conditions can cause high cladding temperatures, even if the reactor is quickly shut down. The rate of temperature change is governed by the time constant of the fuel which is typically about 5 seconds (cf 3.3.5).

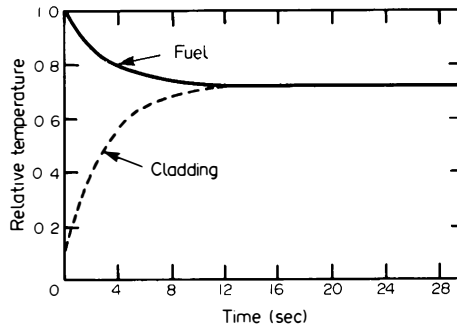


FIG. 3.18. Temperature variation in a fuel rod after a sudden loss of power and cooling

3.4.3 Heat transfer to the coolant

Heat is transferred from the cladding to the coolant by convection, which depends on several phenomena such as the coolant mass flow, viscosity, heat capacity and thermal conductivity. For calculational purposes, a *heat transfer coefficient* α ($\text{W/m}^2 \text{K}$) is defined by the following equation:

$$\phi = \alpha(T_w - T_c) \quad (3.5)$$

where ϕ = surface heat flux (W/m^2),
 T_w = clad wall temperature (K),
 T_c = coolant bulk temperature (K).

The relationship between the surface heat flux and the temperature difference $T_w - T_c$ is shown in Fig. 3.19. Branch 1 to 2 represents single-phase flow. The heat transfer coefficient increases with increasing mass flow.

At wall temperatures just above saturation, vapour bubbles start to form at the wall. At a somewhat higher temperature, the bubbles dissolve and condense in the coolant. This phenomenon is called *subcooled boiling*, branch 2 to 3, since the bulk temperature of the coolant is below saturation temperature.

When the bulk temperature of the coolant reaches the boiling point, a net generation of steam bubbles occurs. This is known as *nucleate boiling*, branch 3 to 4. Two-phase flow prevails and heat transfer is very efficient. At full nucleate boiling, the heat transfer coefficient increases proportional to the third power of the difference between the wall temperature and the saturation temperature, branch 4 to 5.

A *boiling crisis* is eventually reached, where the bubbling becomes so violent that the coolant cannot reach the heated surface and a vapour film with low heat conductivity forms at the surface. This is known as *film boil-*

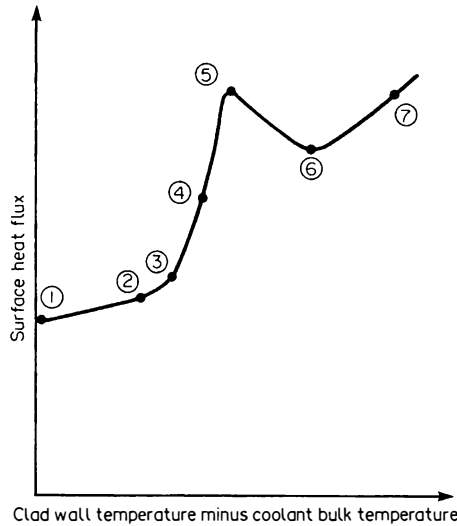


FIG. 3.19. Schematic forced convection boiling curve for a typical fuel rod bundle

ing. The heat transfer coefficient then decreases, even if the wall temperature is increased, branch 5 to 6. When the wall temperature has increased so much that heat radiation starts to contribute, the heat transfer rises again, branch 6 to 7.

The surface heat flux at which departure from nucleate boiling (DNB) occurs, i.e. where the heat transfer coefficient begins to fall, is called the *critical heat flux*. In a fuel rod, where the surface heat flux is determined by the nuclear power in the rod, the clad temperature will increase sharply when the critical point is reached. The transition boiling region from 5 to 6 can only be realized in a temperature-controlled experiment.

Figure 3.19 essentially applies to conditions in pressurized water reactors. In these reactors, net boiling is only permitted at the fuel rods with the highest power density. In boiling water reactors, water enters the core at a temperature below the saturation temperature. Fairly soon, subcooled boiling occurs in the coolant channel. When more heat is transferred to the coolant further up in the channel, nucleate boiling takes place. The bubbles grow and coalesce to form large bubbles which almost fill up the entire flow area. Additional steam generation results in *annular flow* when water flows partly along the clad wall and the box wall, and partly in the form of water drops in the steam flow.

The concept of critical heat flux also applies to boiling water reactors, but the boiling crisis mechanism is different from that at low void fraction typical of pressurized water reactors. At high void fraction there is a film of water on the clad wall and water droplets suspended in the steam flow.

Heat is mainly transferred by evaporation from the film surface to the steam. If the thickness of the water film is below a critical value, the film detaches from the clad wall resulting in a radical decrease in heat transfer. This phenomenon is called *dryout* (DO) and leads to a sharp increase of the clad temperature. The different boiling crisis mechanisms at low and high void fraction are illustrated in Fig. 3.20.

A large data base exists on the critical heat flux and heat transfer coefficient for real fuel assemblies. The data have been obtained through experiments with electrically heated rod bundles, where the power is increased or the coolant flow decreased in small steps until the critical heat flux is reached. During subcooled boiling, the critical heat flux is largely determined by the pressure, mass flow and coolant enthalpy. During net boiling, the void fraction is an additional and essential parameter, and the critical heat flux decreases as the void fraction increases.

The fuel assemblies are designed with a large margin to the critical heat flux during steady-state operation. The minimum ratio of critical to actual rod surface heat flux is at least 1.5 at full power. During transient conditions the ratio may temporarily fall below its stationary value. Experiments have shown that local, short duration exceedance of the critical heat flux does not threaten the integrity of the fuel rod.

3.4.4 Stored energy

In a typical boiling water reactor (Oskarshamn II), the average fuel temperature is 530°C, and the coolant temperature 270–286°C, where 286°C corresponds to the saturation temperature at a system pressure of 7 MPa. During heat-up to operating temperatures, energy is stored as sensible heat in the fuel, coolant, reactor vessel and internals in proportion to the respective mass, heat capacity and temperature difference relative to the ambient temperature. It is instructive to measure the stored energy in full power seconds (Table 3.5).

The energy storage means that the reactor acts as a buffer during changes

TABLE 3.5. *Stored energy in a boiling water reactor (Oskarshamn II, 595 MWel)*

Item	Stored energy (full power seconds)
Fuel from operating temperature to 286°C	4.7
Fuel from 286°C to 100°C	4.2
Reactor coolant from 286°C to 100°C	112
Subcooling of reactor coolant during normal operation	5.8
Reactor vessel and internals from 286°C to 100°C	43

Source: *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES Konsult AB, 1985

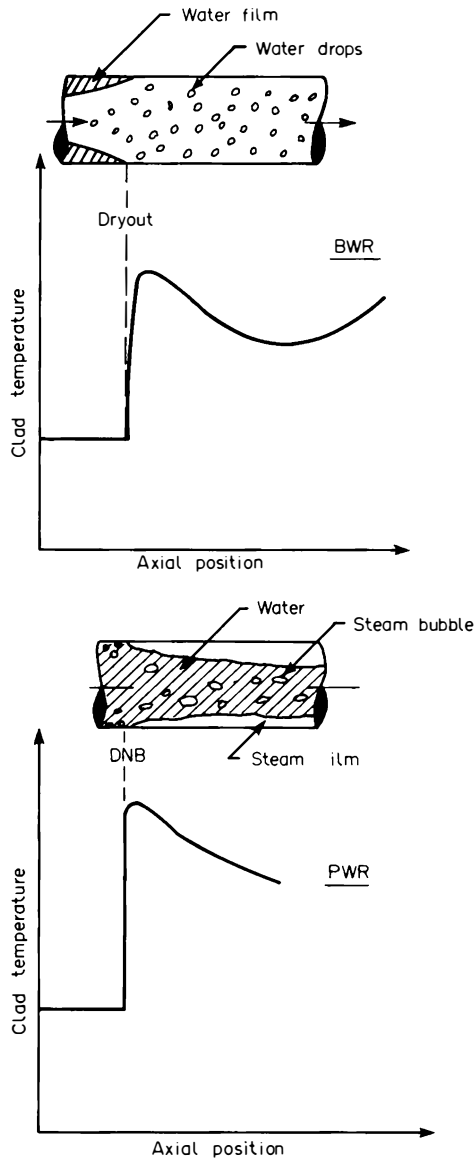


FIG. 3.20. Boiling crisis flux in pressurized water reactors and boiling water reactors. Adapted from R T Lahey, Jr and F J Moody, *The Thermal Hydraulics of a Boiling Water Nuclear Reactor*, American Nuclear Society, 1977

of the heat generated or heat removed from the reactor system. When the reactor is shut down, the stored energy is released (cf Fig. 3.18). The table shows that the fuel has a relatively small buffer capacity, whilst that of the

moderator and reactor vessel and internals is large. The energy content from operating temperature to saturation temperature is approximately the same as the energy content in the subcooling of the coolant. In core cooling calculations for reactor shutdown, it is a good approximation to assume that the entire reactor has an initial temperature of 286°C.

3.4.5. Decay heat

About 7% on the fission energy is released as radiation energy of the fission products. Even if the fission reactions stop when the reactor is shut down, energy continues to be released from the decay of the fission products, and it only decreases slowly. The decay heat is substantial in large reactors. The fuel must therefore be cooled to prevent overheating after the nuclear chain reaction has ceased. The decay heat cannot be “switched off”

Decay heat depends on burn-up, i.e. the reactor power and operating time, and on the time after shutdown, the *cooling time*. If the irradiation time at full power is T (sec) and the cooling time t (sec), the following approximate formula holds

$$P_d(t, T) = 0.622 P_o (t^{0.2} - (T + t)^{-0.2})$$

where P_d is the decay heat power and P_o the reactor power. The formula gives correct results within a factor of two for cooling times between 10 seconds and 100 days. The contribution from beta and gamma radiation (see 6.1.2) is about equal.

For more accurate calculations, the composition of the fuel must be taken into account, since the fission product yield depends on the kind of nuclide undergoing fission. Detailed tables of decay heat for different nuclides have been published (307). Figure 3.2.1 shows the decay power from fission products produced during the fission of uranium-235 at a steady rate over an (infinitely) long period of time.

If the values on the curve are represented by $F(t, \infty)$, the decay heat after cooling time t and operating time T is given by

$$F(t, T) = F(t, \infty) - F(T + t, \infty)$$

The decay heat for plutonium-239 is somewhat lower than that of uranium-235.

In practice, the decay heat of individual fuel assemblies and the entire reactor core increases during the operating cycle. The decay heat is lowest shortly after refuelling, since the core then contains a large part of fresh fuel. It then builds up within about a month to a level close to that existing towards the end of the operating cycle. The decay heat is highest in the fuel assemblies which have reached their target burn-up and which are ready to be removed from the core, but the increase is small after the first operating cycle.

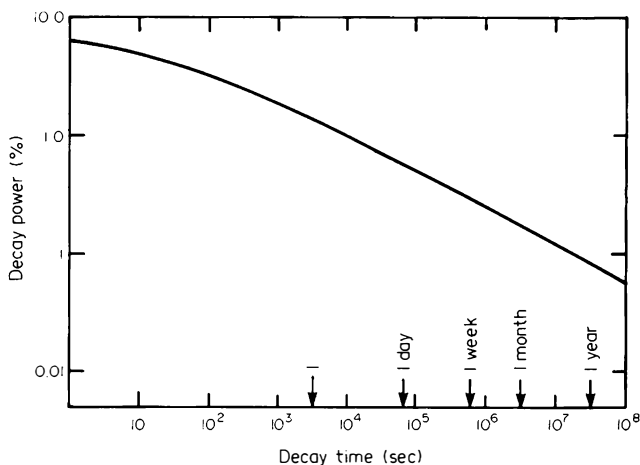


FIG. 3.21. The decay power of fission products from U-235 fission. The decay power is given in percent of the fission power

3.4.6 Metal–water reaction

Another heat source which can be very important under accident conditions is the metal–water reaction between zirconium and steam. The metal–water reaction causes oxidation of the cladding, which is favoured by high temperature. Heat is released during the reaction, thereby further increasing the temperature and the reaction rate.

Normally, the temperature of the cladding is some ten degrees higher than that of the coolant, i.e. about 330–350°C. If the cooling deteriorates and the critical heat flux is exceeded, the clad temperature will suddenly increase by several hundred degrees. At temperatures of 880–900°C, clad oxidation begins to increase, leading to the formation of hydrogen and the release of heat, as expressed by:



When 1 kg of zirconium oxidizes, 0.5 m³ hydrogen and 6500 kJ of heat are formed.

The reaction rate depends strongly on the temperature and on the thickness of the oxide deposit (Fig. 3.22). At 1200°C the heat release is about as large as the average nuclear power in the fuel during normal operation. Within 15 minutes, about 15% of the cladding is oxidized. The hydrogen and the heat produced make the cladding brittle. Criteria have been established for limiting clad oxidation in accident situations (see 9.2.1).

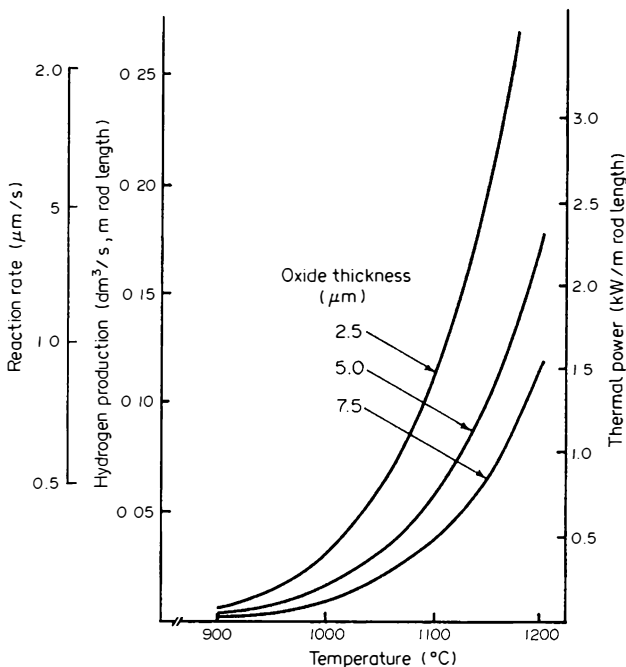


FIG. 3.22. Reaction rate, hydrogen production and heat generation for the zirconium–water reaction. From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

3.4.7 Fuel-coolant interaction

Fast reactivity insertion in a reactor may result in a power burst as illustrated in Fig. 3.10. The energy deposited in the fuel causes adiabatic heating which may damage or even destroy the fuel if the burst is sufficiently rapid and energetic. The damaged fuel interacts with the coolant water, converting nuclear energy into mechanical energy which could conceivably disarrange the core or breach the primary system.

Early experiments in the SPERT and TREAT reactor facilities in the USA indicated that the failure consequences were small for total energy depositions below 300 cal/g UO_2 (1250 J/g) for both irradiated and unirradiated UO_2 fuel rods subjected to rapid power excursions (308). This may be compared with the energy stored in the hottest fuel pellet during normal operation, which is about 125 cal/g UO_2 . Critical surface heat flux is reached at approximately 170 cal/g UO_2 . Substantial clad melting occurs at about 280 cal/g UO_2 .

In the 300–500 cal/g UO_2 range and fuel fails and is broken into pieces before the cladding melts. The conversion of nuclear-to-mechanical energy

is estimated at less than 1%. The extent of metal–water reaction, discussed in the previous section, increases with the energy deposition reaching about 50% of the cladding at 500 cal/g UO_2 .

For energy depositions in excess of 500 cal/g UO_2 the fuel is completely fragmented, partly in finely divided particulate form. The fragments will cause instantaneous vaporization of the coolant water and essentially a 100% metal–water reaction. The conversion of nuclear-to-mechanical energy may be 1–3%. Experiments on fuel–coolant interaction have continued at the PBF facility in the USA and the NSSR in Japan. The detailed mechanisms occurring when molten fuel interacts with coolant are not yet completely understood (309).

3.5 Structural Mechanics

3.5.1 Pressurized components and systems

The integrity of the primary system pressure boundary under all possible operating conditions is of prime importance, since rupture or leakage results in coolant loss which can lead to fuel overheating. In addition, the leaktightness of the primary system prevents any radioactive substances in the coolant from spreading to the environment. In the boiling water reactor, the pressurized components of the primary system consist of the reactor vessel with connecting pipelines, pumps and valves, and in the pressurized water reactor they also include the pressurizer and steam generator (Fig. 3.2).

There is a long tradition of designing pressurized systems. Modern research has provided more insight into material properties and failure causes. The reactor vessel poses special problems through its large size and the catastrophic consequences of failure. In practice, it must be possible to rule out failure of the reactor vessel. This can be achieved by proper design and choice of material as well as by stringent control during manufacture, testing and recurrent inspection.

3.5.2 Fracture mechanics

A pressure vessel may rupture in one of two ways. If the mechanical stress exceeds the yield stress of the material, the load-bearing section starts to deform plastically. If the load is increased, the section deforms more and more and the load-bearing area becomes smaller until it ultimately breaks. Failure which is preceded by plastic deformation is called *ductile fracture*. Designing a component to prevent ductile fracture is a well-proven procedure for which there are generally accepted standards. The loads that the component has to withstand in abnormal situations and the variations in material properties are also taken into consideration in the design process.

In the design against ductile fracture it is tacitly assumed that the material

is essentially homogeneous and perfect. In practice, various types of defects occur, e.g. small cracks and inhomogeneities. These defects arise during the manufacture, processing and welding of steel. Under load, the stress at the tip of a crack will be greatly magnified and can cause the crack to grow. Under certain conditions the crack extends indefinitely, resulting in fracture. This mode of failure is called non-ductile or *brittle fracture*. Brittle fracture occurs very quickly over the entire section before any major plastic deformation takes place.

The resistance of a material to crack extension is known as toughness or *fracture toughness*. Pressure vessel steel is characterized by a high toughness and a relatively low yield stress. The fracture toughness depends strongly on temperature. It is low at low temperatures and high at high temperatures (Fig. 3.23). The transition takes place within a narrow interval, characterized by the *transition temperature*. The transition temperature increases as the neutron irradiation increases which also causes the upper ductility level to decrease.

The working region of the reactor vessel is above the transition temperature, i.e. at temperatures where the ductility is high. In this region, a crack can only grow in a slow and stable manner and lead to ductile fracture when the load-bearing section becomes sufficiently small. Widespread plastic deformation is required in front of the crack, i.e. the yield stress must be exceeded in the entire wall section and not just at the tip of the crack. Since the yield stress is substantially higher than the stresses that may arise in the reactor vessel, unstable rapid crack extension is not possible in the ductile region.

In the brittle region, unstable crack growth can occur at stress levels well below the yield stress. In the transition region between the brittle area at low temperatures and the ductile area at high temperatures, a limited plastic deformation takes place in front of the crack and the failure mode changes successively from brittle to ductile.

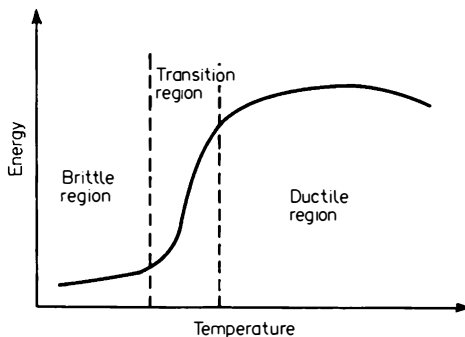


FIG. 3.23. Typical impact toughness ("Charpy V-notch energy") curve for pressure vessel steel. Impact toughness is a measure of the energy absorbed before a sample of the material fails during impact testing

In order to design the reactor vessel to avoid brittle fracture, the methods of *fracture mechanics* are used. The interaction between three factors are treated:

- fracture toughness of the material,
- occurrence and type of defects,
- stress, strain and energy fields ahead of defects.

For characterizing the stress field around the tip of a crack the *stress intensity factor*, K_I , is used. In order for unstable crack growth to occur, the stress intensity factor must be larger or equal to a critical value, K_{IC} , which is a measure of the fracture toughness:

$$K_I > K_{IC} \quad (3.6)$$

K_{IC} is a characteristic of the material which is determined in carefully prescribed experiments. Besides temperature and irradiation, it depends on the composition and structure of the material.

The condition for unstable crack growth can also be expressed by a *critical crack length*, which is calculated from the fracture toughness, stress field and crack geometry. If the length of the crack is greater than the critical length, the crack will quickly grow to fracture, while a crack shorter than critical will not develop into fracture. If the critical crack length is greater than the thickness of the section, “*leak before break*” will result. This is often the case for conventional pressure vessels and for the pipelines of reactor systems, but not for reactor vessels because of their large size.

The condition in (3.6) is strictly applicable only within the elastic area of the material. The theory is known as linear elastic fracture mechanics (LEFM). In conditions where there is plastic yielding in a large volume around the crack tip, the elastic-plastic fracture mechanics (EPFM) is used. The various areas of application are shown schematically in Fig. 3.24. The application of LEFM gives conservative results for reactor vessel design.

Suitable properties for pressure vessel steel, i.e. a compromise between the demand for high toughness and high yield stress, are achieved by small additions of alloying material, such as manganese, nickel and molybdenum. The content of certain materials, e.g. phosphorus, sulphur and copper, must be kept very low, since their presence increases irradiation embrittlement. Data for a typical low-alloy pressure vessel steel are shown in Table 3.6.

Reactor vessels are manufactured of rolled and moulded plates or forged rings of steel which are welded together. Cracks may be present in the base material and may arise during manufacture, especially during welding. In spite of thorough quality control of the base material and quality control during the manufacturing process, small cracks or crack-like flaws in the finished vessel cannot be avoided. Hydrostatic tests are therefore carried out at higher than operating pressure to assure the absence of critical cracks.

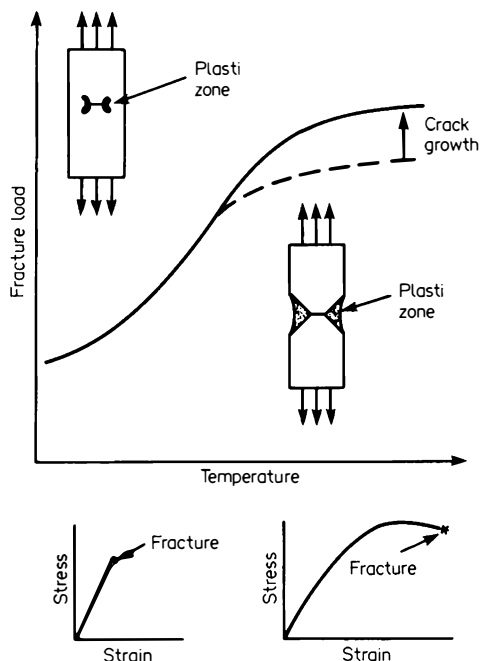


FIG. 3.24. Simplified diagram of various fracture modes in pressure vessel steel. Adapted from *An Assessment of the Integrity of PWR Pressure Vessels*, Second Report by a Study Group under the Chairmanship of Dr W Marshall, U.K. Atomic Energy Authority, March 1982

TABLE 3.6. *Composition and strength properties for pressure vessel steel A 533B*

Composition (percentage by weight)									
C	Si	Mn	P	S	Al	Cr	Cu	Ni	V
0.15–0.25	0.1–0.35	1.2–1.5	<0.01	<0.015	0.01–0.04	<0.02	<0.1	0.5–1.2	0.01–0.02
Strength properties									
Yield stress				($\sigma_{0.2}$)	430–500N/mm ²				
Ultimate strength				(σ_B)	580–650N/mm ²				
Impact toughness				(C_v)	100–180J/cm ²				
Transition temperature				(NDT)	–10––20 °C				

Source: D Smidt, *Reaktorsicherheitstechnik*, Springer Verlag, 1979

The vessel also undergoes extensive ultrasonic testing so that sub-critical defects will also be detected.

The reactor vessel is exposed to varying loads during normal start-up and shutdown, operating disturbances and transient events. At these conditions, pre-existing cracks could conceivably extend to critical dimensions, bearing

in mind also that the fracture toughness decreases through neutron irradiation. All operating conditions are accounted for in the design process, by leaving ample margins for the critical value of the stress intensity factor. Control is further exercised by surveying the changes of the transition temperature with irradiation and by periodic inspection and testing of the vessel during shutdown periods.

3.5.3 *Fatigue and corrosion*

Crack growth during operating conditions can result from three possible mechanisms:

- fatigue*, i.e. cyclic stress variations in an inert environment;
- static stress conditions in a reactive environment, *stress corrosion*;
- cyclic stress variations in a corrosive environment, *corrosion fatigue*.

The pressure vessel steel itself is usually not in contact with the reactive coolant, but is protected by a stainless steel lining. If a crack occurs in the lining, stress corrosion can arise in the vessel. Cracking due to fatigue is considered to be minor in reactor vessels. On the other hand, it can be considerable in the primary system pipelines, due to vibrations induced by the coolant flow.

The pipelines are usually made of austenitic stainless steel. This type of steel is susceptible to stress corrosion under certain conditions. During 1974 the U.S. safety authorities ordered the shutdown of twenty-three boiling water reactors in order to examine them for cracks in the primary system pipes. The mechanism could be identified as intergranular stress corrosion. The corrosion was caused by a combination of an oxidizing environment and a relatively high carbon content in the stainless steel, which resulted in carbide deposits at the grain boundaries. Cracks similar to those observed in the U.S. reactors have also been observed in Swedish BWRs.

Conventional testing methods using ultrasonics have limitations for the detection of cracks in stainless steel. However, it is possible that a crack will result in leakage before fracture in the kind of pipes that occurs in the primary system. The probability of a main coolant pipeline failure is estimated at about 3 in 10,000 operating years.

The tubes in the steam generators of the pressurized water reactor are of particular interest since they are part of the primary system and present in large numbers. They are subject to a series of phenomena which can lead to damage, such as fatigue, corrosion and fretting. Corrosion on the outside of the tubes is usually connected with leakage in the turbine condenser. This can be counteracted by a suitable choice of material and by chemical purification and treatment of the feedwater. During normal operation a

limited number of failed tubes can be accepted without compromising the performance of the primary system.

The failed tubes are plugged to prevent leakage of radioactive water and steam from the primary system. More extensive damage can be a threat, particularly in accident situations. Effective methods for locating, inspecting and repairing failed tubes are in use.

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4

Boiling Water Reactors

This chapter briefly describes the main components and systems of Forsmark-type boiling water reactors. This includes the reactor vessel and internals, primary process systems, reactor containment, turbine generator, control systems and electrical systems. Several of the normal operating systems also have safety-related functions. Clean-up systems and radioactive waste management systems are discussed in Chapter 6. Safety-related auxiliary systems are described in Chapter 8.

4.1 Reactor Vessel and Internals

Figure 4.1 shows a cutaway of a boiling water reactor. The reactor vessel contains the core and the core structure, the control rods with guide tubes, steam separators and steam driers, main recirculation pumps and nozzles for steam and feedwater. The reactor vessel is designed for a pressure of 8.5 MPa and a temperature of 300°C. The pressure and temperature during operation are 7.0 MPa and 275–286°C. The size of the vessel depends upon the power capacity. In a 1000 MWel reactor like Forsmark 3, the inside length of the vessel is 20.8 m and the diameter 6.4 m. The vessel wall thickness is 160 mm. The entire vessel weighs about 760 tons.

4.1.1. Core and core structure

The core of Forsmark 3 consists of 700 vertical fuel assemblies arranged in a quadratic pattern. Each assembly contains 8×8 fuel rods, surrounded by a square fuel box which also serves as a coolant channel (see Figs. 3.3 and 4.10). Between the boxes there are gaps containing cruciform control rods, neutron flux detectors, etc. A group of four assemblies around a control rod forms a fuel module. The fuel modules are located on top of the control rod guide tubes, which support the core.

The fuel assemblies are laterally supported by a core grid which is kept in place by the moderator tank head. The moderator tank also supports the lower part of the fuel assemblies and the upper part of the control rod guide tubes. The wall of the moderator tank separates the core from the downcomer inside the reactor vessel wall. The downcomer is part of the

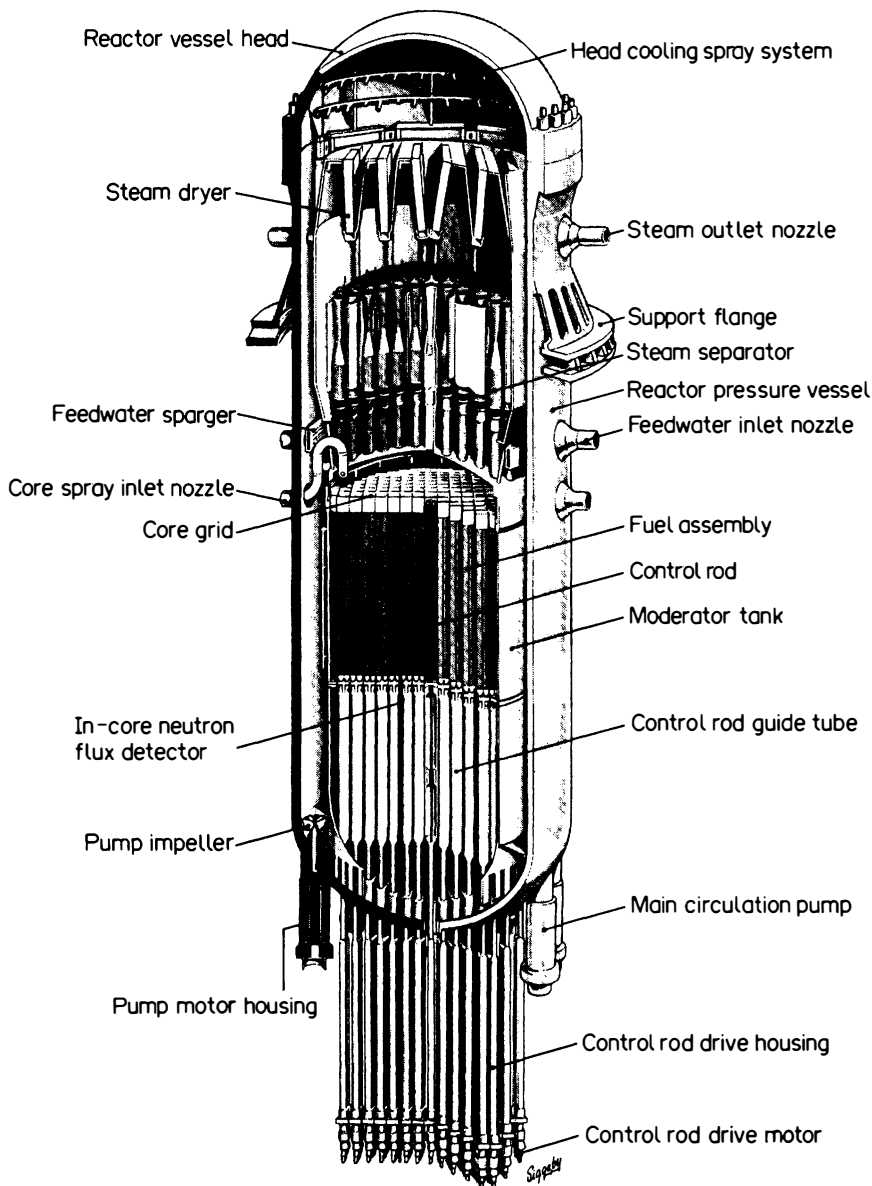


FIG. 4.1. Boiling water reactor vessel and internals. Courtesy AB Asea-Atom

main recirculation system of the reactor. The moderator tank head supports the section above the core, which consists of the steam separators and the steam driers.

4.1.2 Control rods and drive mechanisms

There are 169 control rods in Forsmark 3. Each control rod consists of a cruciform absorber section and a control rod shaft which is connected to the drive mechanism. On the absorber blades there are horizontally drilled channels filled with boron carbide. The blades form a cross which is guided by pads along the sides of the fuel box. The total length of the control rod is about 6.9 m and the weight is about 140 kg.

A drive mechanism consists of an electric motor, a mechanical screw transmission, a piston tube and a guide tube (Fig. 4.2). The lower end of the piston rests on a nut, and the top is connected to the control rod shaft. There are latches located at the lower end of the piston tube which are actuated when the piston tube and the nut come into contact. Each latch fits into a hole in the guide tube. One latch is sufficient to hold the piston tube and the control rod in position.

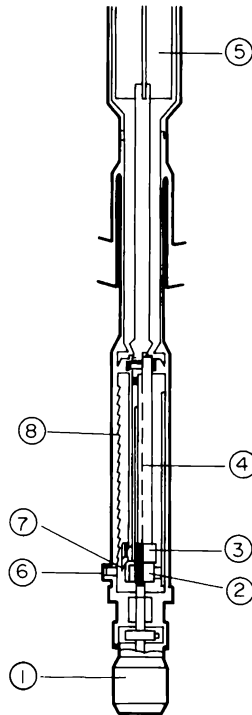


FIG. 4.2. Drive mechanism for an Asea-Atom BWR control rod. The rotation of the motor (1) is transferred via a gear to the screw (2). Depending on the direction of the rotation, the nut (3) is threaded upwards or downwards and carries with it the piston tube (4) and hence the control rod (5). During scram, pressurized water enters through the nozzle (6). The water lifts the piston tube and the control rod. The piston tube leaves the nut, letting down the latches which block its return by catching into holes in the guide tube (8)

During normal operation, the screw is turned by the motor. The nut, the piston tube and the control rod are then pushed up or down depending on the direction in which the screw turns. By counting the number of revolutions made by the screw, the position of the nut and the control rod can be determined.

There is also a hydraulic system for reactor scram whereby high-pressure water passes through an inlet in the drive mechanism, automatically inserting in the piston tube and the control rod. The water is supplied from accumulator tanks, pressurized with nitrogen.

The control rods are divided into scram groups of eight to ten rods each. Each scram group is served by a scram module, comprising a water accumulator tank connected to a high-pressure nitrogen receiver through a scram valve.

The grouping is made so that the reactivity coupling between the rods in a group is negligible. In this way, malfunction of one scram group is equivalent to the loss of only one control rod.

4.1.3. Steam separators and steam driers

At the core outlet, the steam fraction is on average 10–15% by weight. The steam and water must be thoroughly separated for two reasons. Firstly, the steam to the turbine should have as low a moisture content as possible to ensure high efficiency and low risk of erosion of the turbine blades. A low moisture content also minimizes contamination of the turbine with radioactive corrosion products from the reactor. Secondly, the water returned to the downcomer should contain as little steam as possible in order to maintain the required pressure head for coolant recirculation and subcooling at the core inlet.

The steam separation system assembly in Forsmark 3 consists of 165 individual steam separators, which are located on standpipes in the moderator tank head. Each steam separator consists of a riser pipe with vanes at the inlet, giving the steam–water mixture a rotation such that the centrifugal forces separate the steam from the water. The water impinges on the pipe wall and passes through holes and gaps in the wall. The steam concentrates in the middle of the riser and is led upwards through a connecting steam pipe. The outlet steam contains 0–10% of water. The separated water is returned to the downcomer.

The steam drier assembly is made up of several units of corrugated metal sheets. Water from the wet steam settles on the metal sheets and the water is drained to the reactor recirculation system. Normally, the percentage of water in the outlet live steam is at most 0.1% by weight. The walls of the steam drier and the moderator tank head form a cylinder which separates the inlet wet steam from the outlet dry steam. The bottom of the cylinder is

open in order to allow the return of the separated water to the downcomer.

4.2 Primary Process Systems

The purpose of the reactor primary process system is to cool the reactor core and to supply steam to the turbine and feedwater to the reactor. The systems used during normal operation are described in this section. These are:

- the main recirculation system,
- the main steam lines,
- the feedwater system.

Emergency core cooling systems are described in Chapter 8.

4.2.1 Main recirculation system

The main recirculation system cools the reactor core. The inlet feedwater is mixed in the upper part of the downcomer with water returning from the steam separators. The main recirculation pumps take suction from the bottom of the downcomer and force the water through the coolant channels in the core (Fig. 4.3). The pumps are driven by “wet” electrical motors situated vertically under the vessel. The pump motor housing is welded onto the reactor vessel forming an integral part of the reactor vessel (Fig. 4.4). In Forsmark 3 there are eight *internal* main recirculation pumps.

Internal recirculation pumps eliminate the need for major pipe connections in the lower part of the reactor vessel. The risk for loss of coolant due to a break in a recirculation line is thereby also avoided. Older Swedish boiling water reactors have three main recirculation loops with *external* pumps. Another type of recirculation system is used in General Electric boiling water reactors, where about one-third of the coolant flow passes through external recirculation loops. The external pumps supply the driving flow for jet pumps, located in the downcomer (see Fig. 4.5). The jet pumps provide the driving pressure for coolant recirculation through the core. Typically, there are twenty to twenty-four jet pumps depending on the size of the reactor.

4.2.2 Main steam lines

The main steam lines carry steam from the reactor to the turbine. They comprise four 600 mm diameter pipelines. In Forsmark 3, about 1620 kg/s of steam is supplied from the reactor vessel to the turbine at full power.

Each steam line has an internal and an external *isolation valve* close to the reactor containment wall. The internal isolation valve will rapidly close

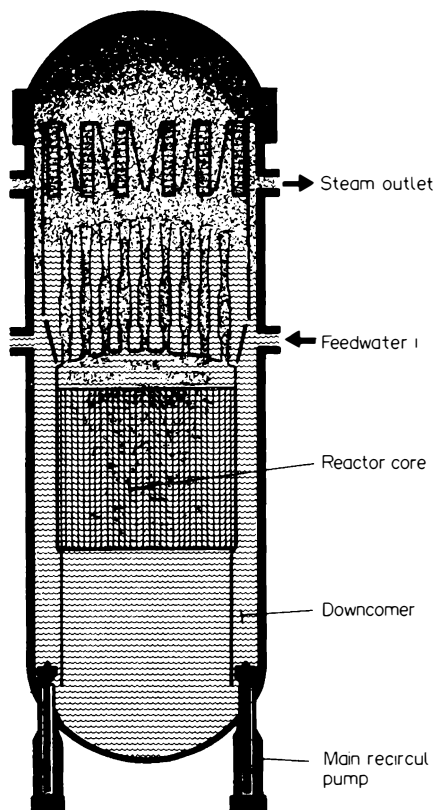


FIG. 4.3. Main recirculation system for a Forsmark-type boiling water reactor.
Courtesy AB Asea-Atom

and interrupt the outlet steam flow in the event of a pipe break outside the containment. The function of the external isolation valve is to isolate the reactor in the event of a pipe break inside the reactor containment, when the main concern is the leaktightness of the containment. The closure time of the isolation valves is 0.5–2.0 seconds.

There are connections for safety and pressure relief valves in every steam line. In Forsmark 3 there are eight safety valves and eight pressure relief valves. Each valve opens on high pressure via a spring-loaded pilot valve. The pressure relief valves also open in response to electric signals. Steam from the safety and pressure relief valves is channelled through a pipe system which discharges below the surface of the water in the condensation pool of the reactor containment.

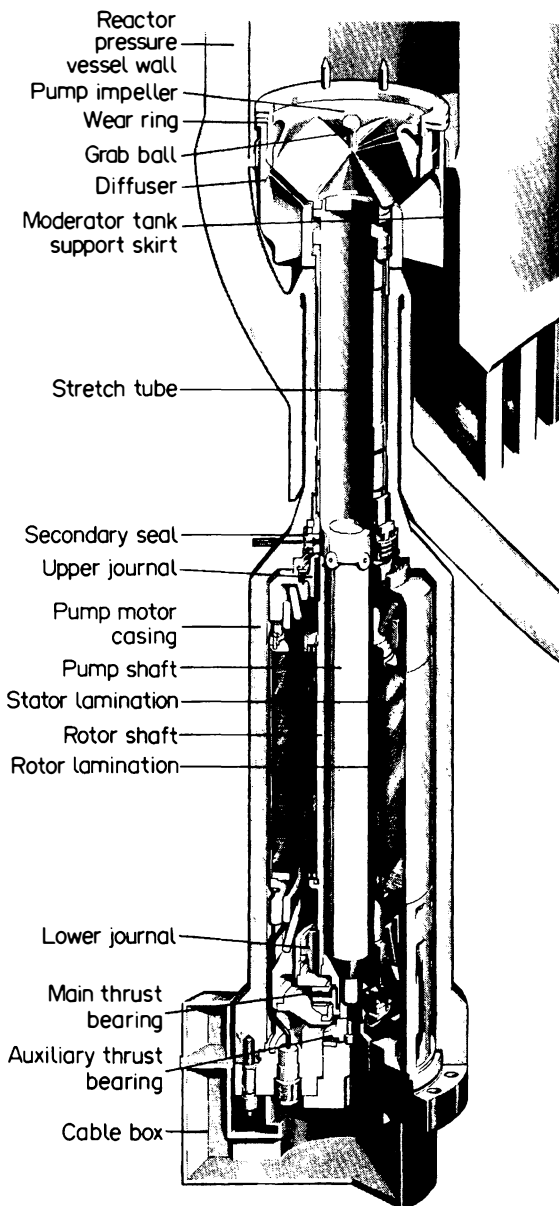
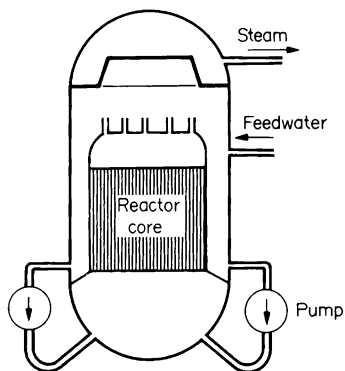
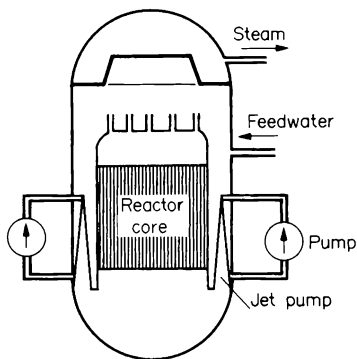


FIG. 4.4. Internal main recirculation pump for Forsmark 3. Courtesy AB Asea-Atom



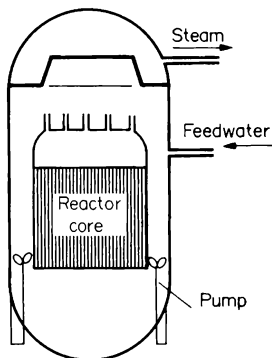
(a)

Older design

Recirculation of entire flow
by external pumps.

(b)

Current U.S. design

Part of flow recirculated by
external pumps. This flow drives the rest of
the core flow by means of internal
jet pumps.

New design

Recirculation of
entire flow by
internal pumps.

FIG. 4.5. Systems for coolant recirculation in boiling water reactors

4.2.3 Feedwater system

The feedwater system carries water from the turbine condenser to the reactor vessel. The system consists of two pipelines, each penetrating the containment wall and equipped with internal and external isolation valves. In Forsmark 3, about 1620 kg/s of water with a temperature of about 215°C is supplied. The water flow is automatically controlled so that the water level in the reactor vessel is held constant.

The internal isolation valves are check valves. The external isolation valves are motor-driven and controlled from the central control room of the reactor. They also close automatically in response to certain safety-related signals.

4.3 Reactor Containment

The reactor containment is a leaktight building surrounding the reactor and the central part of the primary process system. It fulfils several important functions during normal operation and during accident conditions. It acts as a biological shield around the reactor and prevents the release of radioactive substances in the event of leakage in the reactor's primary system. The containment also protects the reactor from the effects of external events.

4.3.1 Pressure suppression principle

The boiling water reactor containment is designed in accordance with the pressure suppression (PS) principle. The lower part of the building houses a water reservoir for the condensation and cooling of steam escaping from the primary system. This steam flow occurs, for example, at high pressure in the reactor when the pressure relief valves open, or when there is a pipe break in the primary system. Because the water reservoir acts as a heat sink, the pressure increase in the containment is limited allowing a smaller containment volume.

Figure 4.6 illustrates the principle of the pressure suppression containment. The containment has two main parts: a primary containment or drywell, and a secondary containment or wetwell. The drywell encloses the reactor and primary system piping. The wetwell contains a condensation pool and a compression chamber. The drywell and wetwell are connected through the blowdown pipes which discharge below the surface of the condensation pool.

During operation, the containment is filled with nitrogen at atmospheric pressure in order to eliminate the risk of hydrogen explosion in accident situations. This is called *inerted* containment.

In the event of a pipe break in the primary system, the overpressure in the drywell is relieved by the steam flow through the blowdown pipes and the steam condensation in the condensation pool. Under certain conditions, the pressure in the wetwell can increase due to the flow of non-condensable gases from the drywell to the wetwell which then collect in the compression chamber above the surface of the condensation pool. There is a vacuum breaker between the compression chamber and the lower drywell for returning the gas to the drywell.

The condensation pool is cooled by a spray system via an intermediate loop to the ultimate heat sink, the sea. After a pipe break, water can also be sprayed in the drywell, thereby contributing to cooling as well as to the removal of airborne radioactive substances from the containment atmosphere.

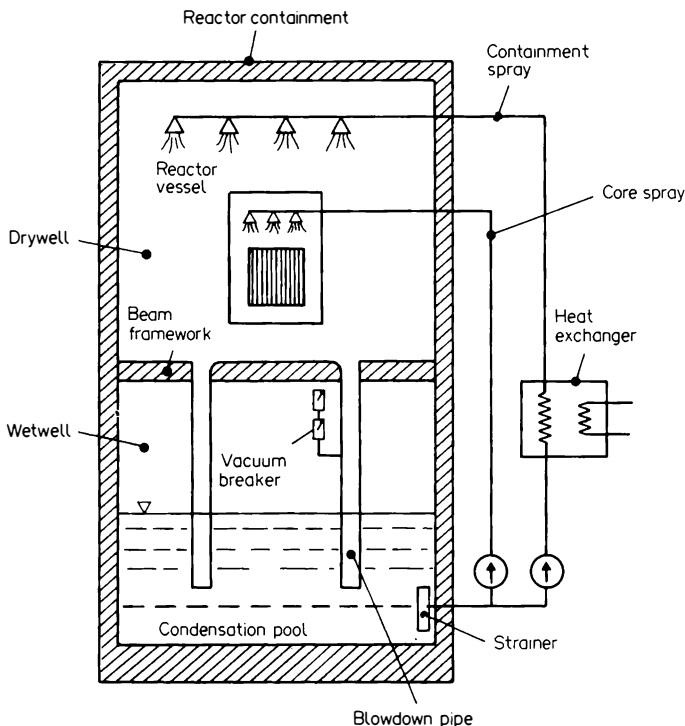


FIG. 4.6. Schematic of a pressure suppression containment

4.3.2 Containment design

The detailed design of the reactor containment varies for the different generations of boiling water reactors. Figure 4.7 shows the reactor containment for Forsmark 3. The containment vessel has a flat base, a circular-cylindrical shell and a roof with a slightly conical underside. The roof forms the base of the reactor pools. The load-bearing walls are of prestressed concrete. Leaktightness is achieved by a 5 mm thick steel liner which is embedded in the concrete at least 200 mm from the inside of the load-bearing parts. The central part of the roof has a removable head in the form of a steel cupola.

The inner framework of the containment is joined at the base to the bottom slab of the containment vessel, but is otherwise separate from the vessel. The central part of the frame is a 1.2 m thick cylindrical concrete wall which serves as a *biological shield* around the reactor. The reactor vessel rests on the upper part of this concrete wall. The framework of concrete beams which separates the upper drywell from the wetwell below is also included in the central part of the containment.

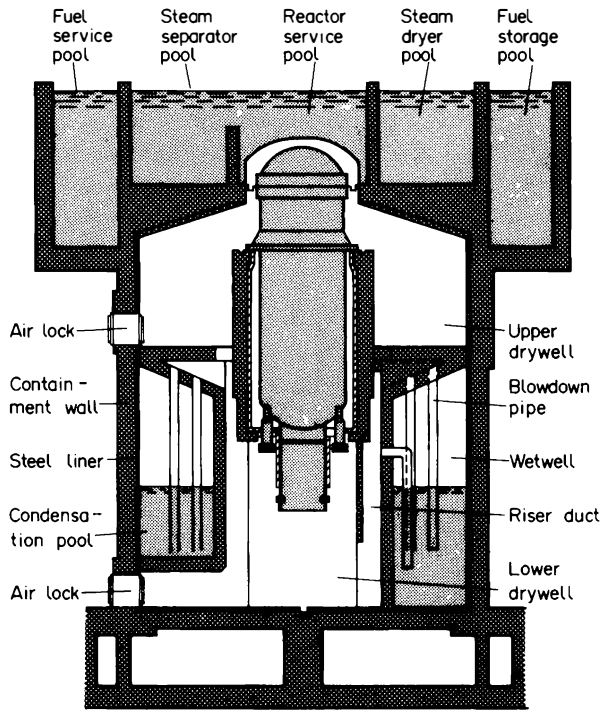


FIG. 4.7 Forsmark 3 reactor containment. Courtesy AB Asca-Atom

There are airlocks in the wall of the containment for access to the building after shutdown. There are also penetrations in the form of embedded pipes welded to the liner of the containment vessel.

4.4 Turbine-generator Plant

In the turbine, the thermal energy of the steam is converted into mechanical energy which is then converted to electrical energy in the generator. After the turbine, the steam is led to the condenser where it condenses into water. The condensate is purified, preheated and pumped back to the reactor vessel through the feedwater system. Figure 4.8 is a schematic diagram of the turbine-generator plant.

4.4.1 Turbine-generator

The steam turbine consists of a high-pressure part and a low-pressure part. The live steam first enters the high-pressure turbine where it yields about 40% of its useful energy. It then enters the moisture separator

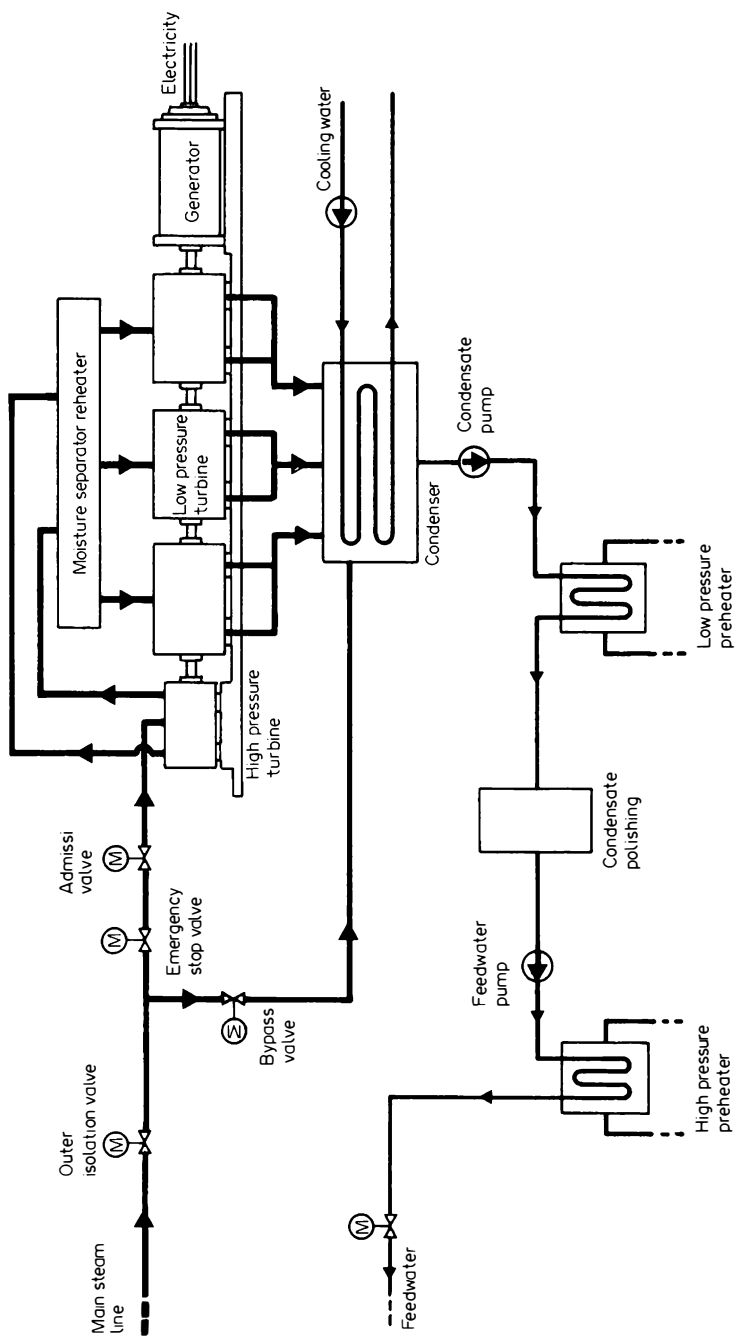


FIG. 4.8. Schematic of the turbogenerator plant

reheater where it is dried and heated. The steam then expands in the low-pressure part which has three turbines. The turbines and the generator are located on the same shaft. All turbines are of the axial, double-flow type where the steam enters in the middle and is exhausted at the ends. The reason for having several low-pressure turbines is that the flow must be directed through several exhausts since the length of the turbine blades cannot exceed 1 m for 3000 rpm (revolutions per minute) and about 1.3 m for 1500 rpm as in Forsmark 3.

The length of the turbine is considerable due to the number of exhausts accommodated on the single shaft. Using two shafts can therefore be advantageous when the turbine power is high. This also provides for operation at reduced power in the event of a single turbine or generator failure. This type of redundancy is realized at Ringhals 1, Forsmark 1 and 2.

4.4.2 Steam system

The turbine plant steam system carries steam from the reactor via four main steam lines to the turbine-generator. The steam flow to the turbine and therefore the reactor pressure is regulated by throttle valves. Intercept and emergency stop valves protect against overspeed. The whole of the live steam flow can be dumped to the main condenser during start-up or upon load rejection.

Live steam is used for reheating the steam after the high-pressure turbine. Bled steam is tapped off from various places in the turbine and used for heating the condensate and feedwater in the preheaters.

Because the steam from the reactor is radioactive, there are stringent requirements for leaktightness in order to prevent water and steam leakage.

4.4.3 Condensate and feedwater system

Exhaust steam is led to the main condenser which is situated directly under the low-pressure turbines. The condenser is, in principle, a large tube-and-shell heat exchanger. Forsmark 3 has three single-flow condensers, i.e. one per low-pressure turbine, each with two inlet water boxes (Fig. 4.9). Cooling water is supplied from six cooling water pumps, one per water box. The condenser is cooled directly with sea water. Condenser vacuum is maintained by two ejector systems, each with full capacity.

The condensate is pressurized, preheated and delivered to a storage tank, from which the feedwater pumps draw their suction. Both the condensate and the feedwater pump sets consist of three half-capacity electrically driven units. The feedwater pumps are speed-controlled by means of hydraulic couplings. There are six feedwater heating stages, consisting of three low-pressure feedheaters, a feedwater tank, and two high-pressure feedheaters.

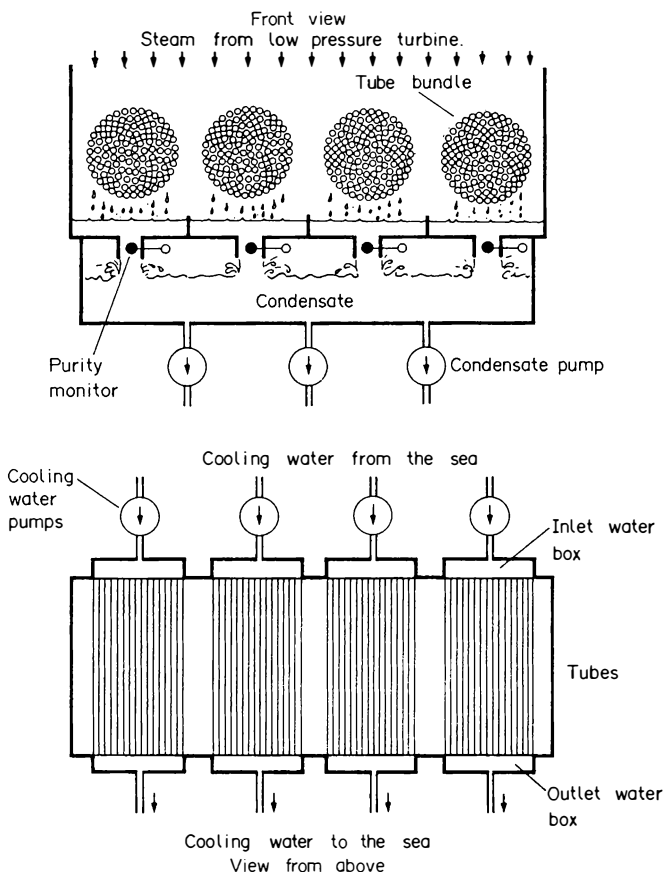


FIG. 4.9. Schematic of a turbine condenser

4.5 Control and Monitoring Systems

The primary process systems are monitored and controlled from the reactor's central control room. Important process variables are presented on control desks and panels. Alarm signals and annunciators attract the reactor operator's attention in the event of malfunction or if bounding values for process variables are exceeded. The reactor has a special computer which registers, processes and presents data for core monitoring. The computer calculates and proposes adjustments of control rod positions. The control rod positions, neutron flux, etc., are stored by the computer and can be displayed on colour TV screens in the control room.

4.5.1 Measuring systems

The neutron flux in the reactor core is monitored by a large number of measuring channels with neutron detectors inside the core. The measuring range of the system is from 10^{-10} to 1.25 of nominal power. In order to cover the entire range from start-up to full power, three overlapping systems are used:

- Source Range Monitoring (SRM) measures the neutron flux from subcritical reactor and criticality up to a neutron flux corresponding to a relative power of about 10^{-6} . Measuring is then taken over by
- Intermediate Range Monitoring (IRM), which covers the range up to a relative power of about 20%, after which
- Power Range Monitoring (PRM) continues the measuring. The PRM system consists of two subsystems—LPRM, which monitors the local power at more than 100 measuring points, and APRM, which provides information on the instantaneous total power.

Start-up neutron sources are used in the subcritical reactor to provide a neutron flux that can be measured in the range covered by the SRM system. The sources are inserted in the core from below. Figure 4.10 shows a section

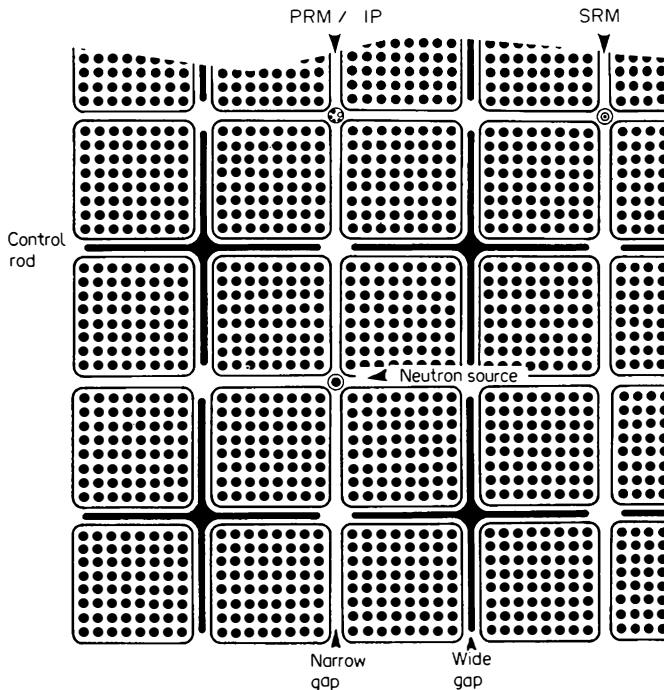


FIG. 4.10 Section of the core region in a boiling water reactor

of the core with the location of a start-up neutron source, SRM and PRM detectors. During operation, the start-up neutron sources and the SRM and IRM detectors are withdrawn from the core. The diagram also shows a TIP (Travelling In-core Probe) detector, which is used for detailed mapping of the axial power distribution and for the calibration of the LPRM detectors.

Special instrumentation provides information on:

- water level in the reactor vessel,
- reactor vessel pressure,
- core coolant mass flow,
- core coolant temperature,
- pressure head of the main recirculation pumps,
- reactor vessel temperature.

4.5.2 Control rod manoeuvring

The main power control of the reactor is effected from the control room by regulating the speed of the main recirculation pumps and by manoeuvring the control rods. During shutdown, all the control rods are inserted into the core. Start-up is achieved by withdrawing banks of control rods.

The final operating condition is adjusted by regulating the pump speed and by manoeuvring individual control rods. At normal full power operation, the majority of the control rods are withdrawn from the core. Only about 10% or less of the total number is completely or partially in the core. Most of the reactivity reduction caused by fuel burn-up is compensated for by burnable absorbers (see 3.3.7). The position of the control rods is adjusted about once a week in order to compensate for reactivity changes due to the depletion of the fuel and the burnable absorbers.

The pattern of fully or partially inserted control rods is important for maintaining a favourable power distribution in the core. In the boiling water reactor, the axial power distribution has a tendency to peak in the lower part of the core, since steam production reduces reactivity in the upper part. Therefore, it is important that the control rods are inserted in the core from below in order to obtain a more uniform power distribution.

The reactor is shut down by inserting the control rods fully into the core. This is achieved in two ways: by actuating the electromechanical transmission of the drive mechanisms or by hydraulic insertion, known as *scram*. It takes about 4 minutes to screw the control rods into the core from a fully withdrawn position. During scram, the control rods are fully inserted within 4–6 seconds.

4.5.3 Water level and pressure control

The water level in the reactor vessel is controlled by a liquid level regulator which compares the measured level with the required level, and a flow regulator which compares the steam flow with the feedwater flow. The level control system varies the speed of the feedwater pumps and the setting of the control valves in the feedwater lines.

The reactor pressure control system monitors the pressure and the neutron flux and affects the position of the turbine throttle valve. Since steam generation is proportional to neutron flux, the neutron flux provides advance information on the reactor pressure, so that a well-damped pressure control is achieved.

4.5.4 Power control

For fixed control rod positions, the reactor power can be varied within certain limits by means of the coolant flow. If the coolant flow decreases, the void content of the core increases, thereby reducing reactivity and power. Similarly, an increase in power is obtained by increasing the coolant flow. Variation of the coolant flow is achieved by varying the speed of the main recirculation pumps.

The power control system has three operating modes which can be selected by the reactor operator:

- Pump speed regulation, when all main recirculation pumps are controlled by the operator.
- Power control, when the electrical power generated by the unit is maintained at a preset value.
- Power and frequency control, when the electric power generated is automatically adjusted to the frequency of the grid.

Power control is the normal operating mode. It is used at both full and partial load for daily and weekend load-following as well as for base load operation. The desired power level can be set manually from the control room or automatically, and by remote control. The power is controlled with a time constant of 10–15 seconds.

If the plant is to contribute to grid frequency control, a time constant of 5 seconds or less is required. In order to achieve this, the reactor pressure is allowed to vary within about 0.3 MPa by coordinating the pressure and power control systems.

A schematic diagram of the reactor control systems is shown in Fig. 4.11. The power level is established by coordinating the pressure and power control system. The turbine plant acts as a slave to the reactor. The steam flow to the turbine is normally regulated in order to keep the reactor pressure

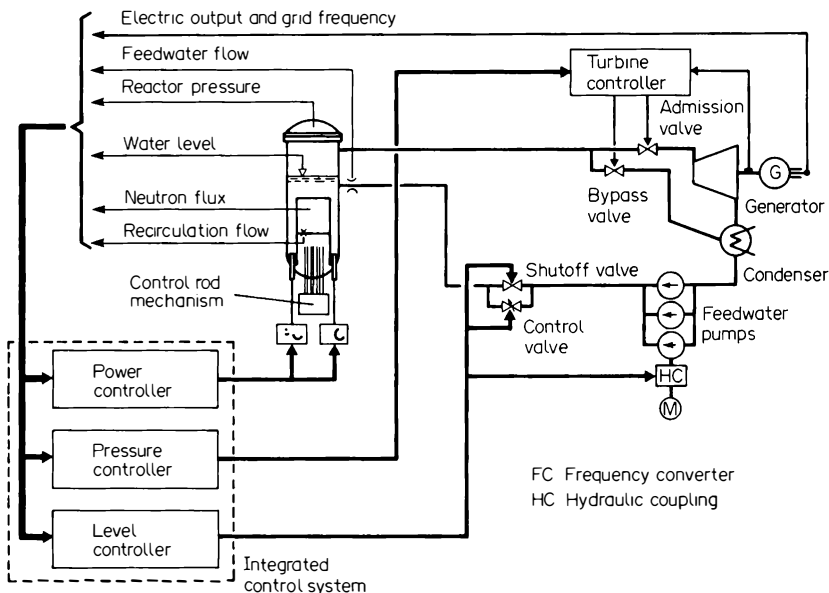


FIG. 4.11. Boiling water reactor control systems. Courtesy AB Asea-Atom

constant. The feedwater flow is controlled to follow the power so as to maintain a constant water level in the reactor vessel.

Continuous operation is possible from 100% to 25% of nominal power (Fig. 4.12). In the region of 100% to about 65%, the power is controlled by varying the speed of the main recirculation pumps. The position of the control rods is only changed in order to adjust the pump speed at constant power. At power levels lower than 65%, a power change is normally achieved by changing the position of the control rods at constant, low recirculation flow.

At reactor start-up, the power is increased by fine-motion control rod operation at a rate of 1–2% of nominal power per minute. Power increase in the range above 65% of nominal power is normally achieved by first maintaining the power level constant while withdrawing the control rods and simultaneously reducing the pump speed. The power can then be rapidly increased by using the recirculation pumps. At power levels below 65%, power changes are normally achieved by changing the position of the control rods at a constant, low recirculation flow.

4.6 Electrical Systems

The electrical systems in a reactor plant can be divided into offsite power supply systems and onsite power supply systems.

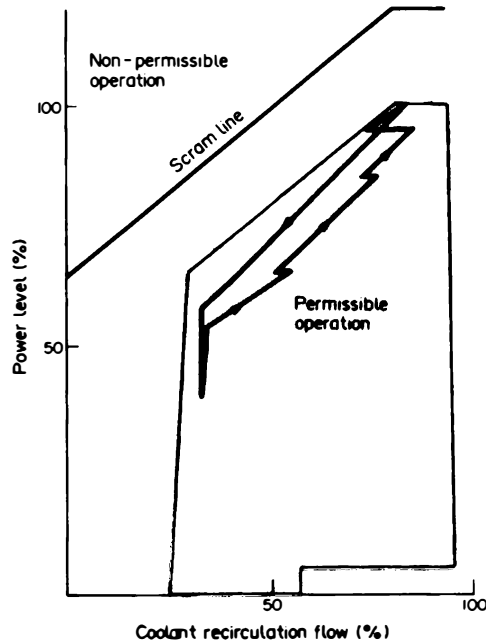


FIG. 4.12. Operating range for a boiling water reactor. From *BWR 75 Operational Flexibility*, AB Asea-Atom, 1978

4.6.1 Offsite power supply

The electrical generator of a reactor plant is connected to the main grid via the main transformer and the switchyard. During start-up and shutdown, the generator is connected to or disconnected from the main transformer with the generator breaker. The generator breaker is also used to isolate the generator if a failure should occur in the turbine generator or in the electrical system.

There is also a second offsite power supply from the so-called start-up grid which is connected to the plant's auxiliary power supply system via a separate transformer. The start-up grid is automatically re-energized by gas-turbine generators if it goes down.

4.6.2 Onsite power supply

Auxiliary power is needed in a reactor plant: AC power for the operation of pump motors, etc., and DC power for the control and measuring systems. The auxiliary power demand is about 3% of the gross electrical power generated by the plant. During normal operation, the auxiliary power sup-

ply system is connected to the plant's main generator busbars via the plant transformers. In the event of a failure of the offsite power supply, the plant changes over to house load operation, i.e. the main generator supplies electricity only to the auxiliary system. Excess steam is dumped directly into the condenser as the reactor power level is adjusted to the reduced load.

Each reactor unit has its own auxiliary power supply system in Swedish power stations. The auxiliary power supply consists of general systems, and of diesel-backed and battery-backed emergency systems for safety-related equipment. In modern plants, the auxiliary power supply is subdivided into four buses, as shown in the circuit diagram (Fig. 4.13).

A reliable auxiliary power supply is very important for reactor safety. If the turbine plant is shut off, auxiliary power is supplied from the main external grid via the main transformer and the plant transformers. If this supply is not available, auxiliary power is obtained from the separate gas-turbine-backed start-up grid via the start-up transformer. In the event of a total failure of auxiliary power, emergency power to safety-related equip-

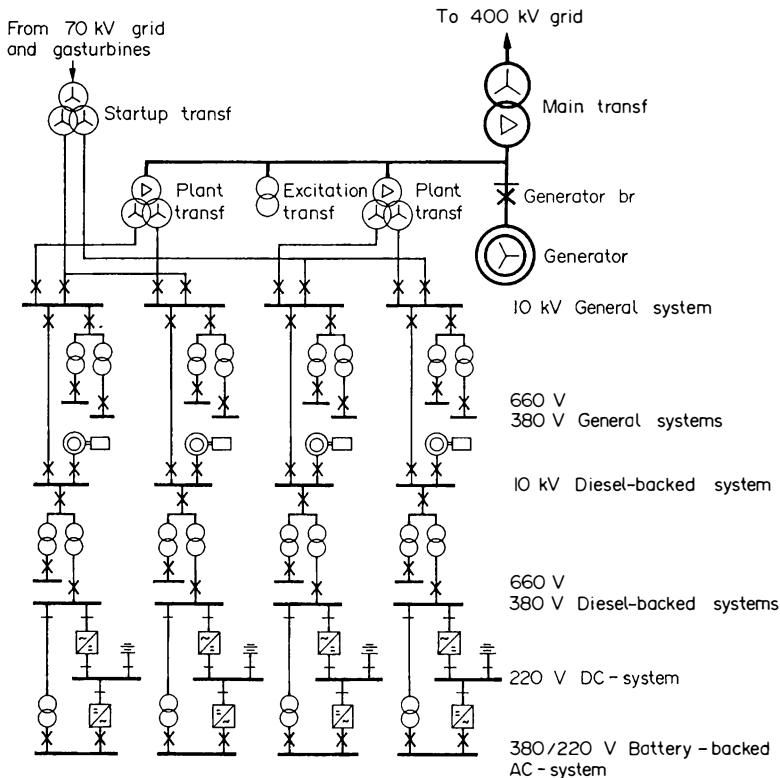


FIG. 4.13. Circuit diagram of the Forsmark 3 power supply systems. Courtesy AB Asea-Atom

ment is supplied from the diesel-driven emergency system and from the battery system. The battery system is charged by transformers in the diesel-driven system (see Fig. 4.13).

4.7 Main Technical Data for Swedish Boiling Water Reactors

The description of the boiling water reactor is summarized in Table 4.1 which provides main technical data for the following typical Swedish boiling water reactors:

<i>Reactor</i>	<i>Commissioned (year)</i>	<i>Capacity (Mwel)</i>
Oskarshamn I	1972	440
Oskarshamn II	1974	595
Forsmark 1	1980	970
Forsmark 3	1985	1063

Oskarshamn I and II are first generation reactors with external main recirculation loops, while Forsmark 1 and 3 have internal recirculation pumps. Otherwise, the basic design of the reactors has largely remained unchanged. The thermohydraulic design is characterized by a successive increase in the

TABLE 4.1 *Main technical data for Swedish boiling water reactors*

Parameter	Unit	OI	OII	F1	F3
REACTOR VESSEL					
Design pressure	MPa	8.5	8.5	8.5	8.5
Design temperature	°C	300	300	300	300
Total weight	kg	405,000	655,000	740,000	760,000
Inner height	m	17.6	20.0	21.2	20.8
Inner diameter	m	5.0	5.2	6.4	6.4
Wall thickness, carbon steel	mm	125	126	154	150
THERMOHYDRAULICS					
Thermal power	MWth	1375	1700	2700	3000
Steam flow rate	kg/s	640	850	1345	1620
Coolant flow rate	kg/s	6900	6070	10,400	11,400
Operating pressure	MPa	7.0	7.0	7.0	7.0
Feedwater temperature	°C	160	180	180	215
Subcooling at core inlet	°C	11	14	13	8.3
Steam quality at core outlet	wt%	9.8	14.4	13.2	14.5
Fuel power density	kW/kgU	17.3	21.5	22.1	23.7
Core heat transfer surface	m ²	3997	4028	6080	6296
Fuel rod surface heat flux, average	MW/m ²	0.33	0.41	0.43	0.46
Fuel rod surface heat flux, max	MW/m ²	0.95	1.05	1.08	1.08
Fuel rod linear heat rate, average	kW/m	13.1	15.5	16.3	17.5
Fuel rod linear heat rate, max	kW/m	36.6	40.4	41.4	41.5
Total power peaking factor		2.70	2.52	2.52	2.35
Minimum critical power ratio	—	1.7	1.4	1.4	1.3

Parameter	Unit	OI	OII	F1	F3
REACTOR CORE					
<i>Fuel</i>					
Fuel weight, total	kg U	74,900	81,000	122,300	126,300
Fuel density	kg UO ₂ /m ³	10,400	10,500	10,500	10,500
Max UO ₂ temperature	°C	1600	1700	1800	1800
Number of fuel assemblies	—	448	444	676	700
Weight of assembly incl box	kg	290	307	306	315
Weight of fuel per assembly	kg U	177.2	182.5	180.9	180.9
Number of fuel rods per assembly	—	63	63	63	63
Rod length	mm	3650	3712	3680	3680
Rod outer diameter	mm	12.25/11.75	12.25/11.75	12.25/11.75	12.25/11.75
Cladding thickness	mm	0.80	0.74	0.80	0.80
Number of pellets per rod	—	243	247	245	245
Pellet diameter (cold)	mm	10.47/9.97	10.58/10.08	10.46/9.96	10.46/9.96
Pellet length (cold)	mm	15	15	15	15
<i>Control rods</i>					
Control rods, number	—	112	109	161	169
stroke	mm	3650	3650	3650	3650
span	mm	272	272	272	272
length	mm	6383	6383	6383	6872
Absorber section, length	mm	3646	3646	3646	3646
Total weight of a control rod	kg	140	134	134	140
<i>In-core neutron detectors</i>					
Number of fixed detectors	—	92	96	144	148
Number of fixed detector probes	—	23	24	36	37
Total number of detector probes	—	27	32	48	49
PRIMARY PROCESS SYSTEMS					
<i>Main recirculation system</i>					
Number of recirculation loops	—	4	4	—	—
Number of internal pumps	—	—	—	8	8
Mass flow rate per pump	kg/s	1725	1500	1300	1425
Pump shaft power	kW	700	500	620	650
<i>Main steam lines</i>					
Number of main steam lines	—	2	4	4	4
Design pressure	MPa	8.5	8.5	8.5	8.5
Design temperature	°C	300	300	300	300
Pipe diameter	mm	650	500	600	600
<i>Pressure relief system</i>					
Number of pressure relief and control valves	—	16	22	13	18
Total capacity	kg/s	1000	1250	1070	1870
REACTOR CONTAINMENT					
<i>Design pressure</i>					
Absolute pressure	MPa	0.45	0.50	0.55	0.60
Underpressure	MPa	0.05	0.05	0.05	0.05
Drywell, free volume	m ³	3460	5115	4320	5562
Compression chamber, free volume	m ³	1860	2960	3560	2775
Condensation pool, water volume	m ³	1950	1940	3050	3000
Design temp, drywell	°C	150	170	180	172
Design temp, wetwell	°C	110	157	150	150
Max leak rate, free volume	%/d	1	1	1	1

Parameter	Unit	OI	OII	F1	F3
TURBINE-GENERATOR					
Rated power	MW	460	600	2 × 470	1040
Generator speed	rpm	3000	3000	3000	1500
Admission pressure	MPa	6.80	6.78	6.70	6.70
Dump capacity	%	100	100	100	100
Main condenser, cooling capacity	MW	875	1090	2 × 870	2000
<i>Condensate and feedwater system</i>					
Feedwater flow rate	kg/s	640	850	1345	1620
Number of condensate pumps	—	3	3	2 × 3	3
Number of feedwater pumps		3	3	2 × 3	3
ELECTRICAL SYSTEM					
Number of plant transformers		1	1	2	2
Number of start-up transformers		1	1	1	1
Number of diesel generators		2	2	4	4
Number of diesel-backed busbars		3	2	4	4

Source: *Oskarshamn Nuclear Power Plant Unit 3. Preliminary Safety Analysis Report*, AB Asea-Atom and OKG AB, 1975

mean power density of the fuel, while essentially retaining the maximum surface heat flux. This has been achieved by attaining a more uniform power distribution in the core, i.e. a lower total form factor. Consequently, it was not necessary to increase the volume of the core and the reactor vessel in proportion to the increase in total power.

The basic design of the fuel assemblies and the control rods has remained unchanged. The reduction of the total form factor was partly achieved by burnable absorbers in the form of gadolinia (Gd_2O_3) in the fuel rods. Due to the burnable absorbers it has been possible to reduce the excess reactivity for a given burn-up at the beginning of an operating cycle. This can be used to increase the energy output of the fuel (the average burn-up) without raising the requirements on reactivity compensation with the control rods, i.e. without increasing the relative number of control rods.

Important modifications have been made in safety-related auxiliary systems. These developments are discussed in Chapter 8.

References

- 401 Swedish Department of Industry, *Safety Study Forsmark 3*, DsI 1978:3 (In Swedish)
- 402 *Final Safety Analysis Report Forsmark Unit 3*, AB Asea-Atom and State Power Board, June 1983

5

Pressurized Water Reactors

The pressurized water reactor is the most common type of reactor in today's nuclear power plants. Although the basic design remains unchanged, there are variations in the detailed design by different reactor manufacturers. This chapter describes the main characteristics of Westinghouse-type reactors, which are represented in Sweden by Ringhals 2, 3 and 4. The description refers to those reactor components and systems which are essential for normal operation. Safety-related auxiliary systems are treated in Chapter 8. The presentation is structured in the same manner as in the previous chapter to facilitate a comparison of the two reactor types. The turbine-generator plant and the power supply systems are the same regardless of reactor type. Consequently, sections 4.4 and 4.6 also apply to pressurized water reactors.

5.1 Reactor Vessel and Internals

The pressurized water reactor has a more compact core and a higher system pressure than the boiling water reactor. There is no equipment for steam separation in the reactor vessel since the water does not boil in the core. The pressurized water reactor vessel is therefore not as high and has a smaller diameter and thicker walls than the boiling water reactor vessel. Typical values (Ringhals 3) are: total height 13.0 m, internal diameter 3.99 m, wall thickness 20 mm, and weight 330 tonnes.

Figure 5.1 shows a section through a typical reactor vessel with internals (Sizewell 3). The vessel contains the reactor core and core structure, control rods with guide tubes, and instrumentation. There are nozzles for cooling water pipes, control rods and core instrumentation. The vessel has a removable upper head which is retained by a gasketed bolted flange.

5.1.1 Core and core structure

The reactor core is located below the coolant nozzles. The core of Ringhals 3 and 4 each holds 157 fuel assemblies containing 17×17 rod positions. A cross-section of the core is shown in Fig. 5.2. There are no fuel channels in the core so that radial mixing of the coolant flow is possible. The core is

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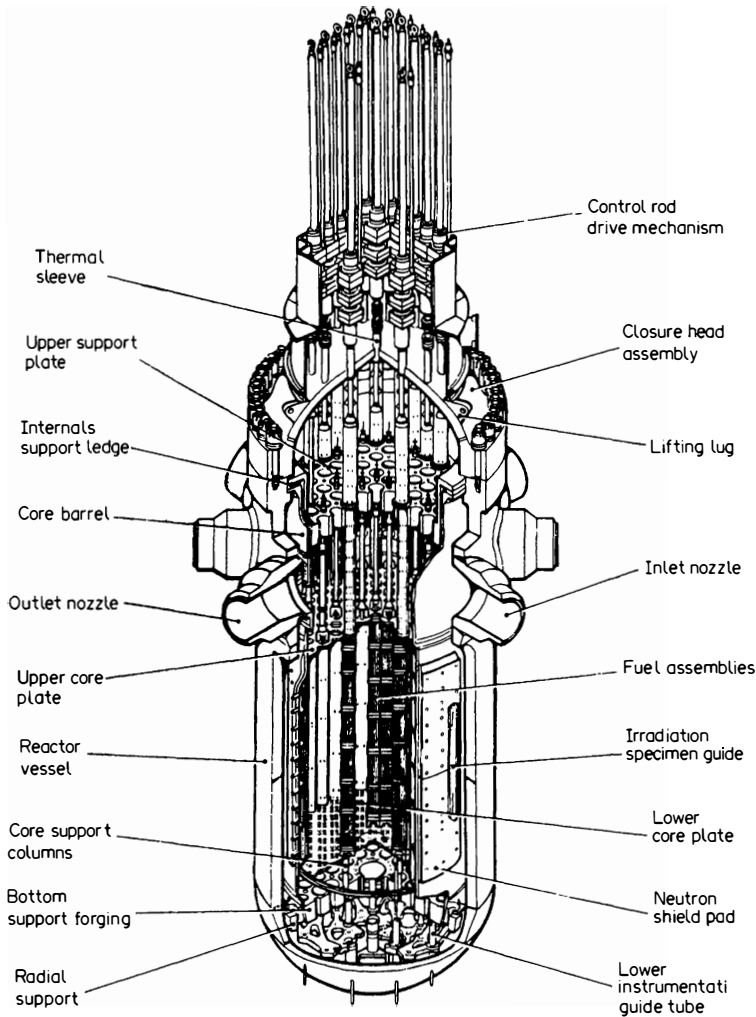


FIG. 5.1. Reactor internal structure. From *Advances in Power Construction*, Pergamon Press, 1986

surrounded by a baffle which is attached to the moderator tank (the “core barrel”).

There are guide thimbles instead of fuel rods in twenty-four of the rod positions in each fuel assembly. In about one-third of the fuel assemblies, *cluster control rods* can be inserted in the thimbles from above (Fig. 5.3). The guide thimbles which do not contain control rods are either empty and plugged or occupied by fixed rods made of boron glass acting as a burnable

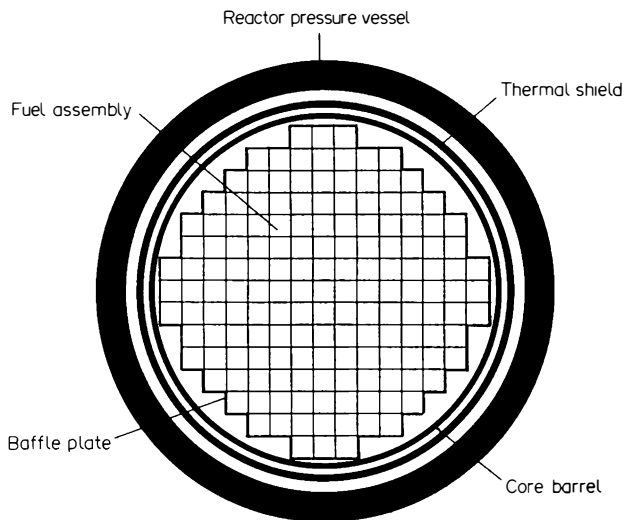


FIG. 5.2. Cross-section of a pressurized water reactor core

absorber. The central position in the fuel assembly can be used to hold detector probes which are inserted from below.

The core structure mainly consists of an upper and a lower support structure, the core barrel and the thermal shield. The upper core support structure acts as a support and anchor for the upper ends of the fuel assemblies, while protecting and guiding the control rods. The lower core support structure carries the core, the core barrel and the thermal shield. The core barrel separates the core from the downcomer space nearest to the vessel wall. The thermal shield which is integral with the core barrel provides shielding from core radiation, thereby reducing irradiation damage and thermal stress in the pressure vessel wall.

5.1.2 Control rods and drive mechanisms

Each control rod consists of an absorber section and a drive shaft which is connected to the drive mechanism above the reactor vessel. The absorber section comprises a rod cluster which is inserted into the fuel assembly guide thimbles (see Fig. 5.3). Each absorber rod is a silver-indium-cadmium alloy contained in stainless steel tubing. In general, there are two types of control rods: those entirely composed of the highly neutron-absorbant alloy, and those only partly composed of neutron absorber. The full-length rods are mainly used for shutting down the reactor. Most of them are withdrawn from the core during normal operation. The part-length rods are used to achieve a stable and axially flattened power distribution in the core. Swedish PWRs only have full-length rods.

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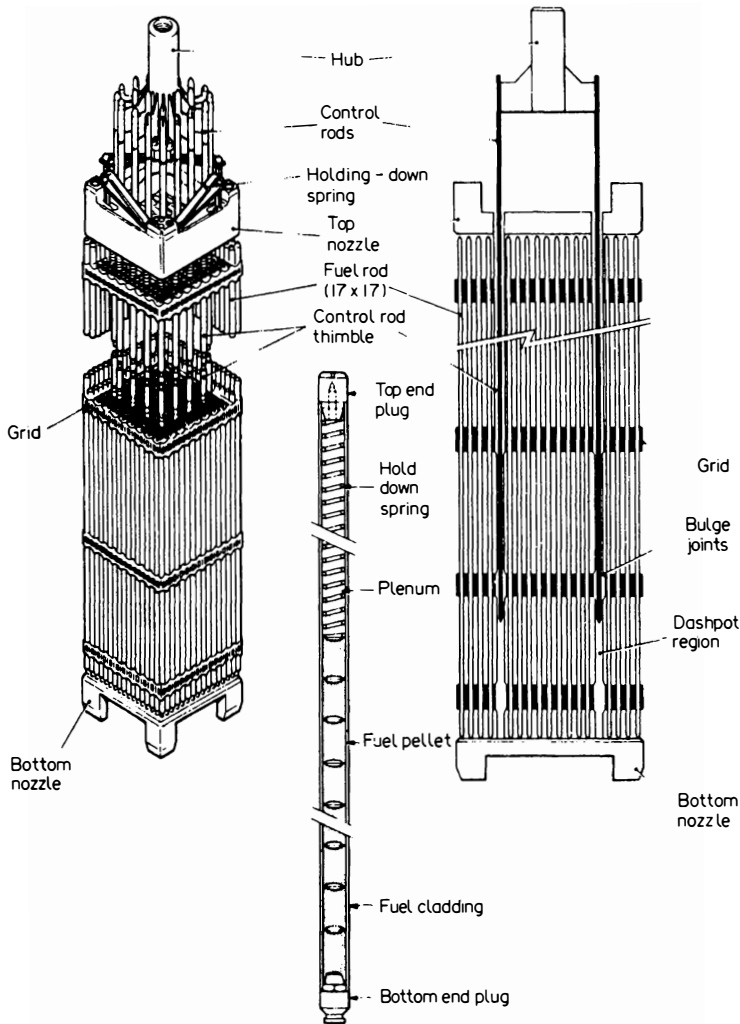


FIG. 5.3. Pressurized water reactor fuel assembly and cluster control rod. From *Advances in Power Construction*, Pergamon Press, 1986

The drive mechanism of the full-length rods uses magnetic coils to operate the working components that move the drive shaft and the attached control rod. Fast total insertion (scram) is obtained by simply removing the electrical power, allowing the control rod assemblies to fall by gravity.

5.1.3 Instrumentation

The core instrumentation consists of thermocouples for measuring the temperature of the water leaving certain fuel assemblies, and movable miniature detectors for measuring the neutron flux in the core. The thermocouples are inserted into the core from above through thimbles, and the neutron detectors are similarly inserted from below. The core instrumentation provides information which can be used to calculate the burn-up and to estimate the distribution of the coolant flow in the core.

A system based on measuring the leakage flux of neutrons from the reactor is used for continuously monitoring the fission rate and thereby the nuclear power. Typically, the system comprises two SRM, two IRM and four PRM measuring channels with detectors placed outside the reactor vessel in the biological shield.

5.2 Reactor Coolant System

The reactor coolant system consists of the reactor vessel and connecting coolant loops. Swedish pressurized water reactors have three parallel coolant loops (Fig. 5.4). Each loop contains a coolant pump and a steam generator with pipelines. The system also includes a pressurizer.

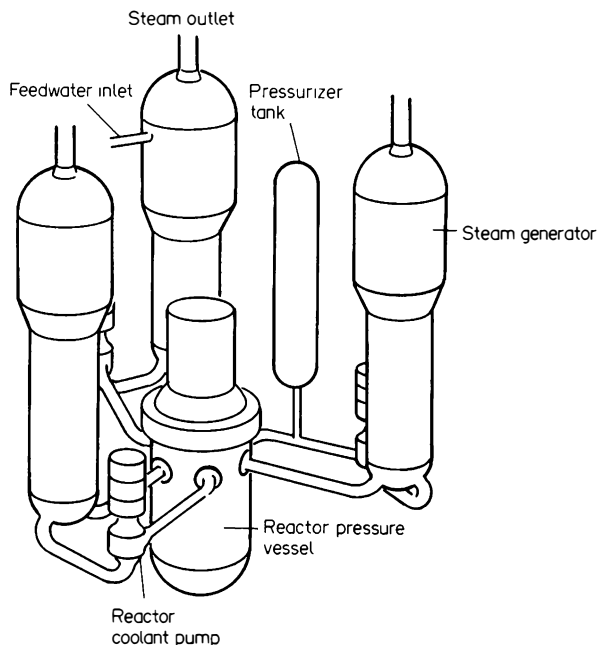


FIG. 5.4. Pressurized water three-loop reactor coolant system

5.2.1 Main coolant system

The coolant enters the reactor vessel through the inlet nozzles and flows downward on both sides of the thermal shield in the downcomer between the core barrel and the reactor vessel. When it reaches the lower plenum formed by the bottom head, the coolant reverses direction and flows upwards through the lower core support structure where it is uniformly distributed over the core inlet. After passing through the core to the upper plenum, the coolant flows out through nozzles in the core barrel and reactor vessel. All inlet and outlet nozzles are located above the upper edge of the core, which makes it easier to keep the core covered with water and cooled in the event of a pipe break in a coolant loop.

The mass flow and temperature of the coolant are regulated in order to maintain the required thermohydraulic performance in the core and the balance between the heat transferred to the coolant in the core and the heat removed from the coolant in the steam generators. The total coolant flow in Ringhals 3 is 12,860 kg/s. The inlet and outlet temperatures are 284° and 323°C at an operating pressure of 15.5 MPa. The large thermal capacity of the coolant effectively reduces any minor mismatch in the heat transferred to and the heat removed from the coolant.

The coolant pumps are located on the inlet side of the reactor vessel, in the “cold leg” of the coolant loop. Each pump is a vertical, single-stage, shaft-seal centrifugal pump (Fig. 5.5). The coolant is sucked by the impeller through the bottom of the casing and discharged through the diffuser (where the velocity is converted to pressure) and via an exit nozzle in the side of the casing.

The pump employs a controlled leakage seal system to restrict the leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water and vapour from the pump into the reactor containment atmosphere.

The pump has an air-cooled motor with oil-lubricated thrust and radial bearings. The motor is equipped with a flywheel to reduce the effects of a power loss on the coolant circulation. The coolant loops are designed so that the steam generators are placed at a higher level than the reactor in order to facilitate natural circulation of the coolant. In the event of a power failure, the reactor is tripped and natural circulation ensures decay heat removal to prevent core overheating.

5.2.2 Pressurizer

The reactor vessel is completely filled with water during normal operation. The only free water surface in the reactor coolant system is in the pressurizer vessel. The pressurizer maintains the required amount of coolant, limits the pressure changes caused by coolant thermal expansion and

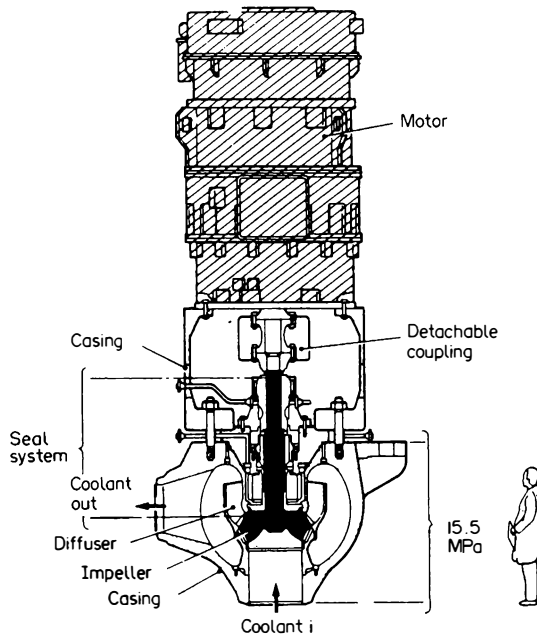


FIG. 5.5. PWR main coolant pump (Westinghouse). Flow capacity $5.7 \text{ m}^3/\text{s}$.
Pressure head 0.8 MPa

contraction during normal load transients, and prevents the pressure in the primary system from exceeding the design pressure.

A typical pressurizer is shown in Fig. 5.6. The lower part of the pressurizer vessel is filled with saturated water and the upper part with steam. The bottom section contains electric heaters and the upper part spray nozzles.

The pressurizer is connected to a high point in the hot leg in one of the coolant loops via a surge line. The pressure is controlled by increasing or decreasing the steam cushion above the water surface in the pressurizer.

The electric heaters are automatically actuated if the pressure in the primary system decreases, thereby flashing water to steam and compensating for the pressure decrease. If the pressure increases, the spray system, which is fed from the cold legs of two coolant loops, is automatically actuated, causing the steam to condense and thus counterbalancing the pressure increase. If the pressure exceeds a preset value, the safety and relief valves in the upper head of the pressurizer vessel open and discharge steam into the pressurizer relief tank.

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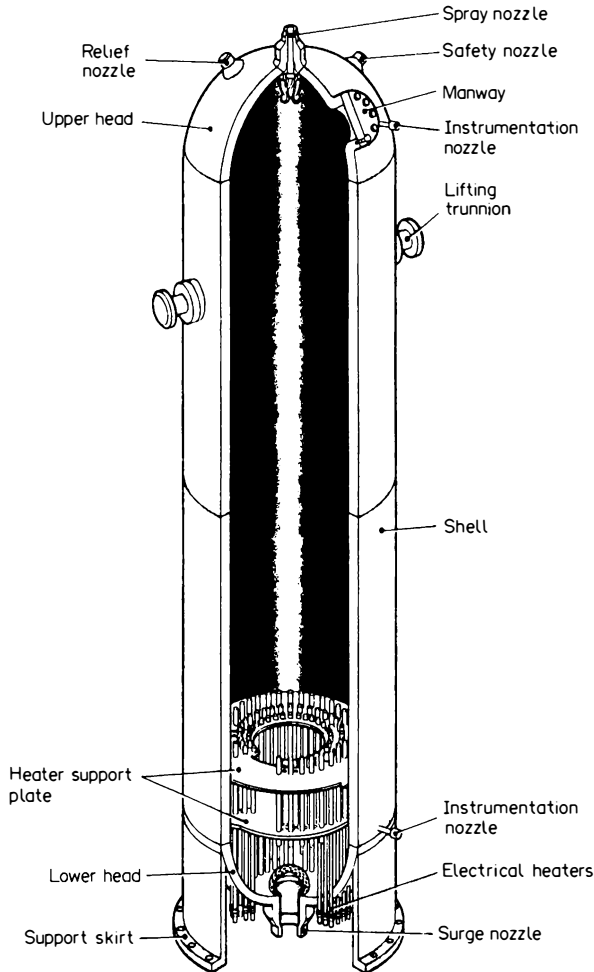


FIG. 5.6. A PWR pressurizer. From *Advances in Power Construction*, Pergamon Press, 1986

5.2.3 Steam generators

The steam generators in Westinghouse reactors are of the shell and U-tube type (Fig. 5.7). The hot primary water enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle. It passes through several thousand U-shaped tubes and leaves the steam generator via an outlet nozzle at about the same level as the inlet. The inlet and outlet channels are separated by a partition.

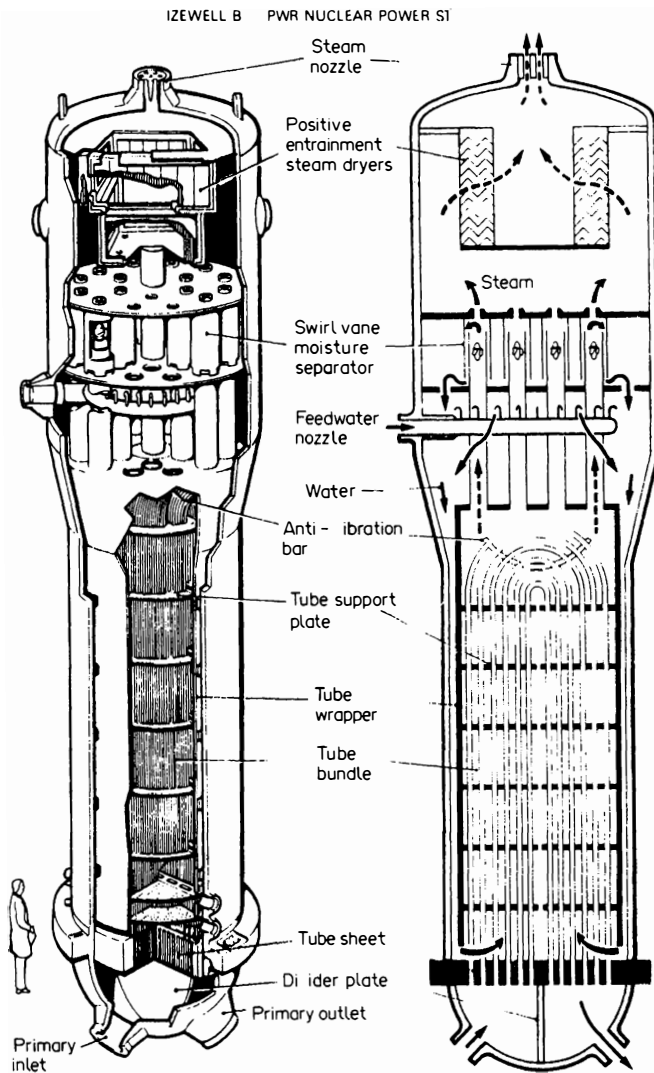


FIG. 5.7. Inverted U-tube type steam generator. From *Advances in Power Construction*, Pergamon Press, 1986

On the secondary side, the feedwater passes through the downcomer located between the tube wrapper and the steam generator wall. The flow reverses at the tube sheet in the bottom of the steam generator and is directed upwards along and across the tube bundles. The feedwater is heated to saturation temperature and enters the boiler section. Subsequently, the water-steam mixture flows upwards to the steam drum section. The moisture separators recirculate water to the downcomer section

where it mixes with incoming feedwater. The steam rises through steam driers which limit the moisture content of the steam to a quarter of a percent or less under all design load conditions.

The steam generator is about 20 m high and has an outer diameter of about 4.5 m in the upper part of the shell. The operating pressure is 6 MPa. In Ringhals 3 and 4, the heat transfer surface of the tubes is about 4500 m² and the steam flow is about 500 kg/s.

The detailed design of different models of steam generators varies slightly. In Ringhals 3 and 4, the feedwater inlet is located in the bottom part of the shell near the tube sheet. The feedwater enters through the preheater section of the tube bundle cold leg, at right angles to the tubes.

The steam generators are mainly manufactured of carbon steel, clad with stainless steel on the primary side. The tubes are made of Inconel, a corrosion-resistant, nickel-based alloy. The tubes are rolled to the tube sheet and supported by several horizontal plates located at intervals along the length of the tube bundle. Water leakage from the primary side to the secondary side due to faulty tubes has occurred in several pressurized water reactors. The area around the tube sheet is particularly susceptible to various types of damage.

5.3 Reactor Containment

The reactor containment is a leaktight, pressure-resistant structure surrounding the reactor coolant system. It forms a biological shield around the reactor vessel and the steam generators and prevents the release of radioactive substances to the environment. The pipes passing through the containment are equipped with isolation valves.

5.3.1 Dry containment

The pressurized water reactor containment has a greater volume than that of the boiling water reactor, since, in addition to the reactor vessel and the main coolant pumps, it also contains the steam generators and the pressurizer (see Fig. 5.8). The containment also acts as the base for the overhead travelling crane which is used to lift the reactor vessel. Because the containment is very large, it can withstand pressure increases due to leakage or pipe breaks in the primary system without special equipment for pressure suppression. Moreover, the containment does not have to be inerted since any hydrogen formed in an accident will be so diluted that the likelihood of a global hydrogen explosion is minimal.

Swedish pressurized water reactors have prestressed concrete containments with embedded steel liners. The volume is 58,000 m³ and the design pressure 0.4–0.5 MPa. The vessel is a 55 m high concrete cylinder with an inner diameter of 35.4 m and with a wall thickness of 1.1 m. The internal

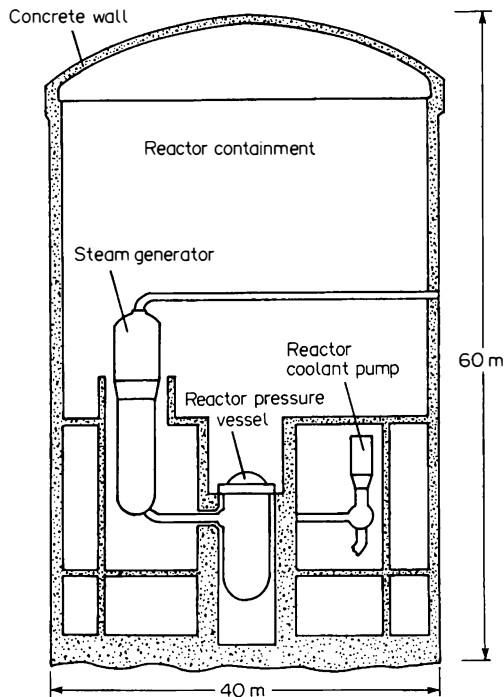


FIG. 5.8. Reactor containment for a Swedish pressurized water reactor

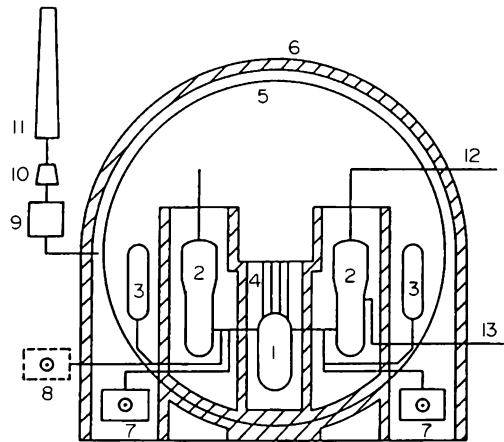
concrete structures consist of beam frameworks and radiation shields around the reactor vessel, the main coolant pumps, and the steam generators.

All the pipelines passing through the walls of the containment have inner and outer isolation valves. The valves allow the containment to be sealed off if required, thereby preventing the escape of any radioactive substances to the environment. There is also a closed ventilation system with fans and a heat exchanger for cooling the components inside the containment. No air from the containment is released during normal operation.

In the event of a pipe break inside the containment, the atmosphere is cooled by water from the spray system in the ceiling. The containment spray system uses water from a sump in the base of the building forming a closed circuit. The water is cooled by heat exchangers to the ultimate heat sink, which is the sea.

5.3.2 Other containment designs

German pressurized water reactors have a *double containment*. This type of containment comprises an inner spherical steel structure and an outer



1 Reactor pressure vessel	8 Chemical and volume control system
2 Steam generator	9 Off-gas system
3 Pressurizer	10 Filter
4 Control rods	11 Stack
5 Inner containment	12 To turbine
6 Outer containment	13 From feedwater pump
7 Emergency core cooling system	

FIG. 5.9. Reactor containment for a German pressurized water reactor. From *The German Risk Study Nuclear Power Plants*, Verlag TÜV Rheinland, 1980

hemispherical concrete structure (Fig. 5.9). The space between the two structures is kept below atmospheric pressure by a ventilation system. Any minor leakage flow from the inner containment is filtered before reaching the environment.

Another concept is the *ice condenser containment*, introduced by Westinghouse. Ice is used as a heat sink, condensing any steam that may leak from the reactor coolant system and limiting the containment pressure in a major loss of coolant accident. The ice is stored in the space around the containment walls. The design pressure and volume of the ice condenser containment are lower than those of an ordinary dry containment.

5.4 Control Systems

Pressurized water reactors have inherently stable power control characteristics. If the load on the turbine generator increases, the heat extracted on the secondary side of the steam generators increases, and the temperature on the primary side decreases. The lower moderator temperature results in an increase in reactivity and, consequently in an increase in fission power.

In order to balance the heat supplied and the heat removed during differ-

ent operating conditions, several control systems are employed. The most important control parameters are:

- reactivity,
- coolant volume,
- water level of the steam generators,
- steam flow to the turbine.

For a description of reactor pressure control, see section 5.2.2.

5.4.1 Reactivity control

Full-length control rods are used for fast reactivity control. A few rods are partially inserted into the core, and by varying their position it is possible to rapidly compensate for variations in reactor power and temperature. During normal operation, the other full-length control rods are completely withdrawn from the reactor and only used for reactor shutdown.

Slow variations in reactivity, such as those resulting from fuel burn-up, are compensated for by changing the boron concentration in the coolant, which is called *chemical shim*ing. Boron is dissolved in the coolant as boric acid. The boron concentration is highest at the beginning of the operating cycle shortly after refuelling.

The boron system can be used for shutting down the reactor should the control rods be inoperable. During start-up, the boron concentration is changed in order to compensate for the reactivity temperature defect (cf 3.3.5). When the operating temperature is reached, the control rods are used to increase the power.

5.4.2 Chemical and volume control

The purpose of the chemical and volume control system is to:

- offset variations in coolant volume due to changes in temperature;
- replace any coolant lost during minor leakage in the primary system;
- adjust the boron concentration in the primary coolant.

The chemical and volume control system includes the volume control tank and three parallel charging pumps as well as storage tanks containing boric acid and deionized water. The water level in the volume control tank is adjusted so as to maintain the required inventory of coolant in the primary system. The composition of the make-up water is adjusted so that the required concentration of boric acid is maintained in the primary coolant.

The system is manually controlled from the central control room. The normal operating mode is “automatic make-up” in which boric acid and

deionized water are blended to the same composition as that of the reactor coolant. The solution is fed to the suction side of the charging pump. When the water level in the volume control tank reaches the required level, the make-up ceases. Other operating modes are “dilution” and “boration”. Deionized water and concentrated boric acid are then supplied at the required rate and amount.

5.4.3 Feedwater control system

The purpose of the feedwater control system is to balance the feedwater flow to the steam generators and the steam flow to the turbine. This is achieved by regulating the water level on the secondary side of the steam generators.

5.4.4 Power control

During normal operation, the generator power is adjusted to the grid demand by regulating the admission of steam to the turbine so that the turbine generator speed is kept constant (frequency control). The reactor power follows the turbine power, i.e. the reactor acts as slave to the turbine (cf 4.5.4). The speed of the main coolant pumps is constant. The position of the control rods is automatically adjusted so that the average temperature of the reactor coolant is kept constant within 30–100% of nominal power.

When, under operating conditions, more steam is generated than required by the turbine, the excess steam is led directly to the turbine condenser via bypass valves. The dumping capacity is sufficient to accommodate the steam flow in a full load-rejection transient. If the reactor power cannot follow the load variations on the grid, the turbine power can be reduced by means of a steam pressure regulator to prevent the pressure from dropping below a preset value.

5.5 Main Technical Data for Swedish Pressurized Water Reactors

The description of the pressurized water reactor is summarized in Table 5.1 for the Swedish PWRs:

- Ringhals 2, commissioned in 1975, capacity 800 MWel;
- Ringhals 3, commissioned in 1980, capacity 915 MWel.

Ringhals 4, which was put into operation in 1982, has the same data as Ringhals 3.

TABLE 5.1 Main technical data for Swedish pressurized water reactors

Parameter	Unit	R2	R3/4
REACTOR VESSEL			
Operating pressure	MPa	17.1	17.1
Operating temperature	°C	343	343
Total weight	kg	327,000	330,000
Total height	m	13.0	13.0
Inner diameter	m	3.99	3.99
Wall thickness incl liner	mm	200	200
THERMOHYDRAULICS			
Thermal power	MWth	2440	2783
Steam flow rate	kg/s	1333	1521
Coolant flow rate	kg/s	12,640	12,860
Operating pressure	MPa	15.4	15.5
Feedwater temperature	°C	221	221
Coolant temperature, inlet	°C	289	284
Coolant temperature, outlet	°C	323	323
Fuel power density	kW/kg U	35.8	38.4
Fuel rod linear heat rate, average	kW/m	20.2	17.0
Fuel rod linear heat rate, max	kW/m	52.6	38.7
REACTOR CORE			
Fuel weight, total	kg U	68,200	72,400
Number of fuel assemblies	—	157	157
Number of rod positions per assembly	—	15 × 15	17 × 17
Rod length	mm	3658	3658
Fuel rods, outer diameter	mm	10.7	9.5
Pellet diameter	mm	9.1	8.2
CONTROL RODS			
Number of control rods	—	53	53
Number of absorbers per control rod	—	20	24
REACTOR COOLANT SYSTEM			
Number of main coolant loops	—	3	3
Number of main coolant pumps	—	3	3
Flow rate per pump	m ³ /s	5.66	5.66
Design head per pump	m	78	81.2
<i>Pressurizer</i>			
Number	—	1	1
Weight	kg	86,000	81,000
Total height	m	12.8	13.0
Outer diameter	m	2.35	2.35
Free volume	m ³	36.8	39.6
Heater capacity	MW	1.3	1.4
<i>Steam generators</i>			
Number	—	3	3
Weight	kg	296,000	312,000
Total height	m	19.0	20.6
Outer diameter, upper part	mm	4464	4475
Outer diameter, lower part	mm	3430	3450
Operating pressure, shell side	MPa	6.0	6.0
Heat transfer surface	m ²	3388	4457
Steam flow rate	kg/s	444	507

Parameter	Unit	R2	R3/4
REACTOR CONTAINMENT			
Volume	m ³	58,000	58,000
Maximum pressure	MPa	0.5	0.4
Maximum temperature	°C	150	150
TURBINE-GENERATOR			
Gross thermal efficiency	%	35.3	34.5
Rated power, net	MW	800	915
Net thermal efficiency	%	32.8	32.9
Number of turbines	—	2	2
Steam flow rate	kg/s	2 × 666.5	2 × 759.7
Steam moisture content	%	0.32	0.40
Pressure/temperature	MPa/°C		
before high pressure turbine		5.9/275	5.9/275
after high pressure turbine		0.6/158	0.7/163
in condenser		0.004/28	0.004/29
Generator speed	rpm	3000	3000
Condenser coolant flow rate	m ³ /s	2 × 17.0	2 × 21.4
Dump capacity	%	90	90
Number of generators	—	2	2
Nominal rating	MVA	508	576.5
Voltage	kV	19.5	21.5
Power factor		0.85	0.85
POWER SUPPLY			
<i>Main transformers</i>			
Number	—	2	2
Nominal rating	MVA	500	500
Voltage	kV	20.5/438	22.6/438.5
<i>Plant transformers</i>			
Number	—	2	2
Nominal rating	MVA	40/25/25	50/25/25
Voltage	kV	19.5/6.8/6.8	21.5/6.8/6.8
<i>Startup transformers</i>			
Number	—	1	1
Nominal rating	MVA	50/40/20	50/25/25
Voltage	kV	145/6.9/6.9	145/6.8/6.8
<i>Diesel generators</i>			
Number	—	4	4
Nominal rating	MVA	3.4	3.45
Voltage	kV	6.9	6.9

Source: Swedish State Power Board, *Ringhals Nuclear Power Station*, 1980

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- 501 Swedish State Power Board, *Ringhals 2 Safety Study*, June 1983
- 502 Swedish State Power Board, *Ringhals 3/4 Final Safety Analysis Report*, April 1984

6

Nuclear Radiation

The radionuclides formed in the reactor fuel during operation are the source of the safety problems associated with nuclear power. To understand these problems, it is necessary to know the conditions for the release of the radionuclides and their health effects. The chapter begins by recalling some basic facts about radioactivity and ionizing radiation. This is followed by an account of the production, release and transport of radionuclides in the reactor during normal operation. Section 6.5 describes the clean-up and waste management systems incorporated in the nuclear power plant. The chapter concludes with a review of principles and practices for radiation protection.

6.1 Basic Concepts

6.1.1 Radioactive transmutation

Radioactivity means that an unstable nucleus, a *radionuclide*, undergoes a spontaneous change through the emission of radiation. Radioactivity was first discovered in certain naturally occurring heavy elements. The radiation was classified into three groups: alpha particles, beta particles and gamma radiation. As a rule, the heaviest elements emit either beta or alpha particles. Although a radionuclide cannot emit both alpha and beta particles, gamma radiation can accompany both alpha and beta radiation.

Alpha particles are helium nuclei containing two protons and two neutrons and thus positively charged. *Beta particles* are positively or negatively charged electrons which arise when a neutron is converted into a proton (or vice-versa) within a nucleus. Alpha particles are emitted with a definite energy, which is specific for the particular radionuclide. Beta particles have a spectrum of energies with a maximum energy characteristic of the emitting nuclide. Gamma radiation is electromagnetic radiation similar to X-rays, but with a higher energy (shorter wavelength).

When an alpha or beta particle is emitted, the chemical identity of the nuclide changes. The daughter nuclide may itself be unstable. A radioactive *decay chain* results, terminating in a stable nuclide. There are three decay

chains in nature starting with U^{238} , U^{235} and Th^{232} and ending with Pb^{206} , Pb^{207} and Pb^{208}

Each radionuclide is characterized by a *half-life*, which is the time taken for half of the radioactivity to decay. The half-life may vary from fractions of a second in short-lived nuclides to millions of years in long-lived nuclides.

The *activity* of a radionuclide is the rate of decay, i.e. the number of nuclear disintegrations per second. The activity is proportional to the number of radionuclides and inversely proportional to the half-life:

$$A = 0.693 N/T_{1/2} \quad (6.1)$$

where A = activity,

N = number of radionuclides,

$T_{1/2}$ = half-life.

Activity is measured in *becquerel* (Bq); 1 Bq = 1 nuclear disintegration per second. An older unit is the *curie* (Ci), where 1 Ci = 3.7×10^{10} disintegrations per second. 1 Ci originally designated the activity in 1 gramme of radium.

6.1.2 Ionizing radiation

As alpha and beta particles pass through matter, their energy is absorbed and the material can become damaged. In general, three types of *radiation damage* occur:

- transmutation of nuclei into other nuclei which may themselves be radioactive;
- displacement of atoms from their normal position in the structure of the material;
- ionization, i.e. the removal of electrons from atoms in the material and the formation of ion pairs in the path of the charged particle.

The first two phenomena arise through the direct interaction between the radiation and the atoms of the material. Neutrons, which have no charge, are particularly efficient at causing this type of radiation damage. This must be considered when designing reactor vessels and core components (cf 3.5.2).

Gamma radiation is electrically neutral and cannot ionize directly. On the other hand, it can cause indirect ionization when colliding with charged particles which are set in motion. Direct ionization is the dominant mechanism for alpha and beta particles. The majority of the ion pairs formed in this way recombine under the release of heat. Nuclear energy, in the form of kinetic energy in the fission products, is converted to heat in the reactor fuel through this process of recombination.

Both alpha and beta particles have a low penetrating power and are easily stopped by relatively small quantities of matter (Fig. 6.1). Alpha particles travel a short, straight distance and have a high ion density along their path. The range of alpha particles in air is a few centimetres. Beta particles are easily scattered due to their small mass and charge. They travel in a non-linear path with a relatively low ion density. The range of beta particles in air is on the order of metres. Gamma radiation is much more penetrating and can only be stopped by thick shielding.

The energy absorbed per unit mass of material is called the *radiation dose* or absorbed dose. The unit for radiation dose is the gray (Gy) which is equivalent to an energy absorption of 1 joule per kilogramme. The unit used earlier was the rad, and $1 \text{ rad} = 0.01 \text{ Gy}$.

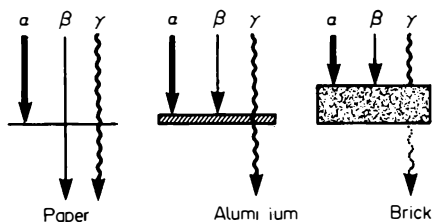


FIG. 6.1. The penetrating power of alpha, beta and gamma radiation

6.1.3 Biological effects

Serious damage can occur to living tissue when it is exposed to ionizing radiation. The effects can be early (acute) or late (latent). *Early effects* arise when so many cells are damaged that the tissue or organ cannot function normally. There is a threshold level of the radiation dose for this type of damage below which no damage occurs. The repair mechanisms of the cell can restore damaged cells at dose levels below the threshold. The extent of damage increases as the radiation dose increases.

Late effects occur when exposure to radiation results in abnormal cell behaviour, e.g. due to changes in the genetic code. Although this type of cell damage occurs randomly, the frequency increases as the radiation dose increases. The degree of damage is independent of the radiation dose. Leukaemia, other cancers and hereditary effects are classed as late radiation effects.

Different kinds of radiation cause different biological damage even if the energy absorbed per mass unit, the radiation dose, is the same. This has to do with the ion density along the radiation path; more heavily ionizing radiation causes greater damage per gray. In order to be able to compare and add total doses for different kinds of radiation, *quality factors* are used. The quality factor $Q = 1$ is by definition used for gamma radiation. $Q = 1$

is also commonly used for beta radiation, which means that gamma and beta radiation have the same biological effects for the same absorbed dose. Q is set equal to 10 for fission neutrons and 20 for alpha particles and fission fragments.

The value of the absorbed dose of a particular kind of radiation is multiplied by its quality factor to obtain the *dose equivalent*. The measure of dose equivalent is the same as that for the absorbed dose, i.e. joule/kg. However, in order to avoid misunderstanding, the unit sievert (Sv) is used when referring to the dose equivalent. Recommendations for radiation dose limits are usually expressed in sieverts. An older unit still in use is the rem, and $1 \text{ rem} = 0.01 \text{ Sv}$.

The dose contribution from a particular radionuclide can be calculated provided that the activity level and the way in which the exposure is obtained are known. The radiation may be *external*, such as gamma radiation from airborne nuclides or ground deposits, or *internal* from substances entering the body through inhalation or ingestion. External radiation affects the whole body, while internal radiation is usually confined to particular critical organs. Doses are expressed as whole-body doses or organ doses.

6.2 Emission Rates

6.2.1 Fission products

During fission, the nucleus splits up into two separate nuclei. Fission does not produce identical nuclei; one nucleus has a larger mass than the other. Moreover, the fission product pairs are not identical for each fission. Irradiated reactor fuel contains up to a few weight percent of fission products consisting of some 200 different nuclides from almost 40 different elements.

Figure 6.2 shows the mass yield of fission products for the three fissile nuclides: uranium-233, uranium-235 and plutonium-239.

Nuclides with mass numbers in the region of 85–105 and 130–150 have a relatively high yield. Many of the fission products are radioactive and decay through the emission of beta particles and gamma radiation. The daughter nuclides can themselves decay into new daughter nuclides, etc. An example of a decay chain is shown in Table 6.1. In this case, the entire fission yield accumulates in the most long-lived nuclide, strontium-90.

There are special computer programs for determining the quantity and composition of the fission products in reactor fuel at an arbitrary time during and after operation. These programs calculate the production of fission products, starting from the number of fissions and the yield per fission. The fission products are then followed with respect to their decay chains and neutron reactions. The formation and transmutation elements heavier than uranium, the transuranic elements or *actinides*, are also represented.

Simplified methods can be used for survey calculations. Two extreme

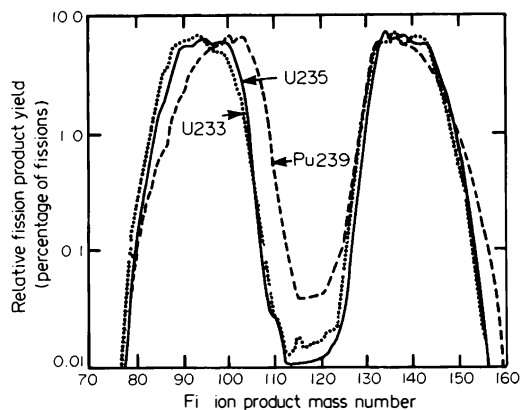


FIG. 6.2. Fission product yield from fission with thermal neutrons. From W Marshall (Editor), *Nuclear Power Technology*, Vol 1 Reactor Technology, Clarendon Press, Oxford, 1983. Used by permission

cases are of interest. If the half-life of the fission product is short compared to the irradiation time, the activity reaches an equilibrium which is determined by

$$A = 310yP \quad (6.2)$$

where A = activity in terabecquerels ($1 \text{ TBq} = 10^{12} \text{ Bq}$),

y = yield in percent of fissions,

P = heat generation in megawatts.

Equation (6.2) can, for example, be used to calculate the activity of the radiologically important nuclides xenon-133 and iodine-131.

TABLE 6.1. Example of a decay chain: mass number 90 from fission of uranium-235

Chain of nuclides	Fission product yield %	Half-life	Cumulated yield %
Selenium-90	0.2	short	0.2
↓			
Bromine-90	1.6	1.4 s	1.8
↓			
Krypton-90	2.7	33 s	4.5
↓			
Rubidium-90	1.2	2.7 m	5.7
↓			
Strontium-90	0.1	30.2 y	5.8

Source: B Lindell, S Löfveberg, *Kärnkraften, människan och säkerheten* (Nuclear Power, Man and Safety), AB Allmänna Förlaget, Stockholm, 1972

If the half-life is very long compared to the irradiation time, the activity increases linearly with time as follows:

$$A = 210yPt/T_{1/2} \quad (6.3)$$

with t = irradiation time,
 $T_{1/2}$ = half-life.

Equation (6.3) is approximately valid for strontium-90 and cesium-137.

The fission products which can be released into the environment are of particular interest for reactor safety. For a release to occur, the fuel cladding, the primary system boundary and the reactor containment shell must be penetrated. The nuclides concerned are mainly gaseous or volatile with a high fission yield, "moderate" half-lives and relevant radiobiological characteristics. Taking all factors into consideration, the analysis can be limited to a few nuclides: certain isotopes of noble gases such as krypton and xenon, volatile elements such as iodine, cesium and tellurium and a few other elements. Some data for these nuclides are shown in Table 6.2.

The noble gases are particularly difficult to contain since they are chemically inert and gaseous. They do not adhere to surfaces or filters, but on the other hand, they neither react with living tissue nor accumulate in the human body. Therefore the health hazards are mainly due to external radiation by airborne activity. Critical nuclides are *krypton-85* and *xenon-133*, which have relatively long half-lives.

TABLE 6.2. Radiologically important fission products

Nuclides	Half-life	Activity ^a TBq/MWth	Radiation
<i>Noble gases</i>			
Krypton-85	10.8 y	7.1	beta, gamma
Krypton-85 m	4.4 h	350	
Krypton-88	2.8 h	830	
Xenon-133	5.3 d	1940	
Xenon-135	9.2 h	410	
<i>Volatile elements</i>			
Iodine-131	8.1 d	940	beta, gamma
Iodine-132	2.3 h	1400	
Iodine-133	21 h	1900	
Iodine-135	6.7 h	1800	
Tellurium-132	3.3 d	1400	
Cesium-134	2.1 y	140	
Cesium-137	30.1 y	70	
<i>Other elements</i>			
Strontium-90	30.2 y	52	beta
Ruthenium-106	1.0 y	310	beta
Barium-140	12.8 d	1800	beta, gamma
Cerium-144	284 d	990	

^aIn fuel with irradiation time 1000 days and cooling time 0 hours. 1 TBq = 10¹² Bq.
Source: B Lindell, S Löfveberg, *loc. cit.*

Iodine isotopes emit high-energy beta and gamma radiation. Therefore, these isotopes contribute to the external dose from a release of airborne radioactive substances in a passing radioactive cloud. The most likely pathway to man is via fallout on grass which is then eaten by grazing animals whose milk is consumed by man. Iodine accumulates in the thyroid gland which is the organ receiving the largest radiation doses. The critical nuclide is *iodine-131* which has the longest half-life (8 days). Calculated releases of iodine-131 have previously been used as a standard measure of the severity of an accident.

The chemical properties of cesium are similar to those of potassium. Cesium reacts chemically with iodine, which affects the magnitude and composition of the release. Cesium is taken up by the muscular tissues of the body but segregates again within a few months. This time period is short when compared to the half-life of the critical nuclide *cesium-137*, which is 30.2 years. Therefore, the content in the body is soon in equilibrium with the content in foodstuffs. The equilibrium value reflects the intake over the previous months. Milk and meat are important pathways to man. Cesium deposition on the ground is the most important potential contributor to long-term health risks following a reactor accident.

Strontium-90 and ruthenium-106 emit only beta radiation and are therefore more difficult to measure than iodine-131 and cesium-137. Elementary strontium is volatile to a certain extent, while the oxide is non-volatile. The opposite is true of ruthenium. For this reason, the oxidation potential in the reactor is important for the composition of the release. The most significant pathway for *strontium-90* is via milk. The critical organ is the skeleton. Strontium segregates slowly; therefore, while the uptake of strontium in the skeleton of an adult is fairly negligible, a growing child will receive a larger quantity. Exposure to *ruthenium-106* by inhalation can result in late effects on the lungs.

6.2.2 Actinides

The actinides are not fission products in the real sense, but are formed through successive neutron capture starting from uranium-238. The most important actinides are presented in Table 6.3. The actinides emit alpha particles and low-energy gamma radiation. They do not, in general, give any external doses and do not accumulate in foodstuffs due to their low solubility. The main health hazard arises from the inhalation of resuspended material from ground deposits. Because of their long half-lives, actinides can contribute to the long-term population dose if they are released into the environment in a severe reactor accident. The long-lived actinides dominate the activity of the spent fuel when the fission products have decayed to stable nuclides. Therefore, they are important for evaluating the long-term

TABLE 6.3. *The most important actinides*

Nuclides	Half-life years	Activity ^a TBq/MWth	Radiation	Critical organs
Plutonium-238	89	1.3	alpha, gamma	skeleton
Plutonium-239	24,000	0.28		
Plutonium-240	6580	0.31		
Plutonium-241	14.7	56		
Plutonium-242	380,000	0.0005		
Curium-242	0.45	15		stomach and intestines
Curium-244	18.2	0.91		

^aIrradiation time 1000 days. Cooling time 0 hours. 1 TBq = 10¹² Bq.

Source: B Lindell, S Löfveberg, *loc. cit.*

environmental effects associated with the final disposal of waste from the nuclear fuel cycle.

6.2.3 Activation products

Activation products are formed when neutrons are absorbed in reactor coolant or structural material in the reactor primary system. Corrosion products can be released into the reactor coolant in dissolved or suspended form and are activated when the coolant passes through the core. Like fission products, the activation products have very different properties, half-lives and harmful effects. As a rule, they are relatively light elements and do not produce any radioactive daughter nuclides. The radiological hazard of activation products is often less than that of the fission products. The most important activation products are given in Table 6.4.

The steam generated in boiling water reactors contains activation products, particularly those originating from the water itself. The most important of these is *nitrogen-16*, which makes it necessary to surround the turbine with radiation shields. Its short half-life, 7.2 seconds, means that the activity rapidly decays when the reactor is shut down. The environmental effects of nitrogen-16 are therefore negligible. In pressurized water reactors, the reactor is isolated from the turbine and therefore the turbine is not radioactive.

The corrosion products in the primary system settle on the surfaces of various components, especially the fuel rods, detach themselves and move on to settle on other components. Therefore, the entire primary system becomes more or less contaminated. The primary coolant is continually purified. It is difficult to determine the production rate of radioactive corrosion products in general. The values in Table 6.4 were estimated on the basis of experience from Oskarshamn I. The critical nuclide is *cobalt-60* due to its long half-life. Cobalt-60 emits high-energy gamma radiation.

TABLE 6.4. *Typical activation products in the primary coolant of a 1000 MWel boiling water reactor*

Nuclides	Half-life	Activity concentration Bq/cm ³
<i>Produced in water</i>		
Nitrogen-13	10 m	220
Nitrogen-16	7.2 s	8.1×10^6
Fluorine-18	1.84 h	190
Fluorine-20	10.7 s	150
Oxygen-19	29 s	0.11×10^6
<i>Corrosion products</i>		
Sodium-24	15 h	70
Chromium-51	27.8 d	100
Manganese-54	313 d	0.4
Manganese-56	2.58 h	190
Cobalt-58	71.4 d	20
Cobalt-60	5.26 y	10
Copper-64	12.8 h	400
Zinc-65	244 d	100

Source: Oskarshamn Nuclear Power Plant Unit 3. Preliminary Safety Analysis Report, AB Asea-Atom and OKG AB, 1975

Also included in the long-lived activation products are *carbon-14*, which has a half-life of 5800 years and *hydrogen-3* or *tritium* (12.3 years). Carbon-14 is mainly produced in the reaction $O^{17}(n,\alpha)C^{14}$. The production of carbon-14 in Swedish boiling water reactors has been estimated at about 2 TBq per GWel and year, of which about 20% is released during reactor operation. The rest is retained in the fuel. The released carbon-14 accumulates in the biosphere and contributes to the global collective dose from nuclear power in the long run.

Although tritium is formed by the activation of deuterium (hydrogen-2) in the primary coolant, it is mainly produced directly in fission and by neutron absorption in boron which is present in boiling water reactor control rods and used for chemical reactivity control in the pressurized water reactor. The tritium which is formed in the fuel and control rods is retained there. The concentration of tritium in the primary coolant is therefore considerably less in boiling water reactors than in pressurized water reactors. For a 1000 MWel boiling water reactor, the tritium content in the primary coolant is estimated at about 700 Bq/cm³. The corresponding content in a pressurized water reactor is at least a factor of 10 higher.

6.3 Fission Product Behaviour

The chemical form and mobility of the fission products in the fuel during normal operation are important factors for the release of the fission products in accident situations. The distribution of the fission products can be deter-

mined if the chemical and physical properties of the elements and the state of the fuel are known. Since the amounts are small and the contents low, the behaviour of the fission products may differ, however, from their usual behaviour in a macrochemical context. For example, surface effects and reactions with small amounts of impurities can be decisive. When studying a particular radionuclide, the decay chain and the presence of stable isotopes of the same element must also be taken into account.

6.3.1 Fission product yields

Some critical fission products were identified in section 6.2.1. In general, these nuclides are not formed directly in fission, but through successive transmutation in decay chains. Table 6.5 provides an overview of the situation for mass numbers 127 to 138, which include isotopes of the chemical elements tin (Sn), antimony (Sb), tellurium (Te), iodine (I), xenon (Xe), cesium (Cs) and barium (Ba). The half-lives of the radiologically important nuclides are in italics.

It can be seen, for example, that most iodine isotopes originate from tellurium. Therefore, the mobility and chemical properties of this element can be the determining factor for the release of iodine in the fuel. Cesium-134, formed by neutron absorption in cesium-133, which in turn derives from iodine-133 and xenon-133, can be expected to behave differently from other cesium isotopes. The table also shows that the yield of stable isotopes of tellurium and cesium is significantly greater than that of iodine.

TABLE 6.5. *The half-life and yield of fission products with mass number 127 to 138. Nuclides produced in fission are placed in brackets. Through the emission of beta radiation, an unstable nuclide will successively change into the stable nuclide on the same line*

Mass number	Total yield %	Sn	Sb	Te	Half-life I	Xe	Cs	Ba
127	0.14	(4.4 m)	3.8 d	9.4 h	stable			
128	0.46	(60 m)	10 m	stable				
129	1.0	(7.5 m)	4.3 h	70 m	stable			
130	2.0	(3.7 m)	(6.3 m)	stable				
131	2.93		(23 m)	(25 m)	8.0 d	stable		
132	4.31		(2.8 m)	(78 h)	2.3 h	stable		
133	6.69		(2.7 m)	(55 h)	21 h	5.3 d	stable	
134	7.92			(42 m)	53 m	stable	2.1 y ^a	
135	6.43			(18 s)	(6.6 h)	9.1 h	stable	
136	6.45			(21 s)	(46 s)	stable	13 d ^a	
137	6.18				(25 s)	(2.8 m)	30 y	stable
138	6.71				(62 s)	(14 m)	32 m	stable

^aFormed by neutron absorption.

Source: *Technical Basis for Estimating Fission Product Behaviour during LWR Accidents*, USNRC Report NUREG-0772, U.S. Nuclear Regulatory Commission, 1981

The critical nuclide iodine-131 has a relatively short half-life and the amount reaches an equilibrium value of about 0.3 g/MWth according to equations (6.1) and (6.2). This value is eventually exceeded by the stable iodine-127 and iodine-129, which, according to equation (6.3), accumulate at the rate of about 2 g/MWth per year. The *total* amount of iodine formed is important for the amount retained in the containment in the event of an accident. The total quantities of various elements are given in Table 6.6.

Fission gases build up an internal pressure in the fuel rods, which can contribute to clad failure if the cladding is overheated. The total yield of krypton and xenon corresponds to about 25 cm³ gas of normal state per MWd of energy.

TABLE 6.6. *Rate of formation of fission products*

Element	mg/MWd	Element	mg/MWd	Element	mg/MWd
Ge	0.011	Ru	65.4	Ba	38.6
As	0.003	Rh	17.1	La	39.8
Se	1.20	Pd	33.4	Ce	86
Ra	0.36	Ag	2.7	Pr	37
Kr	10.4	Cd	1.67	Nd	140.6
Rb	10.2	In	0.08	Pm	8.86
Sr	28.2	Sn	0.97	Sm	27.2
Y	15.2	Sb	0.53	Eu	3.48
Zr	119.6	Te	15.7	Gd	0.036
Nb	0.33	I	5.86	Tb	1.67
Mo	107	Xe	149	Dy	0.005
Tc	27.4	Cs	90.4		

Source: F Abbey, Radioactivity and the Fission Products, in *Nuclear Reactor Safety*, Edited by F R Farmer, Academic Press, 1977

6.3.2 Fission product distribution in fuel

When the fission products are emitted, their kinetic energy is about ten million times greater than the energy of a typical chemical binding. They therefore cause severe disturbances to the atoms in the crystalline lattice of the fuel material. Energy is released as heat along the track of the fission products. This results in local melting and evaporation of UO₂, which however immediately solidifies and recrystallizes. After some burn-up, each molecule will have taken part in the melting and solidification process thousands of times. This leads to sintering and grain growth. At high burn-up, further grain growth is prevented by fission products accumulating in the grain boundaries.

The fission products are foreign atoms in the uranium dioxide lattice. Their behaviour is determined first and foremost by the temperature. Above about 1100°C, the fission products can move fairly freely and search for a

thermodynamically more stable state. This movement is characterized as *diffusion*. There are several different mechanisms at work which all have in common the fact that the diffusion rate increases with the temperature and the oxygen content of the fuel.

The oxygen content of the fuel material is measured by *stoichiometry*, i.e. the ratio of oxygen to uranium atoms. Because the need of the fission products for oxygen is lower than that of uranium, the oxygen content and thereby the atom mobility increases with fuel burn-up. The elements forming stable oxides, such as rare earth metals, strontium, barium, zirconium and others, will exist as oxides under all conditions of practical interest. If the oxygen content is low enough, and if they are sufficiently volatile certain other elements will exist in their elementary form and behave like gases. Such elements include cesium, rubidium, tellurium, iodine and bromine. However, complications arise since the elements can react with each other and with uranium.

Cesium and iodine are of special interest. While iodine does not react with uranium under normal conditions, it probably exists as *cesium iodide* rather than as atomic or molecular iodine. Since cesium and iodine are formed at different places in the lattice structure of the fuel material, it is possible that the iodine will migrate to and be carried away by noble gas bubbles before it meets cesium. The cumulative yield of cesium is about 15 times that of iodine (see Table 6.6). Cesium reacts with uranium and appears at temperatures below about 1000°C mainly as cesium uranate and to a lesser extent as cesium iodide.

The behaviour of the fission products and their distribution in the fuel is very complex. The fission products mostly consist of stable and long-lived nuclides which accumulate as fuel burn-up proceeds. The majority of the fission products are retained in the crystal grains of the fuel material. A small part of them is released to the grain boundaries and an even smaller amount of gaseous and volatile elements is released into the gap between the pellet and the cladding. The temperature, which is proportional to the linear heat rate, is the decisive factor for the release of fission products.

6.4 Fission Product Release

Fission products will be released into the coolant if the cladding is damaged. It is anticipated that minor leaks can occur during normal operation. The filter and clean-up systems of the plant are designed to deal with such leaks. Major radioactive releases can only occur if fuel damage is extensive. This section describes the mechanisms in effect during different conditions and the transport of the released radionuclides in the plant.

6.4.1 Fission product leakage

The fuel rods may have small defects, such as porous end welds, which may remain undetected in spite of careful quality control. The external surface of the rod may be contaminated with microscopic amounts of uranium. Cracks may develop in the cladding during operation, for example, through pellet-clad interaction during too rapid power changes. Fission product activity in the primary coolant system is continually monitored. By analysing the observed activity, three different mechanisms have been found to describe fission product leakage (604). These mechanisms are characterized by different leakage rates and power dependencies (see Table 6.7).

TABLE 6.7. *Mechanisms for fission product leakage.* y is the cumulated yield and T the half-life of the relevant fission product. k_1 , k_2 and k_3 are constants

Mechanism	Activity	Leak rate	Power dependence
Recoil	$k_1 y T^{-1}$	$k_1 y$	linear
Diffusion	$k_2 y T^{-1/2}$	$k_2 y T^{1/2}$	exponential
Equilibrium	$k_3 y$	$k_3 y T$	irregular

The recoil mechanism is characterized by the “leakage” of the fission product at the moment of formation, i.e. the leak rate (at a certain power) solely depends on the fission product yield. Consequently, the observed activity is inversely proportional to the half-life. The activity increases linearly with power. This mechanism is typical of *surface contamination*.

During “diffusion” the leak rate is proportional to the square root of the nuclide’s half-life. This is typical for the time it takes for the nuclide to migrate from its birthplace in the fuel pellet to the surface of the pellet and out into the coolant through a *clad defect*. The activity increases exponentially with power since the fission product release depends exponentially on the fuel temperature.

The mechanism of “equilibrium” refers to cases where the time to leakage is long compared to the nuclide’s half-life. This is typical of leakage through *pinholes* (small pores) in the cladding. The power dependency is irregular in as much as burst releases can be observed during reactor power changes, e.g. at reactor shutdown. These burst releases are characterized as “spikes” in the activity level. Such spikes are mainly found to be associated with iodine-131 and xenon-133.

6.4.2 Release mechanisms during fuel overheating

Fuel heat-up to temperatures from 700° to 1100°C can lead to clad failure due to a combination of internal pressure and the deterioration of cladding

strength. At the moment of failure, a burst of activity takes place. The fission gas inventory of the pellet-clad gap and that of the plenum (see 3.2.1) is released into the coolant. During this *gap release* a few percent of the inventory of stable and long-lived noble gas nuclides in the rod may escape. Cesium and iodine are also released, although in considerably smaller quantities. For isotopes with shorter half-lives than about 30 days, the amount released is essentially lower, since they occur in smaller quantities.

After the instantaneous gap release, the remainder of the cesium and iodine in the gap diffuses out through the crack or via water leaking into the crack (“waterlogging”). This occurs slowly as long as there is no further increase in temperature. At temperatures above 1400°C, noble gases, cesium and iodine accumulating in the grain boundaries of the fuel will be released to the pellet surface and escape through the crack. In a rod with high burn-up, this *grain boundary release* may result in a release of up to 20% of the inventory of stable isotopes of noble gases, cesium and iodine. Grain boundary release can also occur at lower temperatures if the burn-up is high and the grain boundaries are saturated with fission gas.

After gap release and grain boundary release have taken place, 70–90% of the inventory of noble gases, cesium and iodine is left within pores in the crystal grains of the fuel. Fission product release then occurs through *diffusion* from the crystal grains themselves. The rate of release increases exponentially with temperature and is doubled approximately every hundredth degree. This means that at 2000°C about 10% of the remaining noble gas, cesium and iodine inventory is released per minute.

At still higher temperatures, release occurs from *molten fuel*. This process starts when the clad material melts at about 1800°C. Zirconium can then either form alloys with uranium, melting at a lower temperature than the melting point of uranium dioxide (2800°C), or form zirconium dioxide, which melts at 2700°C. The details of the melting process are not completely known. Gaseous and volatile elements are thought to be entirely released from molten fuel while only part of the non-volatile elements is released. The release, transport and removal of fission products during a core melt-down accident are further discussed in Chapter 11.

6.4.3 Transport routes in the plant

Released fission products may escape from the primary system through leakage, removal in the filter and clean-up systems or deposition on surfaces in the primary cooling loops, or they may remain in the coolant. The activity concentration in the coolant depends on the extent of the leakage and the efficiency of the removal systems.

The noble gases are dissolved in the primary coolant. In the boiling water reactor, they follow the steam and are carried to the turbine and turbine condenser where they are evacuated by the condenser’s ejector system. In

the pressurized water reactor, the noble gases are removed from several places, notably from the volume control tank (5.4.2).

Iodine occurs in several different forms dissolved in the primary coolant and is separated in the reactor's clean-up system. Iodine is also dissolved in steam to a certain extent and is carried to the turbine in the boiling water reactor. Some of this iodine is removed by the condenser's off-gas system. The remainder is dissolved in the condensate and separated in the condensate clean-up system.

Iodine can also occur in organic form as *methyl iodide*. Methyl iodide has a low reaction tendency and is difficult to remove with filters. It can therefore be limiting as far as releases from the reactor are concerned. Considerable efforts have been made to identify organic iodine.

Other fission products generally appear as ions in solution or as colloidal oxide particles. They largely remain in the primary coolant and are separated by filters in the clean-up system. A small amount is transferred to the gaseous phase in the form of aerosols. Figure 6.3 shows the most important routes for fission products in boiling water reactors.

Table 6.8 gives an example of the calculated activity concentrations for Oskarshamn III, serving as the design basis for the fission product removal systems. The values correspond to a situation where 1% of all fuel rods is assumed to leak. In reality, the number of leaking rods is considerably smaller. Often there is no leakage at all.

The calculated distribution of fission products between steam and water was mainly based on experience from Oskarshamn I, and shows that the concentration of a particular nuclide in steam is about a hundredth of the concentration in the primary coolant.

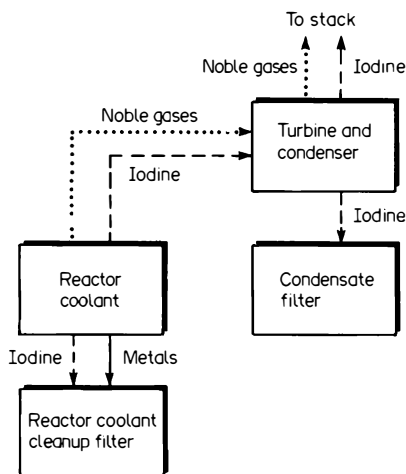


FIG. 6.3. Fission product transport routes in boiling water reactors

TABLE 6.8. *The calculated fission product activity in primary coolant and steam in a 1000 MWel boiling water reactor with 1% failed rods*

Nuclides	Half-life	Activity concentration	
		Reactor coolant MBq/m ³	Steam flow MBq/s
Krypton-85	10.8 y		2.6
Krypton-85m	4.4 h		700
Krypton-88	2.8 h		2300
Xenon-133	5.3 d		930
Xenon-135	9.2 h		1300
Iodine-131	8.1 d	1400	22
Iodine-132	2.3 h	14,000	230
Iodine-133	21 h	7800	126
Iodine-135	6.7 h	12,000	240
Tellurium-132	3.3 d	410	0.67
Cesium-134	2.1 y	7.4	0.011
Cesium-137	30.1 y	9.3	0.015
Strontium-90	30.2 y	9.3	0.015
Barium-140	12.8 d	300	0.48
Neptunium-239	2.4 d	7000	11

Source: *Oskarshamn Nuclear Power Plant Unit 3. Preliminary Safety Analysis Report*, A B Asea-Atom and OKG AB, 1975

6.5 Activity Removal Facilities

In the reactor plant there are special facilities for separating and treating airborne and waterborne radioactive substances. These activity removal facilities include ventilation systems, off-gas systems and clean-up systems. The systems are designed to maintain the releases to the environment below permissible levels during normal operation.

6.5.1 Ventilation systems

Radioactive gases and airborne particulates may escape into the containment and auxiliary buildings through leakage via valves, stuffing-boxes, etc. Ventilation systems for the reactor buildings are therefore equipped with filters for iodine and aerosols. A sub-atmospheric pressure level is maintained in the entire plant so as to prevent airborne radionuclides from escaping through any route other than the stack.

The building compartments of Swedish boiling water reactors are completely isolated from each other as regards ventilation. Each building is served by one or more ventilation systems. In pressurized water reactors all high-pressure systems are located inside the containment. The risk of airborne activity leaking into other plant buildings is therefore minimal. Hence, only the reactor containment needs to be equipped with ventilation for radioactive air.

6.5.2 Off-gas systems

The prime purpose of the off-gas system is to limit the release of radioactive noble gases from the plant. The radioactive noble gas nuclides are mainly isotopes of krypton and xenon. The critical nuclides are xenon-133 with a half-life of 5.3 days and krypton-85 (10.8 years). Other noble gas nuclides have a shorter half-life.

The off-gas system delays the noble gases, so that the radionuclides, particularly the short-lived ones, have time to decay. In boiling water reactors this process occurs after the ejector system of the turbine condenser, and in pressurized water reactors after the volume control tank. In principle, the noble gases are separated from the carrier gas (air) and allowed to decay in one or several vessels. Separation normally takes place through the adsorption of gas molecules on filters with a large surface-to-mass ratio. Since heavy molecules are adsorbed to a higher degree than light molecules, the heavier noble gas molecules are separated from the lighter air molecules.

In modern off-gas systems, adsorption is carried out in charcoal and sand beds (Fig. 6.4). The gas first passes through recombiners for hydrogen and oxygen, resulting from the radiolysis of water in the reactor. The gases then pass through the first sand bed to the first adsorption column after which the flow separates into two streams. The main stream is driven by a fan through the second (outer) sand bed and through filters to the stack. The second stream is returned to the turbine condenser through the second column.

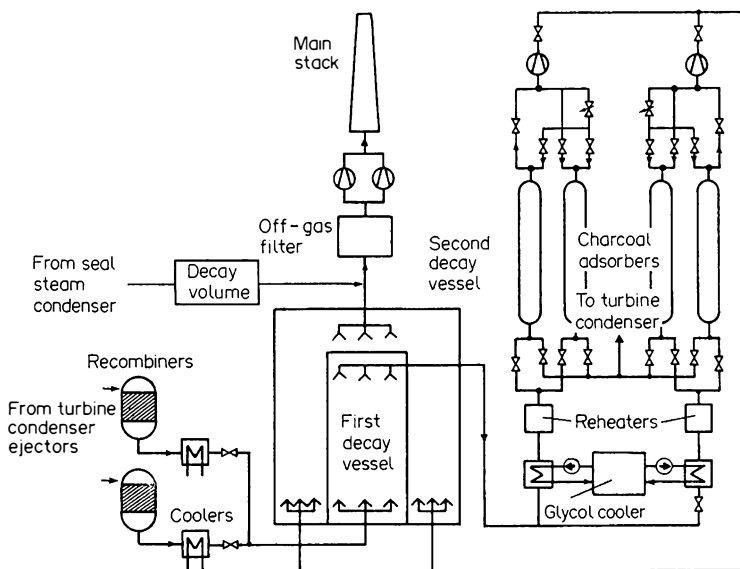


FIG. 6.4 Flow chart of an off-gas system. Courtesy AB Asea-Atom

The columns operate alternately in accordance with the pressure oscillation principle. In the first column, adsorption at atmospheric pressure takes place and in the second, desorption at lower pressure. The adsorption column delays the noble gases and iodine in the off-gases relative to the air. Iodine is completely retained in the column. Krypton passes through in a couple of hours. When xenon begins to break through after 20–30 hours at nominal air flow, a change-over to another column is made.

6.5.3 Clean-up systems

The water in a reactor plant must be continually cleaned during operation to remove active and inactive impurities. In boiling water reactors, the water clean-up systems comprise a full-flow system for the condensate and a partial-flow system for the primary coolant (Fig. 6.5).

The condensate clean-up system contains parallel filters with ion-exchange resins. The majority of the corrosion products formed in the turbine, condenser and the preheater located before the condensate clean-up system are removed. The ionogenous impurities, such as chlorides, which can enter the condenser with in-leaking condenser cooling water, are also removed.

The purpose of the clean-up circuit in parallel to the main coolant recirculation system is to separate ionogenous and colloidal impurities from the primary coolant. This occurs in bed-type ion exchangers. The working temperature is lower than 90°C, and the primary coolant must therefore be cooled before it passes through the filter.

In pressurized water reactors, the main coolant system has a parallel ion-exchanger clean-up circuit connected to the volume control system. The

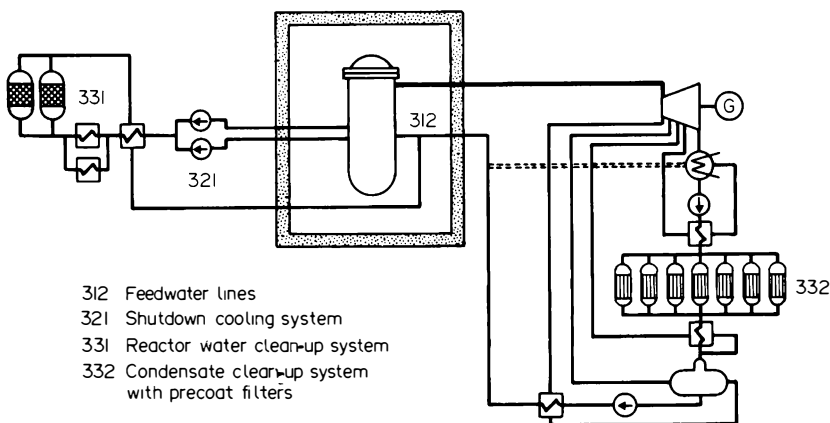


FIG. 6.5 Water clean-up systems for boiling water reactors. Courtesy AB Asea-Atom

secondary system is usually purified by means of a blowdown flow in the steam generator.

6.5.4 Decontamination

A successive deposition of radioactive materials takes place on surfaces in contact with the primary coolant water. This contamination is mainly caused by corrosion products but also by fission products. The corrosion products are deposited and activated on the fuel rod surfaces. The thickness of the deposits increases with time and the level of radioactivity becomes very high. The deposits change character as they grow and become less adhesive. They flake away from the surface in the form of particles which are carried by the coolant to other parts of the primary system and deposit there. Since a certain fraction of the fuel is replaced each year, an equilibrium is eventually reached when the concentration of radioactive substances in the primary coolant is approximately constant.

The radiation level in outer parts of the reactor system may be so high as to prevent or severely limit access by service personnel. When repair and maintenance are made difficult by the radiation hazard, it may be necessary to remove the radioactive deposits from, i.e. to *decontaminate*, certain components or even entire subsystems. The build-up of cobalt-60 on system surfaces poses particularly severe problems.

Decontamination can be carried out by mechanical or chemical means or by a combination of both. Mechanical decontamination consists of brushing, blasting or flushing and is often used on components. Chemical methods can be used on both components and systems and consist of the complete or partial dissolution of the radioactive oxide on system surfaces, e.g. by decreasing the pH. Oxide solubility can also be increased by the application of suitable complexing agents.

6.5.5 Waste management systems

Spent ion exchange resins from filters, drainage water from reactor systems and decontamination fluids, etc., are collected in tanks for liquid effluents. The liquid effluents are distributed to different subsystems depending on the activity level and impurity content.

Low-level effluents are discharged under controlled conditions into the coolant channels of the reactor plant. Intermediate level effluents are purified by ion-exchange filters or are evaporated. The clean water is returned to the reactor system. Active filter resins and concentrated active solutions are taken to storage tanks, where the majority of the short-lived nuclides decay, and then to the treatment system for radioactive waste.

In the solid radioactive waste system, the filter resins and evaporation concentrates are processed and cast into concrete or bitumen. Other solid,

low-level wastes from the reactor plant are compacted and enclosed in steel drums.

6.6 Radiation Protection

Radiation protection generally concerns the radiological safety of the plant staff and the general public during *normal* reactor operation. In this section the basic approach to radiation protection is outlined.

6.6.1 Recommendations and regulations

Radiation protection activities are generally governed by recommendations of international organizations and by standards established by national supervisory authorities. International bodies such as the International Commission on Radiological Protection (ICRP), the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the World Health Organization (WHO) advocate the following main principles:

- no practice involving radiation exposure shall be accepted unless it can be shown to produce a net benefit to society;
- all radiation doses shall be kept as low as reasonably achievable, economic and social factors being taken into account;
- the dose equivalent received by individuals shall not exceed specified limits, allowance being made for future developments.

According to the ICRP's recommendations, the following individual dose equivalent limits are applicable (1985):

- dose equivalent to occupational workers, 50 millisieverts (mSv) per year;
- dose equivalent to individual members of the general public, 1 mSv per year.

The above are whole-body dose equivalents. There are also ICRP recommendations on dose equivalents to organs (cf 6.1.3). The weighted whole-body dose equivalent, or *effective dose equivalent*, is the sum of the dose equivalents to the affected organs, multiplied by weighting factors. The weighting factors (Table 6.9) give the proportion of the risk for cancer and hereditary effects which the organ represents in whole-body exposure.

The *collective dose* is the sum of all individual effective dose equivalents to the population. The unit for measuring the collective dose is the mansievert. The *dose commitment* is the sum of all future annual collective doses resulting from one year's release (Fig. 6.6). The aim of the dose commitment

TABLE 6.9. *Weighting factors for calculating the effective dose equivalent*

Organ or tissue	Weighting factor
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung tissue	0.12
Thyroid glands	0.03
Bone tissue	0.03
Other organs	0.30
Whole body	1.00

Source: International Commission on Radiological Protection, Recommendations of the ICRP, ICRP Publication No 26, *Annals of the ICRP*, Vol 1, No 3, 1977

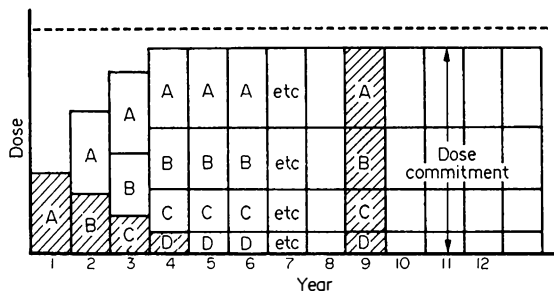


FIG. 6.6. The concept of dose commitment. From B Lindell, S Löfveberg, *Kärnkraften, människan och säkerheten* (Nuclear Power, Man and Safety), Allmänna Förlaget, Stockholm, 1972

concept is to estimate and limit the future collective dose arising from an expanding nuclear industry.

Since 1981 the following regulations have been in effect in Sweden concerning the release of radioactive substances from nuclear power plants (605):

- the sum of the effective dose equivalents to residents in the vicinity of the plant shall not exceed 0.1 millisieverts per year;
- the global collective dose commitment shall not exceed 5 mansieverts per year and gigawatt electrical power;
- the discharge of radioactive substances shall be monitored and regularly reported to the radiation protection authority. The accuracy and function of the measuring equipment shall be approved by the authority and shall be subject to periodic inspection;

- if the discharge per week exceeds a prescribed value, a report shall be submitted to the radiation protection authority within one week with a proposal for countermeasures;
- if the discharge per hour exceeds a prescribed value the reactor shall be shut down.

If these requirements are fulfilled, acute radiation effects to the individual are ruled out. The reference value 0.1 mSv/year gives an additional contribution to the natural radiation environment which is less than 10%.

6.6.2 The ALARA principle

Safety in normal operation means ensuring that radiation exposure of reactor operators and the general public are within specified limits. This is achieved by operating the activity removal facilities according to the design specifications, by minimizing the gaseous and liquid discharges, and by carefully planned service and maintenance operations.

Keeping radiation exposure within limits is not enough, however. It is also required that the radiation doses are held “as low as reasonably achievable” This is known as the ALARA principle which was formulated by the ICRP at the end of the 1970s (606). The ALARA principle is essentially a guideline for optimizing radiation protection measures, based on the possibility of making quantitative risk estimates.

The ALARA principle can be applied, for example, by using cost-benefit analysis. This means that any effort to reduce collective doses, costing less than a specified amount per dose reduction decrement, should also be undertaken. The rationale behind the ALARA principle is that, while it is always possible in theory to further reduce radiation dose, this will require successively increasing expenditure. Thus, there must be an optimum level of radiation protection beyond which it is unreasonable to go. The problem is to define an acceptable level of maximum incremental cost per dose reduction decrement.

6.6.3 Radiation protection at the plant

The nuclear power plant staff can be exposed to external radiation from radioactive components and systems as well as radiation from airborne radioactivity entering the body by inhalation or ingestion. The plant staff is protected from external radiation by shielding and by restricted access to certain areas. Airborne activity is controlled by room segregation and ventilation.

The shielding mainly consists of concrete, although the steel and water in the reactor systems as well as the reactor pools also act as shields. The concrete shields are to a large extent identical with the walls of the buildings

and the reactor containment (cf Fig. 4.7). However, they are thicker than normal in some places. Around the reactor vessel and turbine (BWR) they can be up to 2 metres thick.

With regard to radiation protection, the rooms of the plant are classified by successively increasing limits for the radiation level. In areas with the lowest radiation level, the entire working week could be spent without exposure to doses higher than those specified in the ICRP's recommendations. Access to areas in the highest radiation category can only be allowed for a short period of time and under the control of personnel with direct-reading radiation counters.

The room classification is also applicable to areas where airborne and surface contamination can occur. Since the airborne activity can change rapidly, the classification is usually based on the risk of contamination rather than on the normal radiation level. This means, for example, that areas with systems that are pressurized from the reactor, must not be entered without radiation monitoring, while there is no time limit for access to clean areas along the external walls of the building.

An important radiation protection measure is the division of the plant into controlled and uncontrolled areas. All areas subjected to high levels of external radiation or airborne and surface contamination belong to the controlled area. There is usually only *one* normal entrance to the controlled area which is under the surveillance of a guard or monitored from the control room via TV camera. All other entrances to the controlled area are usually locked and can only be opened with special permission.

When an employee enters the controlled area, he wears a personal dosimeter which he must return on leaving the area. In general, these are not direct-reading instruments and therefore must be read once a week. At the entrance, employees can be monitored by direct-reading counters to find out whether or not they have been contaminated with radioactive materials. Every nuclear power plant also has a whole-body counter for registering and monitoring any intake of radioactive substances into the body.

As previously mentioned (6.5.1), the ventilation systems contribute to minimizing airborne activity. Ventilation is arranged so that air flows from low to high radiation level areas from where it is then filtered and exhausted through the stack. Airborne activity is thereby prevented from spreading from more to less contaminated areas.

6.6.4 Discharge of airborne activity

Individuals and residents in the vicinity of a nuclear power plant can be exposed to radiation from radioactive substances discharged via stack air or drainage water. The airborne materials will primarily expose nearby residents to external radiation from passing radioactive clouds or lead to inter-

nal doses through inhalation. Secondly, ground deposition of certain nuclides may become important.

The discharge of radionuclides is continually monitored by nuclide-specific measuring systems. Radiation doses in the environment can be calculated from these measurements and meteorological data. Direct measurements of activity concentrations are carried out in the surrounding area. However, permissible dose limits are so low that variations in the natural background radiation almost completely disguise the activity contributions from the stack air.

Stacks in Swedish boiling water reactor plants are so high that they rise above the leeward vortex of the building (Fig. 6.7). Hence, radioactive substances released from the stack do not descend to ground level close to the plant and are therefore not sucked into the plant ventilation air intake. The substances will be carried with the wind, spreading out in a plume which will disperse as it gets further away from the plant. The concentration of radioactive substances will therefore decrease with distance.

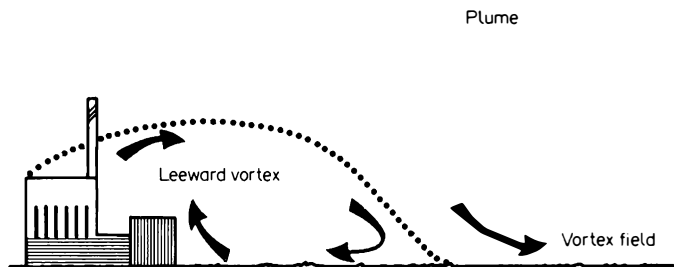


FIG. 6.7. Air flow around a reactor plant. From *Nuclear Power and Safety*, AB Asea-Atom, 1972

The dominant nuclides for the external dose are the noble gas nuclides krypton-85 and xenon-133. In boiling water reactors, the most important factors determining the dose from these nuclides are:

- the extent of clad damage in the core which determines the primary release of fission products;
- the extent of air leakage into the turbine condenser which affects the delay time in the off-gas system.

Figure 6.8 illustrates how a combination of clad damage and condenser air leakage in Oskarshamn III could result in a calculated whole-body dose of 0.05 mSv/year at a distance of 1 km from the plant. In practice, the leakage rate would probably be about 10 kg/hour or lower and the number of failed fuel rods substantially less than 1%. Consequently, the whole-body dose is only some per mille of the permissible values.

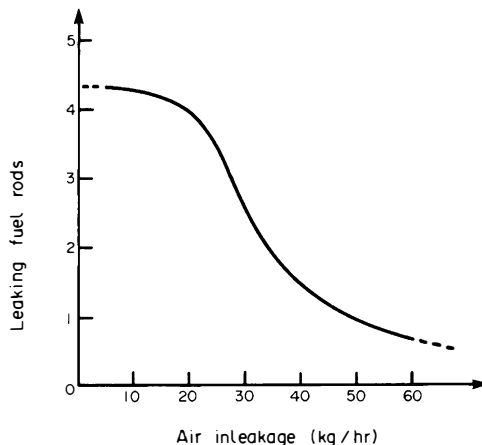


FIG. 6.8. Combinations of clad damage and air leakage into the turbine condenser which will result in a whole-body dose of 0.05 mSv/year 1 km from Oskarshamn III. From *Oskarshamn Nuclear Power Plant Unit 3, Preliminary Safety Analysis Report*, AB Asea-Atom, 1975

As previously mentioned (6.2.1), iodine-131 is generally the critical nuclide for individuals living near the plant. Iodine accumulates in the thyroid gland, and to an especially high degree in children. Milk is the most important pathway. In boiling water reactors, the discharge of iodine-131 is mainly affected by:

- the extent of clad damage and thus of the iodine-131 content in the primary coolant;
- the extent of steam leakage into the turbine building, where the ventilation air is not filtered.

Calculations for Oskarshamn III show that even with very unfavourable assumptions, the iodine activity in the stack air during normal operation falls far short of the permissible values.

The discharge of noble gases in the stack air of a pressurized water reactor (Ringhals 3) has been estimated at 300 TBq/year, about equally distributed between krypton-85 and xenon-133, and assuming 1% leaking fuel rods. This can be compared with the corresponding value for Oskarshamn III which has been estimated at 1600 TBq/year. The calculated doses from these releases are negligible compared to those obtained from the natural background radiation.

In practice, the discharges are lower than the calculated values, mainly since the number of leaking rods is much smaller than the assumed 1%. For example, during 1981 a noble gas activity in the stack air of Ringhals 3 was measured at about 50 TBq. The activity mainly originated from xenon-133.

TABLE 6.10. *Airborne discharge from Swedish nuclear power plants expressed in units of reference release*

Nuclear power plant		Annual release		
		1981	1982	1983
Barsebäck	unit 1	1.3E-3*	2.6E-3	3.9E-4
	unit 2	2.0E-5	1.4E-5	1.6E-5
Forsmark	unit 1	7.0E-6	3.2E-6	3.1E-6
	unit 2	2.0E-7	1.8E-6	1.0E-5
Oskarshamn	unit 1	2.0E-1	6.6E-2	4.2E-2
	unit 2	4.8E-3	1.7E-3	9.5E-3
Ringhals	unit 1	2.6E-1	4.1E-1	3.9E-2
	unit 2	7.3E-4	1.3E-3	7.6E-4
	unit 3	2.7E-3	5.0E-4	3.4E-4
	unit 4		1.6E-5	1.4E-4

*1.3E-3 = 1.3×10^{-3} = 0.0013.

Source: National Institute for Radiation Protection, *Activity Releases and Occupational Exposures of the Nuclear Power Industry*, Quarterly Report K82-12, Stockholm, 1983

Table 6.10 gives the air releases from all the Swedish nuclear power units during 1981-3, expressed in units of *reference release*. A reference release is equal to a release giving a radiation dose of 0.1 mSv/year to persons living near the plant, i.e. the limit value prescribed by the radiation protection authority (see 6.6.1).

6.6.5 Discharge of waterborne activity

Waterborne radioactive substances can reach man via drinking-water or fish, shellfish, etc. In many countries, nuclear power plants are situated near rivers and lakes, which can make water-related issues a problem. In Sweden, aqueous wastes are discharged into the sea, which excludes problems relating to drinking-water. Instead, discharge limits arise from the risk of concentrating radioactive substances in foodchains. These chains are often long and difficult to analyse. The kind of comparison which can be made between the natural background radiation and noble gases discharged into the air cannot be performed for discharges of aqueous activity. While it is true that the sea already contains large quantities of naturally radioactive elements, such as radium, the substances discharged from nuclear power plants have other properties which makes a comparison difficult.

As with airborne activity, discharges of waterborne activity are continually monitored. For example, Table 6.11 presents the measured activity of the most important radionuclides in the waste cooling water of Oskarshamn and Ringhals during 1982. Since the units at a site use common cooling channels, the total release for each site is given. The activity from tritium

TABLE 6.11. *The total activity discharged to water during 1982 from Oskarshamn (OI, OII) and Ringhals (R1, R2, R3, R4) in gigabequerels (1 GBq = 10⁹ Bq)*

Nuclide	Half-life	Oskarshamn GBq/yr	Ringhals GBq/yr
Tritium	12.3 y	560	19,000
Chromium-51	27.7 d	14	5.5
Manganese-54	312 d	8.7	2.2
Cobalt-58	70.8 d	14	18
Cobalt-60	5.3 y	62	33
Zinc-65	244 d	41	23
Antimony-124	60 d	1.3	12
Iodine-131	8.0 d	2.6	3.0
Cesium-134	2.1 y	4.3	25
Cesium-137	30.3 y	11	34
Barium-140	12.8 d	2.4	23
Lanthanum-140	40.3 h	5.4	0.02

Source: National Institute for Radiation Protection, *Activity Releases and Occupational Exposures of the Nuclear Power Industry*, Quarterly Report K82-12, Stockholm, 1983

TABLE 6.12. *Waterborne discharge from Swedish power plants, expressed in units of reference release*

Nuclear power plant	Annual release			Units
	1981	1982	1983	
Barsebäck	6.0E-3	9.6E-3	4.8E-3	B1 and B2
Forsmark	6.0E-5	1.1E-3	2.4E-3	F1 and F2
Oskarshamn	8.8E-3	1.2E-2	7.5E-3	OI and OII
Ringhals	2.6E-2	1.1E-2	1.6E-2	R1, R2, R3 and R4

Source: National Institute for Radiation Protection, *Activity Releases and Occupational Exposures of the Nuclear Power Industry*, Quarterly Report K82-12, Stockholm, 1983

dominates, especially in Ringhals. The higher tritium activity in Ringhals is due to the higher production of tritium in pressurized water reactors than in boiling water reactors (cf. 6.2.3).

Measured aqueous discharge from Swedish power plants during 1981-3 is given in Table 6.12.

When the releases from Tables 6.10 and 6.12 are summed, it can be seen that the total annual dose of airborne and waterborne activity during the 3-year period is far below the prescribed limits. The highest value, 0.42 from Ringhals 1982, means that the actual release was 42% of the limit value of 0.1 mSv/year. The total annual dose to persons living near the plant was thus about 4% of that obtained from the natural background radiation.

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7

Safety Principles

The prime purpose of reactor safety is to minimize the release of radioactive substances. As shown in the previous chapter, the releases during normal operation are kept well below prescribed levels. Normal operation therefore does not imply any hazards to the environment and the general public. The important safety issue is the risk of accidents with potentially large releases. The probability of large releases must be so low that the risk of harm to the public is negligibly small.

The basic approach to safety is to specify criteria for radiation doses and accident probabilities, and then to design, construct and operate the power station so that the criteria are met. In this chapter the main aspects of the safety design process are described, including the specification of radiological criteria, the principles of safety design and safe operation, and the administration of safety.

7.1 Radiological Criteria

The radiological criteria are dose-related and have the character of either dose limits or action limits. Dose limits are specified for normal operation (cf 6.6.1) and for accident conditions. Action limits apply to uncontrolled releases in severe accident situations. Criteria for accident conditions may also be probability-related or source-related.

7.1.1 *Dose-related criteria*

Historically, dose-related criteria for accident conditions were first applied in the Reactor Site Criteria, validated in the USA in 1962 (701). These criteria use the concepts of “exclusion area”, “low-population zone”, and “population centre distance”. The exclusion area is the area surrounding the site where permanent residence is normally not permitted. The low population zone is the area immediately outside the exclusion area, where appropriate safety measures can be adopted if an accident should occur.

In order to determine the size of the zones, a Maximum Credible Accident (MCA) within the design basis is postulated. The MCA involves the release of gaseous and volatile fission products from the core to the reactor contain-

ment. The containment is assumed to leak at a rate corresponding to the highest permissible value according to the design specifications. The atmospheric dispersion of the radioactive substances is calculated using the relevant meteorological conditions at the site. For the purpose of analysis, the following dose-related criteria are applied:

- (a) an individual located at the boundary of the exclusion area for 2 hours immediately after the accident would not receive a total radiation dose to the whole body in excess of 25 rem (250 mSv, see 6.1.3) and a total radiation dose in excess of 300 rem (3 Sv) to the thyroid from iodine exposure;
- (b) an individual located at the outer boundary of the low population zone for an indefinite period of time would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure;
- (c) a population centre distance of at least 1.3 times the distance from the reactor to the outer boundary of the low population zone. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations.

The application of criterion (a) generally results in an exclusion area with a radius of 1–2 kilometres. By various means of improving safety, it has not been necessary to increase the size of the area in spite of a substantial increase in power output since the criteria were formulated.

The siting policy of the Swedish safety authorities has been largely based on the U.S. criteria. The Swedish nuclear power plants are sited in areas where there is a very limited population within 2 km of the plants.

7.1.2 Risk-related criteria

Since 1975, when the Reactor Safety Study was published in the USA, a probabilistic approach to safety criteria for accident conditions has gained widespread support. In 1986 the U.S. Nuclear Regulatory Commission adopted safety goals for the operation of nuclear power plants (702). Two *qualitative goals* were established as follows:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following *quantitative objectives* are to be used in determining the achievement of the above goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

In applying these objectives, the “vicinity of a nuclear power plant” is defined as the area within 1 mile of the nuclear power plant site boundary. The “area near a nuclear power plant” for determining the population risk is defined as the area within 10 miles of the plant site.

In addition, a general performance guideline is proposed to the effect that the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 (10^{-6}) per year of reactor operation. What constitutes a large release is not explicitly defined.

Risk-related criteria have not yet been generally adopted in the regulatory process. A case where probabilistic criteria were used in the assessment of safety is that of the Sizewell B pressurized water reactor plant in the United Kingdom. In this case the criteria are expressed as follows (703):

- (a) For any single accident which could give rise to a large uncontrolled release, the frequency of occurrence should be less than 10^{-7} per reactor year.
- (b) The total frequency of all accidents leading to uncontrolled releases should be less than 10^{-6} per reactor year.
- (c) The predicted frequency of accidents from which radiation doses equivalent to the “emergency reference level” could be expected should not exceed 10^{-4} per reactor year.

The emergency reference level is an example of an action limit, e.g. 100 mSv whole-body dose, below which countermeasures such as evacuation of people are unlikely to be justified, because the risks associated with the countermeasures may exceed the radiological hazard.

7.1.3 Source-related criteria

Another approach to establishing criteria for accident conditions is to specify a limit for the amount of radioactive substances released, regardless

of the expected accident frequency. For this to make sense, certain low-frequency events with potentially large releases must be deemed practically impossible.

An example of this approach is the criterion adopted in Sweden in 1986 that the release of radioactive substances should not exceed 0.1% of the core inventory, excluding noble gases, for a severe accident in an 1800 MWth reactor (704). If this criterion is fulfilled, it is expected that no early fatalities and no intolerable land contamination will occur.

7.2 Safety Design

The approach to safety design is generally based on a philosophy known as defence-in-depth and the application of design criteria and guidelines as well as stringent standards of quality assurance. This section begins with a review of some basic concepts and safety requirements.

7.2.1 Basic principles

A reactor plant consists of a large number of interrelated systems and components. The very complexity of the plant makes it difficult to completely envisage all the possible combinations of faults and events which can jeopardize the safety of the plant. The best approach is to use natural safety characteristics in the design process, i.e. to rely on *inherent safety* as far as possible. For example, an intrinsic characteristic of light water reactors is that the nuclear chain reaction ceases if the moderator density decreases. Thus, the reactor power will automatically decrease if the temperature of the primary coolant or the void content of the core increases. Similarly, the power decreases if the fuel temperature increases.

Equipment can fail if materials and components do not fulfil the design specifications. This may be due to the variation of material properties or the presence of defects. In order to avoid equipment failure, safety-related components and systems must be designed in accordance with proven technology and with sufficient *safety margins*. For example, there is a long tradition of designing pressurized components and systems which has resulted in the establishment of generally accepted codes and standards. Similarly, for core design, nominal data for heat rates and mechanical stresses are chosen so that temperatures and strains are well below critical values.

Buildings and heavy equipment are generally designed according to the *safe-life* principle, i.e. with sufficient margin to last for the entire lifetime of the plant. Certain electrical and mechanical components may have a more limited lifetime. If such components are a part of essential safety-related equipment, they are designed according to the *fail-safe* principle. This means that any malfunction should result in a safe plant condition. For

example, a malfunction of reactor control instrumentation would lead to automatic reactor shutdown.

The safety of a reactor plant depends on the maintenance of a high and uniform level of quality of materials, components and systems during all stages of design, manufacture, construction, operation and maintenance. Consequently, there are special administrative systems for *quality assurance*, which are applied by suppliers as well as utilities. An important task for the safety authorities is to ensure that the quality assurance systems are adequate. In general, safety-related equipment must be accessible for inspection, testing, service and maintenance, and must be repairable whenever necessary.

In spite of detailed specifications and control, the likelihood of faults and abnormal conditions occurring during operation must be taken into consideration. Minor disturbances are controlled by the ordinary operating and control systems without necessitating reactor shutdown. Special *safety systems* are provided for counteracting major disturbances. The safety systems are engineered safeguards for preventing disturbances from developing into accidents. The safety systems include:

- protection systems, which monitor the reactor processes and initiate counter-measures;
- shutdown systems, which rapidly reduce reactor power when necessary; emergency core cooling systems, which cool the core when normal cooling is inadequate.

Safety systems can be *passive* in the sense that their function does not depend on components changing their state, e.g. the opening or closing of a valve. Examples of passive functions are the insertion of PWR control rods by gravity, the natural circulation of the coolant which removes residual heat in the shutdown reactor, and the steam condensation in the BWR containment pool. Conversely, the systems are said to be *active* if they need an electric signal for actuation and power for operation. An active system may fail if, for example, the power supply to electrically powered pumps is not available.

In order to increase the availability of the safety systems, the principle of *redundancy* is applied, i.e. the systems are duplicated or multiplied. Single component failures are thereby prevented from causing total system failure. For example, the emergency core cooling system consists of several subsystems which function independently of each other, and each subsystem (in duplicated systems) has sufficient capacity to perform the particular function alone.

Another design principle for improving safety is *diversification*. This means that a particular safety function can be performed by two or more systems based on different physical modes of action, thereby reducing the

possibility of systematic failures. For example, reactor shutdown can be achieved by the insertion of control rods or by the injection of boron into the core. The control rods in Swedish BWRs can be inserted by a hydraulic system (scram) or by an electrically powered screw mechanism.

The probability that a spurious failure will lead to the failure of a safety function can be made very small by redundancy and diversification. Instead, the probability of *common cause failure* can become relatively large in redundant, non-diversified systems. A common cause failure may arise from deficient design or manufacture, from environmental effects (high temperature, humidity, etc) or from external events such as fire and flooding.

The probability of common cause failures can be minimized and, in certain cases, practically eliminated by appropriate system design and adequate control measures. The physical segregation of redundant systems in different areas of the plant protects against the effects of adverse environmental conditions and external events. Diversification reduces the influence of design and manufacturing deficiencies. Human error can also result in common cause failure, for example, through erroneous instrument calibration.

An important way of achieving a high level of safety in complex systems is to systematically register, process and analyse abnormal events, in other words, to *learn from experience*. Safety can then be improved by modifying systems and procedures in order to prevent the recurrence of these events. The systematic feedback of operating experience has been instrumental in attaining a high level of safety in the aviation industry.

Experience has shown that technical equipment in itself can be made very safe. On the other hand, *human error* has proved to be a dominant factor in causing system malfunction. Human error can affect safety during all stages of plant design, construction, operation and maintenance. For example, the reactor operator may act hastily in the stressful situation which arises during an abnormal event. He may neglect to initiate the required safety functions or may adopt the wrong countermeasures. On the other hand, correct action in an unforeseen situation can be crucial to safety.

The control room design has been shown to play an important role in the detection of disturbances, the establishment of causes and the adoption of countermeasures by operating staff. *Man-machine* interaction is facilitated by a suitable presentation of essential plant variables, and by an ergonomic layout of control boards and instrument panels. The analysis of human error is very complex and involves technical, medical and psychological aspects.

In order to minimize the risk of human error, the *automation* of important safety features is implemented, especially of those features requiring prompt action. For example, in the operation of Swedish boiling water reactors the “30-minutes rule” is applied. This means that all measures which are necessary within 30 minutes after an event which might lead to

significant releases must be carried out automatically. This allows the operator some time for diagnosis and decision upon further action.

Even with a high degree of automation, the control room crew will always play an important role in the safe operation of the reactor, especially in connection with changes in the operating conditions such as during start-up and shutdown. The *training* of personnel is therefore very important to safety. Adequate instructions and well-practised procedures are essential prerequisites. However, written instructions cannot cover all upcoming situations. A good understanding of the basic processes is therefore necessary to enable the reactor operator to act independently and correctly in an unforeseen situation.

The importance of man to reactor safety is not only limited to the role of the individual, but also includes *attitudes* to safety as well as administrative and organizational conditions. In safety work, there must be a constant awareness of the fact that severe accidents can occur, even if the likelihood is minimal. The administration of safety work must be based on clearly defined regulations and responsibilities. On the other hand, the regulatory system should not be so detailed as to stifle personal initiative for safety improvement.

7.2.2 Fission product barriers

Most of the radionuclides formed during operation are retained in the fuel in the reactor core. A small amount is present in the spent fuel stored in pools in the reactor plant. An even smaller amount is found in the resins of the clean-up systems and in the waste management systems. The radionuclides in the core are prevented from being released by several barriers:

- the structure of the fuel material,
- the cladding of the fuel rod,
- the pressure boundary of the primary system,
- the leaktight shell of the reactor containment,
- the reactor building (of the boiling water reactor).

A large release to the environment can only result if all the barriers are penetrated.

A necessary condition for a large radioactive release is that most of the fuel be overheated. The fuel can overheat if there is imbalance between the heat supplied and the heat removed. This can occur if the reactivity and thus the nuclear power increases in an uncontrolled manner. Imbalance also results if the coolant flow through the core is insufficient to remove the heat. The fuel can also overheat after reactor shutdown if the decay heat removal is inadequate.

If the cladding is damaged by overheating or otherwise, radioactive sub-

stances will escape into the coolant. As long as the primary system boundary remains intact, no uncontrolled releases will take place. In order to prevent overpressure, the primary system is equipped with safety valves. In the event of a pipe break or a large leak leading to loss of primary coolant, water is supplied from reserve systems to maintain core cooling.

In a loss of coolant accident, radioactive substances will be released with escaping hot water and steam to the reactor containment. In the boiling water reactor, the steam is discharged to the containment water pool where it condenses, thereby limiting the pressure increase in the containment. At the same time, the radioactive substances are effectively removed. In the pressurized water reactor (like in the boiling water reactor), the atmosphere of the containment can be sprayed with water from spray nozzles in the containment roof. This results in a decrease of pressure and temperature and the removal of radioactive substances from the containment atmosphere. If the integrity of the reactor containment is preserved, no large releases to the environment can occur.

7.2.3 Defence-in-depth

The basic safety requirements of keeping the fission product barriers intact are embodied in the defence-in-depth principle. This principle provides guidelines for safety design and safe operation on three levels, which partly overlap (Fig. 7.1).

Level	Measures	Examples of systems and principles
I	Preventive	Normal operating and control systems Inherently stable design features Adequate safety margins Quality assurance
II	Protective	Safety systems Redundancy Diversification Physical segregation
III	Mitigative	Reactor containment Activity removal systems Remote siting Emergency preparedness

FIG. 7.1. The defence-in-depth principle

The *first level* implies that the reactor should be designed and operated for maximum safety during normal operation. Radioactive releases should be kept as low as reasonably practical (cf 6.6.2). Disturbances of normal operation should be tolerated without exceeding the prescribed discharge limits. Safety efforts focus on the *prevention* of accidents by:

- utilizing the inherent safety characteristics in the reactor design;
- designing and operating the reactor with adequate margins to critical values of material properties and state variables;
- designing components and systems for the monitoring and control of reactor operation according to the fail-safe principle;
- ensuring a high and uniform level of quality for materials and equipment important for safety;
- carrying out recurrent surveillance, inspection and functional testing of safety-related plant components.

The *second level* presupposes that incidents and accidents will occur in spite of the preventive measures. Systems for *protection* against accidents should therefore be provided to counteract and prevent abnormal events from developing into accidents.

The *third level* is based on the fact that accidents can occur in spite of the measures taken to prevent and counteract them. Systems for the *mitigation* of accident consequences should therefore be provided to minimize releases to the environment and doses to the general public.

The design of the safety systems is based upon the analysis of postulated abnormal events, called *design basis accidents* (DBA). These represent certain limiting conditions which it should be possible to overcome without excessive consequences to the environment. Criteria for the design basis accidents are usually specified by the licensing authorities. The licensee, which is normally the owner and operator of the plant, will have to show by analysis that the criteria are met.

7.2.4 Design criteria

Established standards for the protection of the public in the design of buildings, pressure vessels, electrical equipment, etc., have existed for a long time. Relevant parts of these standards are also applicable to reactor plants. In addition, there are special rules and regulations for the construction and operation of reactor plants. Although the legal status and scope of these regulations differ from country to country, the content is generally based on the criteria and guidelines established in the USA during the late 1960s in accordance with the defence-in-depth principle. These criteria have played an important role in light water reactor design and safety worldwide.

The regulations include *General Design Criteria* (GDC) which have the status of law in the USA. The basic safety requirements are expressed qualitatively. No distinction is made between boiling water reactors and pressurized water reactors. The some fifty criteria that have so far been established are divided into six groups (Fig. 7.2). The groups reflect the three levels of the defence-in-depth principle and determine the design and operating requirements for safety-related equipment.

Group	Number of criteria	Content
I	5	<i>Overall requirements</i> for quality assurance and protection against external events.
II	10	<i>Protection by multiple fission product barriers</i> with requirements for inherent safety, safety margins, instrumentation and control.
III	10	<i>Protection and reactivity control systems</i> with requirements on functions and capacity, redundancy and diversification, reliability and testability.
IV	17	<i>Fluid systems.</i> Regulations on quality, fracture prevention, and inspection of the reactor coolant pressure boundary. Requirements on systems for reactor coolant make-up, residual heat removal, emergency core cooling, containment sprinkling and cooling to ultimate heat sink.
V	8	<i>Reactor containment.</i> Design basis and requirements on leaktightness, penetrations, isolation and testing
VI	5	<i>Fuel and radioactivity control.</i> Requirements on radiological protection and radioactivity control during fuel handling and waste management, and monitoring of radioactivity releases.

FIG. 7.2 General design criteria

The character of the General Design Criteria is best illustrated by way of example:

GDC 34—*Residual heat removal*

“A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

As a general rule, the malfunction of one component or subsystem, should not jeopardize the particular safety function. This *single failure criterion* means that safety-related components and systems should at least be duplicated (redundancy) or that the particular safety function should be achieved by alternative systems of different design (diversification).

The Nuclear Regulatory Commission (NRC) also issues *Regulatory*

Guides (RG). These guides contain recommendations and guidelines which serve to identify safety issues and establish principles and specifications which, if they are fulfilled, would constitute acceptable solutions for the safety authority.

The Regulatory Guides fall into ten divisions, the first of which deals with power reactors. More than 100 titles have so far been issued. Most of the guides concern quality requirements and quality control. For example, RG 1.26 is a classification of systems and components into four quality classes with associated standards. This classification forms the basis of establishing quality requirements for safety-related equipment.

In Sweden, no general safety regulations have been established. A code of practice has been successively developed which is reflected in the licensing conditions for the reactor plants. The USNRC design criteria are applied with certain modifications. Suitable parts of the Regulatory Guides are also used, for example, the above-mentioned division into quality classes with certain modifications (705).

The quality requirements are related to the safety importance of the equipment. Therefore, all plant structures, systems and components are assigned to *safety classes* as follows:

- Class 1 Systems and system parts directly pressurized from the reactor within the containment.
- Class 2 Systems and system parts required for safe reactor shutdown, emergency core cooling, residual heat removal, containment function, and spent fuel storage.
- Class 3 Support systems for Class 2 systems, and systems for radioactive waste management and spent fuel cooling.
- Class 4 Structures, systems and components which have no direct safety function but which may be connected to or influenced by equipment in Class 1–3.

Among specific Swedish safety requirements is the previously mentioned 30-minutes rule.

Another example is the pressure-relief requirements for BWR pressure vessels. The capacity of the safety valves must be sufficient to prevent over-pressure even if the scram system fails.

An area in which Swedish practice is rather extensive concerns fire protection and the segregation of safety-related equipment. Certain weaknesses in the auxiliary electrical supply were observed and rectified at an early stage in the design of the first Swedish boiling water reactor, Oskarshamn I. Since then, the consistent separation of electrical equipment and control systems has been applied in all Swedish plants.

Essential safety-related equipment in the latest Swedish boiling water reactors is divided into four subsystems with 50% capacity, belonging to

separate trains and usually located in separate fire cells. The “ N minus 2” criterion is applied, which means that of N redundant subsystems, the designer must assume that one fails and one is out of order due to repair or maintenance, without jeopardizing the safety function of the total system.

7.2.5 Quality assurance

A high and uniform quality of materials, components and systems is necessary, not only for safety but also for plant availability and maintenance costs. It is required of the plant owner and licensee to maintain a high level of quality during all stages of plant construction and operation. The administrative control and planning of the necessary measures is known as Quality Assurance (QA).

Quality assurance means ensuring that:

- the design fulfils specified quality requirements;
- the manufacture and assembly are conducted according to the design specifications;
- testing is carried out to verify that the specifications have been met;
- the plant is operated and maintained according to the prescribed rules.

Special programmes for quality assurance were originally enforced by experience in the USA, where several contractors and a large number of sub-contractors are usually involved in a reactor project. This places stringent requirements on project coordination and control so that the specified component quality is attained, particularly in conventional components. As a result, regulations concerning QA programmes were included as an important part of the General Design Criteria.

In Sweden, the situation is less complex. Therefore, there was no urgent need for implementing QA programmes according to the U.S. model. Nevertheless, the principles were applied and a code of practice was subsequently established and formalized by the Nuclear Power Inspectorate. The quality assurance system is applied by both utilities and suppliers.

The control of Class 1 components and systems, is particularly important. Testing procedures include official testing by the Swedish Plant Inspectorate and control at the responsibility of the supplier and owner. The testing organization reviews guidelines and calculations for the manufacture, controls the manufacturing process, inspects components prior to their commissioning and subsequently at regular intervals of 1 or 2 years.

7.3 Safety During Operation

Safe operation means that adequate margins to bounding values of essential plant variables are maintained during normal operation as well as during

fault conditions. The overriding requirement is that radioactive releases to the environment are kept within prescribed limits.

7.3.1 Control and instrumentation

The plant conditions are continuously monitored. The main parameters to be monitored are the neutron flux in the core, the temperature and pressure in the reactor system and containment, the mass flow in the main coolant and feedwater systems, and the water level in the reactor pressure vessel (BWR) and steam generators (PWR). The neutron flux directly indicates the power level. Its rate of change is a measure of the reactivity which is particularly important to control during start-up.

Safety is assured by automatic protection systems which act on the detection of abnormal states. The basic control and instrumentation concept has three functional levels—control, alarm and trip—forming a layered protection system with step-raised actuation set points (Fig. 7.3). High reliability is ensured by redundant design.

Information of the plant status is presented in the control room. Extensive use is made of mimic diagrams for representing the reactor core and process systems with dedicated alarm annunciators arranged together on the same boards and panels in the control room. Computer-aided systems are used for handling the large quantities of data and for controlling data logging and data display equipment.

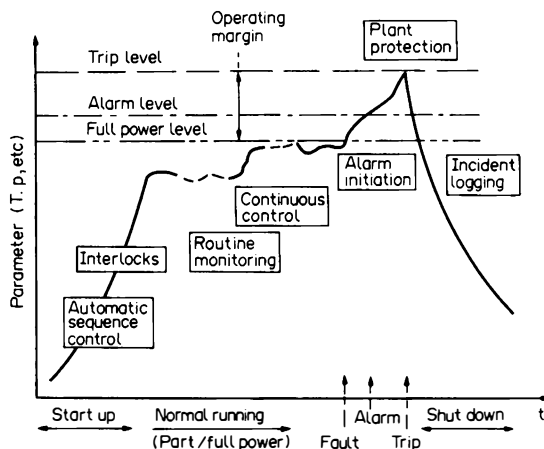


FIG. 7.3. Control and instrumentation functions. Adapted from M. W. Jervis, On-Line Computers in Nuclear Power Plants, *Advances Nucl. Sci. Technol.*, Vol. 11, 1979.

7.3.2 Operating rules

The control and protection systems operate automatically. The role of the reactor operator is mainly to watch over the automatic systems and to put into effect the desired changes of plant states. The manual control actions do not require rapid response by the operator.

Potential errors in the execution of these actions are guarded against by the automatic protection systems and interlock arrangements.

Operating rules are formulated to guide the operator in maintaining plant operation within the limitations imposed by the design specifications and safety considerations. Safety-related equipment is subject to periodic testing and preventive maintenance. Feedback of operating experience (see section 13.6) and recurrent staff training are also important means of maintaining a high level of safety.

Swedish utilities have jointly prepared and the Nuclear Power Inspectorate has approved of Technical Specifications for the Operation of Nuclear Power Plants. They represent a framework of operating rules and guidelines for assuring safety during operation, allowing a certain flexibility for the operator to achieve optimum plant conditions, notably a high plant availability. The Technical Specifications include:

- Bounding values for essential safety-related parameters. If the bounding limits are exceeded, a special investigation and report to the safety authorities is required before operation is resumed.
- Conditions for plant operation with regard to the functional preparedness of standby systems and components. If the conditions cannot be fulfilled, restrictions of operation are imposed and restoring measures required in each particular case.
- Type and frequency of testing and inspection of components and systems. If the prescribed testing is not carried out or if negative results are obtained, the component or system is considered to be out of order resulting in restrictions of operation.
- Rules to be followed during normal operation as well as in abnormal situations and during maintenance work. Requirements on the documentation and reporting of operational events and design modifications.

The operating rules are continuously updated to take into account new experience and plant modifications. A general rule is included which stipulates that the plant should be retained in or brought to a safe condition in any unclear situation which cannot be immediately diagnosed.

Detailed plant operation and maintenance activities are governed by written instructions for procedures such as:

- plant start-up and shutdown,
- power and test operation,

- core operation and monitoring,
- shift turnover and plant status reporting,
- service and maintenance.

A *duty engineer* is always in service at each plant for advising the control room crew on safety matters. The duty engineer takes on special responsibilities in case of emergency.

7.3.3 Accident management

The operating rules include instructions for plant operation during accidents within the design basis. The procedures are trained and retrained on full-scale plant simulators. The operating rules for accidents within design are traditionally event-oriented. After the Three Mile Island accident, guiding instructions were developed also for severe accidents beyond design. These *emergency operations procedures* tend to be symptom-oriented rather than event-oriented, the objective being to meet the basic safety requirements:

- secure sufficient subcriticality,
- maintain adequate core cooling,
- minimize radioactive releases.

The fulfilment of the safety objectives is supervised by continuously monitoring significant plant parameters during the accident. A visual synthesis of the plant status is displayed in the control room without regard to the origin of the particular problem or the detailed sequence of events. The overall strategy of severe accident management is to maintain the long-term integrity of the reactor containment.

A special organization is established for activities within the plant in emergency situations. The duty engineer must contact regional and central authorities while the emergency organization is being set up. Based on the experience from TMI-2, a *technical support centre* will be established at the plant as part of the emergency organization, in which work related to the accident can be performed without disturbing the activities in the central control room.

7.4 Safety Administration

In this section the administrative policies and organizational practices for ensuring safety in the design, construction and operation of nuclear power plants are discussed. The principles are illustrated by the conditions in Sweden.

7.4.1 Roles and responsibilities

Nuclear energy activities at large are regulated by laws, the prime objective being to minimize the risk of harm to the general public and the environment. The authorities issue safety regulations and ensure that they are complied with. The scope of the legislation and the focus of the regulatory activities differ considerably from country to country. The situation in the USA and the UK can be taken as an example.

In the USA, the Nuclear Regulatory Commission (NRC) has established a comprehensive system of rules and regulations which have the status of law. Substantial resources for enforcement and supervision have been set up. There are about 1600 electric utilities of which more than 100 operate nuclear power plants. This requires standardized and detailed safety rules and a large regulatory organization.

In the UK, there are only two nuclear utilities, the largest of which, the Central Electricity Generating Board (CEGB), has its own resources for safety work. Therefore, the detailed regulation of reactor safety activities is not considered necessary. Instead, the prime and sole responsibility of the utility for the safety of the plant is emphasized. The Nuclear Installations Inspectorate has a supervisory rather than a regulatory role.

The situation in Sweden is similar to that of the UK. No extensive regulatory framework has been set up. The direct responsibility for reactor safety rests with the licensee. The function of the supervisory bodies is to set goals for the safety work of the utilities and to evaluate their organization and procedures as well as their ability to achieve the goals. The importance of an open dialogue between the utilities and the authorities is emphasized.

7.4.2 Safety authorities

According to the Nuclear Energy Act in Sweden, permission by the Government is required for the construction, loading of fuel, and operation of nuclear power plants. The *Swedish Nuclear Power Inspectorate* (SKI) acts as the supervisory agency. The SKI formulates the requirements for the ownership, construction and operation of nuclear power plants. This involves:

- establishing safety regulations,
- evaluating safety analysis reports,
- supervising the compliance with the regulations,
- initiating safety research and development.

The SKI has two technical offices (Fig. 7.4). The Office of Inspection is responsible for ensuring that plants are constructed, tested, operated and maintained in accordance with the established regulations. The Office of

Department of Industry			
Nuclear Power Inspectorate (staff about 85)			
Board (Director General and 6 Members)			
Office of Inspection and Enforcement (33)	Office of Regulation and Research (36)	Department of Administration (11)	Information Secretariat
Barsebäck Forsmark Oskarshamn Ringhals Nuclear Materials	Safety Review Safety Analysis Safety Research Nuclear Waste		
Advisory Committees to the Board			
Safety Criteria and Reactor Safety Safeguards Safety Research and Development			

FIG. 7.4. Overview of the Swedish Nuclear Power Inspectorate organization (1984)

Regulation and Research handles licensing matters and prescribes the conditions for construction and operation permits. It also identifies and investigates new safety issues and initiates measures for improving safety, including safety research. The activities are governed by a board comprising the director general and members appointed by the Government.

There are three advisory committees to the SKI Board, which deal with reactor safety in general, safeguards and research. In addition, there is an advisory group to the Office of Regulation and Research, comprising members from the SKI and the utilities, which proposes measures for improving safety and recommends lines of action.

The activities of the SKI have gradually changed over the years, partly because nuclear power plant construction has passed its peak in Sweden. Present activities are mainly directed to supervising the existing plants and reviewing their safety.

Under the Radiation Protection Act, the *National Institute of Radiation Protection* (SSI) formulates regulations and supervises their application. However, no permission according to the Radiation Protection Act is required for activities covered by the Nuclear Energy Act (cf 2.2). In addition to acting as a central supervisory agency for radiation protection, the SSI is responsible for:

—acquiring detailed knowledge of the risks associated with radiation and

- following developments within the sciences of radiobiology and radiation physics;
- coordinating emergency preparedness planning and thereby acting as an advisory body to the county administrations;
 - maintaining a central coordinating responsibility for applied research in radiation protection.

Radiation protection matters within the nuclear power field are managed by the SSI's Nuclear Energy Department. Advisory bodies on radiation protection research and on emergency preparedness are linked to the Board of the SSI.

7.4.3 Licensing procedures

As part of the application to construct a nuclear power plant, the applicant submits a *Preliminary Safety Analysis Report* (PSAR) to the licensing authority. The PSAR contains a detailed description of the site and surroundings, of the plant design and plant performance as well as of the safety policy for the particular plant design. A typical table of contents of a PSAR is shown in Fig. 7.5.

In the PSAR, particular attention is paid to the description of the engineered safety features and the analysis of design basis accidents. The analysis is carried out on the assumption that the safety systems will function as intended and with due regard to insufficiently known phenomena so as to obtain results on the safe side. The impact on the environment of a Maximum Credible Accident (cf 7.1.1) must be shown to be acceptable. The licensing agency evaluates the PSAR and comments are invited from the appropriate authorities. The licensing agency evaluates whether the

1	Introduction and general plant description
2	Site characteristics
3	Design criteria
4	Reactor and reactor coolant system
5	Reactor containment
6	Safety systems
7	Instrumentation and controls
8	Electric power
9	Auxiliary systems
10	Steam and power conversion system
11	Radioactive waste management
12	Radiation protection
13	Conduct of operations
14	Initial tests and operation
15	Accident analysis
16	Quality assurance

FIG. 7.5. Typical content of a PSAR

plant meets the safety requirements and recommends that a construction permit be granted, provided the required conditions are fulfilled. The construction permit is a cabinet decision.

During construction, the licensee prepares a *Final Safety Analysis Report* (FSAR). This report contains a detailed description of how the plant will be operated in order to satisfy the safety requirements. It also describes the operating organization and the quality assurance programme set up by the licensee. The report is submitted to the authorities for evaluation. If the safety requirements are met, the licensing authority approves the final plant design.

When plant construction is in its final stages, components and systems are tested. Prior to fuel loading, a series of *pre-criticality tests* are conducted partly with cold systems, and partly up to full pressure and temperature to check the performance of the different systems and their interaction. Before fuel is admitted into the plant, permission must be obtained from the authorities. In Sweden, permission for fuel loading is obtained from the Government.

After fuel loading, the nuclear tests can begin. They mainly consist of quality tests and measurements at low power. The power is successively raised and tests carried out on the reactor systems as well as with the reactor and turbine together. Once the tests have been completed with satisfactory results, the authorities can grant permission for normal operation at full power.

During *normal operation*, regular reports are submitted to the authorities. The operating conditions and plant output are reported daily. Reports on radiation exposure and activity monitoring in and around the plant are submitted to the supervisory authority every month. In addition, reports are submitted on a non-routine basis of events which are of importance to safety. If discharge limits are exceeded, or if any abnormal occupational exposure occurs, this is communicated to the radiation protection authority.

As part of the *recurrent safety review* of the plant, the Swedish Nuclear Power Inspectorate conducts a systematic evaluation of the safety of each unit every 8–10 years. This report, which is submitted to the Government, is called ASAR (As-built Safety Analysis Report). Basic information for ASAR is compiled by the licensee in consultation with the Inspectorate. ASAR contains a review of the safety management and organization, operating experience, quality issues, safety studies, training, and completed, ongoing and planned safety improvements in the plant. The essence of ASAR is the systematic reliability analysis of plant components and systems, so that dominant contributions to the core damage frequency can be identified as a basis for selecting measures for safety improvement.

7.4.4 Emergency preparedness

The responsibility for emergency planning *within* the nuclear power plant rests with the licensee. Requirements for emergency preparedness are established in the licensing process. The emergency plan includes instructions and rules for accident management and involves the establishment of an emergency organization which replaces the ordinary operative organization.

Emergency preparedness *outside* the plant is regulated by special ordinances. Guidelines were established by the Swedish Government in 1981 (706). The main responsibility for the safety of the general public lies with the pertinent county administration. The emergency plans of the licensee and the county are coordinated and tested at annual emergency preparedness exercises, where the central agencies are also represented.

In principle, the emergency plans shall take into consideration all kinds of accidents, from those with negligible environmental impact to very large accidents. As a guide for emergency planning, the region around the nuclear power stations is divided in zones (Fig. 7.6). Within the *central alarm zone* reaching 5–10 km from the plant, warning can rapidly be given to the population outdoors and indoors. Within an area of about 12–15 km from the plant, known as the *inner emergency zone*, it should be possible to execute a detailed plan of action, e.g. for quick evacuation. In this zone, iodine

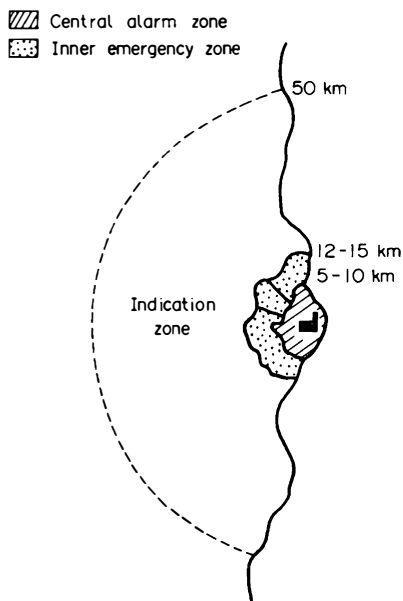


FIG. 7.6. The emergency zones around Swedish nuclear power plants

tablets and advance information are distributed to the households, and there is also a network of fixed measuring points. In the *indication zone* reaching about 50 km from the plant, there are predetermined loops for mobile measurements to be performed by special patrols.

7.4.5 Local safety committees

The supervisory agencies are charged with informing the general public about reactor safety and radiation protection. In order to further improve the quality of information, a local safety committee is appointed at every nuclear power plant. The committee shall find out and inform the general public of completed or planned safety activities. The plant owner is responsible for submitting the required information and granting access to the plant at the committee's request. The committee members are appointed by the Government on the basis of proposals from the pertinent municipality.

7.4.6 Nuclear utilities

In Sweden, the owners of nuclear power plants are Forsmarks Kraftgrupp AB, OKG AB, Vattenfall (the Swedish State Power Board), and Sydkraft AB. The reactors at the Forsmark nuclear power station are operated by Vattenfall who are also responsible for the safety of the plant.

Each utility has a special *safety department* to watch over safety issues. The task of this department includes:

- handling licensing matters;
- ensuring that plant construction is carried out according to established safety requirements;
- preparing technical specifications for reactor operation and supervising their enforcement;
- initiating and managing investigations for reactor safety evaluation.

Each utility has a *central safety committee* which examines all events occurring in the plants of importance to safety. The safety committee reports directly to the top management. The committee has a fixed membership and its activities are carried out in accordance with special instructions. Minutes are taken for each meeting and submitted to the Nuclear Power Inspectorate, thereby becoming public documents according to Swedish law. The safety committees of the utilities co-operate closely.

Each nuclear power station has a *training programme* to provide basic courses and plant-specific training for operating staff as well as special courses for technical support and maintenance personnel. The utilities co-operate at the Nuclear Training and Safety Centre (KSU) at Studsvik. KSU has three full-scale simulators of boiling water reactors and one of a pressu-

rized water reactor. Although no formal examination of reactor operators is required in Sweden, the SKI continually evaluates the training through its competence follow-up system.

The utilities also cooperate within the KSU in compiling, processing and evaluating safety-related events and by providing feedback of experience to the plants. KSU is also engaged in research projects of common interest to the utilities and in public information activities.

7.4.7 Reactor vendors

The reactor vendors play an important role in reactor safety, for example by the development of more efficient safety systems. Vendors perform detailed safety analyses in the design process of contracted plants. Their resources are also utilized by the utilities for service and maintenance work of importance to safety. In Sweden, contacts are facilitated by the fact that there is only one reactor vendor, who is not only responsible for the nuclear steam supply system but also for the plant layout and construction work as well as the specifications for the turbine-generator and other plant components. Thus consistent safety design requirements are specified for the entire plant.

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8

Safety Systems

During normal operation, the basic safety requirements are met by the reactor's ordinary operating systems. During fault conditions, the reactor protection system ensures that automatic shutdown takes place and that the required countermeasures are initiated. In certain cases, the normal operating systems may be insufficient to keep the core well cooled. Emergency cooling systems are then put into operation. The reactor protection, shutdown and emergency cooling systems are commonly known as safety systems. A strict division into operating systems and safety systems cannot be made, however, since both types may have both operating and safety functions. The normal operating systems were described in Chapters 4 and 5. This chapter describes the main safety systems in the boiling water reactor and pressurized water reactor.

8.1 Boiling Water Reactors

The following description applies to boiling water reactors of the Forsmark 3 type. Section 8.1.9 reviews some plant-specific characteristics of other Swedish boiling water reactors.

8.1.1 Reactor protection system

The reactor protection system is designed to initiate measures for preventing fuel overheating and for limiting radioactive releases to the environment. The system mainly consists of sensors, signal processing units, logic circuits, and actuators for alarms, reactor shutdown and other engineered safeguards. The system has a layered structure with step-raised actuation set points and priorities. The input signals are obtained from detectors which monitor safety-related plant variables. Signals requiring the same action are grouped into *safety chains*. There are three main safety chains for:

- reactor shutdown* by hydraulic scram (the “scram chain”), or by fine-motion insertion of the control rods (the “screwstop chain”), see 8.1.2;
- reactor isolation* by closure of the reactor containment isolation valves;

—*emergency core cooling* by actuation of the emergency core cooling systems and the automatic depressurization of the primary system.

Each safety chain has four redundant channels. A signal must be developed from at least two of these channels in order to actuate the required system. Due to the “2-of-4” logic, individual channels can be tested during reactor operation without impairing the safety function.

The scram chain is actuated by abnormal values of primary system variables such as reactor power, system pressure, and water level in the reactor vessel. The logic circuits and actuators are operated in the de-energized mode which means that loss of a voltage supply does not prevent actuation of the corresponding channel.

The screwstop chain acts as a backup for the scram chain. It operates in the energized mode, which means the loss of a voltage supply leads to blockage of the corresponding channel. However, due to the “2-of-4” logic, chain actuation is not prevented.

Reactor isolation and emergency core cooling are actuated by parameters which indicate breaks or large leaks in the primary system, such as pressure and temperature in the containment and low water level in the reactor vessel. There are five different types of reactor isolation, depending on the nature of the break or leak and its position inside or outside the reactor containment. Automatic depressurization is initiated when signals are received that the loss of coolant is large enough for potential core uncovering at full reactor pressure.

8.1.2 Shutdown systems

The reactor is rapidly shut down by the *hydraulic scram* system. The control rods are fully inserted within 4–6 seconds. The control rods can also be screwed into the core using electrically driven motors, which is called *fine-motion control rod insertion*. In this way the control rods are inserted into the core within 4 minutes from a fully withdrawn position. When scram is actuated, fine-motion control rod insertion is also initiated. The drive mechanisms and control rods are described in section 4.1.2.

When scram is actuated, the speed of the main recirculation pumps is automatically reduced to a minimum value via signals to the static frequency converters which regulate the pump speed. This *fast pump runback* effectively contributes to safe reactor shutdown. As a result of the reduced recirculation flow, the amount of steam produced in the core increases which decreases the reactivity and immediately stops the nuclear chain reaction. If auxiliary power is lost, the pumps will stop completely and shut down the reactor.

If it is impossible to insert the control rods, the reactor can be shut down by the injection of boric acid solution into the reactor vessel. The *boron*

injection system consists of two independent circuits with piston pumps, tanks of sodium pentaborate solution, valves and pipelines. The boron injection system is initiated manually.

The control rods are arranged in eighteen independent *scram* groups, each comprising eight to ten rods. The reactor can be kept sufficiently subcritical in its most reactive condition even if one of the scram groups fails. At operating temperature it is sufficient if only half of all rods are inserted into the core.

As shown in Fig. 8.1, each of the following conditions is sufficient to achieve reactor shutdown:

- Automatic or manual scram with failure of a maximum worth scram group.
- Automatic speed reduction of the main recirculation pumps *and* screw insertion.
- Automatic speed reduction of the main recirculation pumps *and* manually initiated boron injection.

These conditions are conservative since the reactor can be shut down at operating temperature even if a large number of control rods should fail.

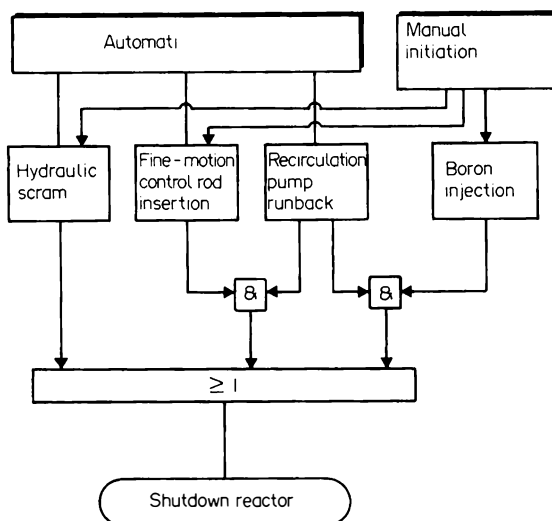


FIG. 8.1. Conditions for reactor shutdown. From Swedish Department of Industry, *Safety Study Forsmark 3*, DSI 1978:3

8.1.3 Pressure relief system

The basic safety function of the pressure relief system is to protect the reactor from overpressure. In certain abnormal situations the system must also be able to rapidly reduce reactor pressure from the normal 7.0 MPa to a low level so that the low-pressure coolant injection system can be used. This function is known as *automatic depressurization*. The pressure relief system is also designed to control the reactor pressure in situations when the turbine condenser is needed but not available to receive steam.

The pressure relief system consists of eight safety valves and eight relief valves with pipelines. The valves are connected to the main steam lines inside the reactor containment and discharge into the condensation pool (Fig. 8.2). In older boiling water reactors, the safety valves discharge directly into the containment drywell.

The safety/relief valves are both power-actuated, either automatically or manually, from the control room, and pressure-operated by means of spring-loaded pilot valves. The spring-set point is such that the valves open

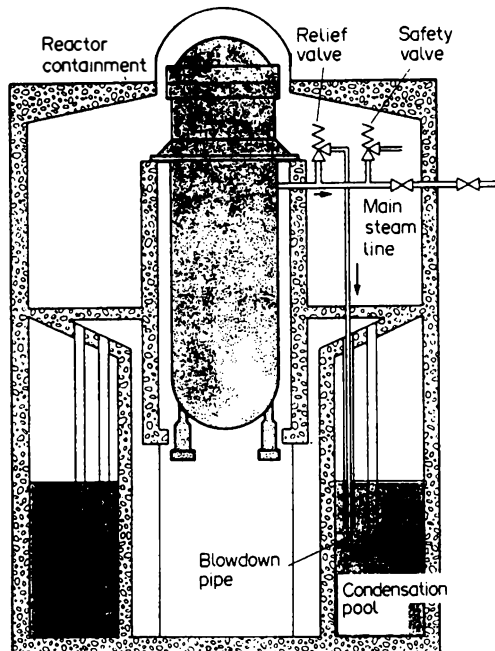


FIG. 8.2. Boiling water reactor pressure relief system schematic. Courtesy Nuclear Training and Safety Centre, Studsvik

at about 8 MPa as compared to the normal system pressure of 7 MPa. All valves can be forced to close by means of block valves in the lines between the main valves and their pilot valves.

The set point pressure for electric opening of the relief valves is 7.4 MPa. The relief valves are also actuated automatically in certain situations involving steam blockage, such as turbine trip with failure of the steam bypass system, and closure of the main steam line isolation valves. The valves remain open for at least 4 seconds, after which closure is actuated as the closure-set point pressure is reached. Failure to close is indicated in the control room.

The safety valves are automatically actuated by electric signal when automatic depressurization is called for. There is no closure signal in this case.

8.1.4 Condensation system

The condensation system consists of the wetwell of the reactor containment (Fig. 4.7), the lower part of which comprises the 9 metre deep annular condensation pool. The pressure relief lines from the safety/relief valves discharge into the condensation pool as well as the blowdown lines from the drywell, which extend 5 metres into the pool.

The condensation system receives and condenses the discharged steam. It is designed to be able to receive all the steam escaping into the containment from a large pipe break in the primary system without the pool water becoming too hot. In addition, the condensation pool serves as a water reservoir for certain auxiliary cooling systems.

The condensation pool is cooled by a heat exchanger via diesel-backed cooling circuits to the sea. The temperature of the water in the pool is normally maintained at about 20°C and must not in any event exceed 95°C.

8.1.5 Auxiliary feedwater system

The auxiliary feedwater system is designed to supply the reactor with water if the ordinary feedwater system is unavailable. It will also contribute to protecting the core against overheating in the event of a large loss of coolant accident.

The auxiliary feedwater system consists of four independent loops, each equipped with a piston pump which draws water from the condensation pool. The water is distributed over the reactor core. The four loops are located outside the reactor containment in separate rooms. The system has a capacity of 22.5 kg/s per loop, and water can be supplied at any reactor pressure.

During normal operation, the system is on standby with the pumps shut down and the external isolation valves in the pressure side pipelines closed. Pump-start and in pumping of water occurs in two steps. During pump-start,

the water is pumped around in bypass pipelines outside the containment. If a signal for in-pumping of water is also obtained, the external isolation valves are opened and the valves in the bypass pipelines are closed. In-pumping of water is interrupted on receipt of a signal that the water level in the reactor is high. The safety function of the system is fulfilled by two loops, in accordance with the “N-minus-2” criterion (cf 7.2.4).

8.1.6 Low-pressure injection system

The low-pressure injection system shall, together with the auxiliary feed-water system and the pressure relief system, protect the reactor core from overheating in the event of a primary system pipe break. The system consists of four independent subsystems by which water can be supplied to the reactor at a pressure below about 1.5 MPa. Water is taken from the condensation pool and pumped via two loops to the downcomer and two loops to the core spray nozzles above the core (Fig. 8.3). In the suction line of each circuit, there is a strainer in the condensation pool and a containment

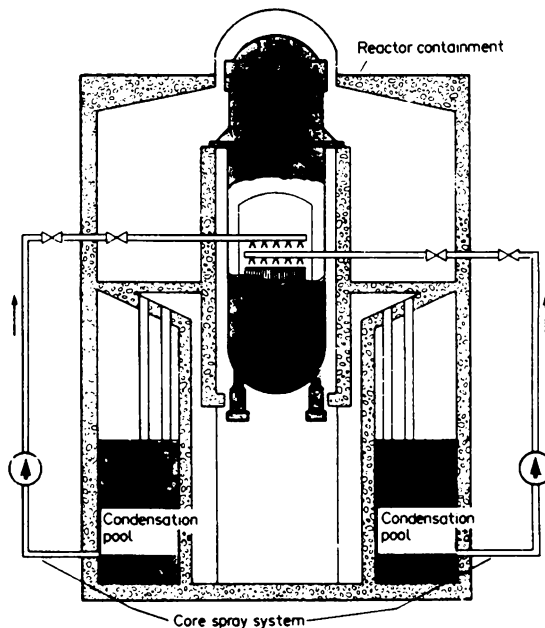


FIG. 8.3. Forsmark 3 low-pressure injection system schematic. Courtesy Nuclear Training and Safety Centre, Studsvik

penetration. The pressure line is connected to the reactor vessel via another containment penetration.

The low-pressure injection system is normally on standby and starts automatically in situations which require emergency core cooling. The power supply to the pump motors is diesel-backed and thus not affected by the loss of auxiliary power. The capacity is 355 kg/s per loop, which is sufficient to compensate for the loss of coolant through a maximum-size pipe break, using only two loops. The system starts automatically on receipt of a signal indicating high temperature or high pressure in the reactor containment or low water level in the reactor vessel.

8.1.7 Containment spray system

The containment spray system (Fig. 8.4) consists of four independent loops, each with a pump and a heat exchanger. The system draws water from the condensation pool via suction lines equipped with strainers which also serve the auxiliary feedwater system and the low-pressure injection system. The water in each loop is pumped back to the condensation pool via sprinklers in the roof of the compression room above the condensation pool. Three of the loops are connected to separate pipelines and spray nozzles in the roof of the drywell on the pressure side of the pump. Drywell spraying is initiated manually. There is normally one loop in operation for cooling the condensation pool. All the loops are automatically actuated by signals indicating high temperature in the pool or start-up of the pressure relief system. In the event of a pipe break or a major leak in the primary system the water spray in the drywell contributes to reducing pressure in the containment by steam condensation. It also removes condensable fission products from the containment atmosphere.

8.1.8 Cooling water systems

The sea is the ultimate heat sink for the reactor power which is not utilized. During normal operation, cooling is primarily via the turbine condenser and the *main cooling water system*. A small part of the heat is removed by the cooling system for the condensation pool via intermediate cooling circuits to the sea. During reactor shutdown to temperatures below 188°C, corresponding to a reactor pressure of 1.2 MPa, steam production is no longer sufficient to maintain the function of the turbine condenser. The isolation valves in the steam lines are then closed and cooling is switched over to the *shutdown cooling system* which ensures continued cooling via the diesel-backed cooling circuits to the sea. Its intermediate cooling system is manually realigned so that the heat exchangers in the shutdown cooling system can receive water, while the normally connected heat exchangers in the condensation pool cooling system are isolated.

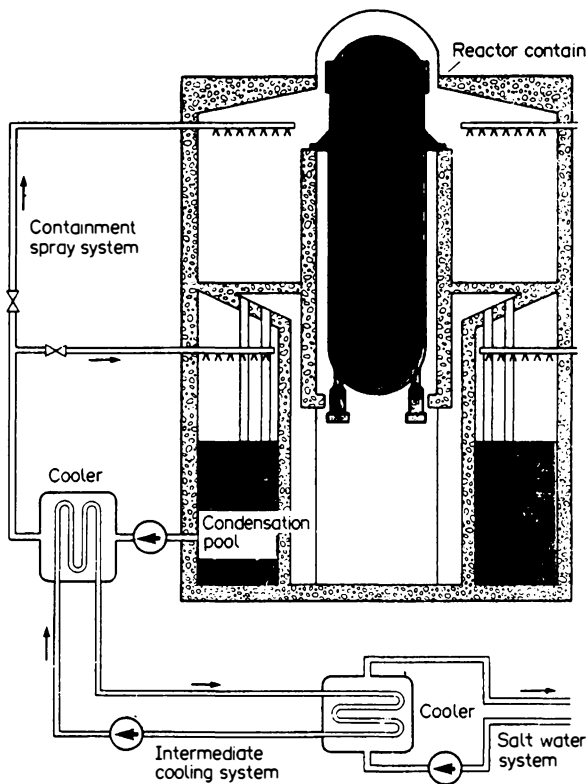


FIG. 8.4. Forsmark 3 containment spray system schematic. Courtesy Nuclear Training and Safety Centre, Studsvik

8.1.9 Plant-specific characteristics

All boiling water reactors are designed along the same basic principles. However, there are certain differences in the system design and in the detailed data, which can be important during fault operating conditions. The system descriptions in the previous sections apply to plants of the Forsmark 3/Oskarshamn III type. This section indicates some specific characteristics of other Swedish BWR plants (cf Table 2.1).

The most significant difference between the older external pump reactors and the newer internal pump reactors is that the risk of large bottom breaks has been virtually eliminated in the latter, by the absence of large pipe connections below the upper edge of the core. In addition, internal pump

reactors have safety systems divided into four trains, whilst external pump reactors have safety systems divided into two trains. The designs of the reactor containment also differs in a way which is important in certain cases.

The first Swedish BWR plant, Oskarshamn I, has, in contrast to the others, an auxiliary condenser for the removal of decay heat when the turbine condenser is unavailable. The condensate flows back to the reactor by natural circulation. The secondary side of the auxiliary condenser is cooled by boiling water, and the steam is blown off to the atmosphere.

In Oskarshamn I, the systems for emergency core cooling and containment cooling are located in the same room and not physically separated as in other plants. Since certain failures could then make both systems unavailable, the plant is provided with a special auxiliary feedwater system in a separate room.

Oskarshamn II and Barsebäck 1 and 2 are almost identical as regards safety-related equipment. They have, in contrast to other plants, a gas turbine-powered backup grid for the power supply of the feedwater pumps. This means that the feedwater system can be regarded as safety-grade.

Ringhals 1 has a high pressure coolant injection system and an auxiliary feedwater system with steam-driven pumps, which is unique among Swedish reactors. This reactor also has a higher steam relief valve capacity and a higher cooling capacity for the condensation pool than other reactors.

Forsmark 1 and 2 were the first Swedish reactors with internal recirculation pumps. Large liquid breaks in the primary system cannot occur in these reactors. It was therefore possible to reduce the number of blowdown lines as compared to the external pump reactors. A typical characteristic of the internal pump reactors is the annular condensation pool (Fig. 4.7), whilst the condensation pool in the external pump reactors covers the entire lower part of the containment (Fig. 11.1). Forsmark 1 and 2 have a spray function only in the drywell like the external pump reactors, whereas Forsmark 3/Oskarshamn III (F3/OIII) have an automatic spray function in the wetwell and a manually initiated spray in the drywell.

In contrast to other internal pump reactors, the emergency core cooling system in F3/OIII is divided into two core spray loops and two flooding loops connected to the downcomer. There is also a storage tank for feedwater which can be used for coolant make-up in cases when the feedwater system is available.

An important difference between F3/OIII and other reactors is that the former are designed to withstand earthquakes without impairing safety. This has meant, for example, that the auxiliary feedwater system draws water from the condensation pool instead of from special supply tanks outside the reactor containment as in the other plants.

Data for the safety systems are presented in Table 8.1.

8.2 Pressurized Water Reactors

The system descriptions in this section apply to Westinghouse reactors of the Ringhals 2–4 type.

8.2.1 Reactor protection system

As in boiling water reactors, the reactor protection system consists of:

- an analog part, comprising sensors and signal processing equipment;
- a logic part which analyses the signals in order to set diagnosis and develop signals to
- relays which initiate required action, such as scram, start-up of the emergency core cooling systems etc.

Ringhals pressurized water reactors have two redundant trains of logic units and relays which receive signals from four separate analog channels for each measurement variable.

Examples of some variables of interest from the aspect of safety are:

- neutron flux,
- rate of change of neutron flux,
- temperature in the hot and cold legs of the reactor coolant system,
- pressure and water level in the pressurizer,
- reactor coolant flow,
- feedwater flow,
- pressure in the main steam lines,
- water level in the steam generators,
- pressure in the reactor containment.

Measured values of these variables are used alone or in combination to derive electrical signals which actuate the required safety functions.

8.2.2 Shutdown systems

The main reactor shutdown system consists of control rods and control rod drive mechanisms as well as two trains of motor-generators and breakers. The control rods are maintained in a withdrawn position by having the motor-generators energize an electromagnetic latch in each drive mechanism. Opening the breaker, which is normally closed, releases the latch and the rods fall into the core by gravity. The breakers open automatically on signal from the reactor protection system. By means of special breakers, testing and maintenance work can be carried out on one of the trains even when the reactor is in operation.

Reactor shutdown can also be achieved by increasing the boron concentration in the coolant using the reactor's chemical and volume control system (see 5.4.2).

8.2.3 Pressure relief systems

The reactor coolant system is protected against overpressure by control and protective circuits such as the high-pressure actuated scram and by safety/relief valves connected to the top head of the pressurizer (Fig. 8.5). The safety/relief valves discharge into the pressurizer relief tank which collects and condenses the valve effluent. The relief tank is protected against a steam discharge exceeding the design pressure value by rupture discs which discharge into the reactor containment.

Each pressure relief valve is pneumatically operated by a pilot valve which is electrically controlled. Opening occurs automatically, when a signal is received indicating high pressure in the pressurizer, or manually, from the

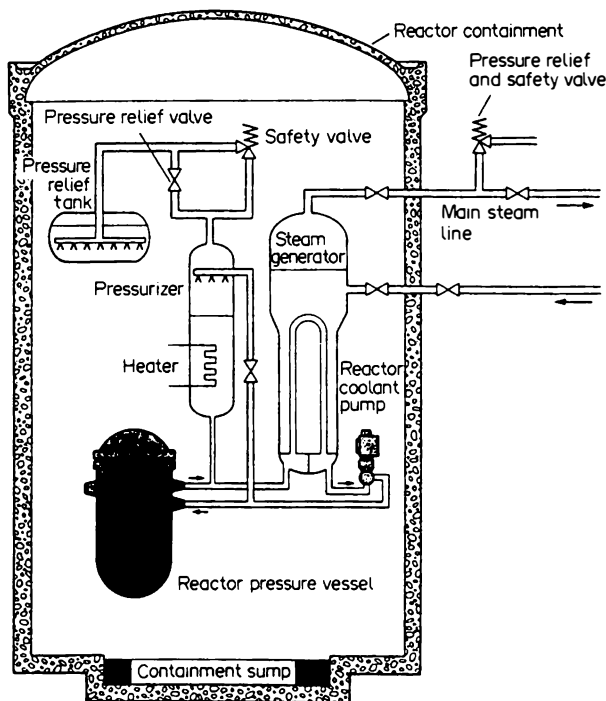


FIG. 8.5. Protection against overpressure in a pressurized water reactor.
Courtesy Nuclear Training and Safety Centre, Studsvik

control room. There is a motor-operated block valve for each relief valve which is normally open but which can be closed in the event of failure or leakage in the relief valve. The opening pressure of the relief valves is set at 16.1 MPa which is 0.35 MPa below the pressure which initiates scram. The safety valves, which are of the spring-loaded self-actuating type, open at 17.1 MPa. The safety valves are designed to cope with power overshoots (about 10%) during scram and turbine trip transients.

There are also pressure relief and safety valves in the main steam lines which discharge into the atmosphere (Fig. 8.5). They protect against over-pressure in the steam lines and are also used to blow off steam when the turbine condenser is unavailable. The valves have a capacity corresponding to full reactor power.

8.2.4 Auxiliary feedwater system

The purpose of the auxiliary feedwater system is to provide a supply of high-pressure feedwater for core decay heat removal following the loss of normal feedwater supply. The system delivers cold water to the steam generators' secondary side allowing heat to be dissipated through the secondary side safety/relief valves.

Two independent subsystems are provided. One subsystem employs a steam turbine driven 100% capacity pump with steam supplied from some or all of the steam generators. The other subsystem utilizes two 50% capacity electric motor driven pumps. The motor-driven units are connected to diesel generators for availability following loss of auxiliary power.

The head developed by the pumps is sufficient to ensure that feedwater can be delivered to the steam generators when the safety/relief valves are discharging. The pumps will normally take suction from the condensate storage tank system. Piping and valves are arranged to provide separate and redundant flow paths to each main feedwater line.

8.2.5 Emergency core cooling system

The purpose of the emergency core cooling system is to replace the lost coolant in the event of a pipe break or large leak in the reactor coolant system, so that core cooling is maintained. The emergency core cooling system consists of three subsystems:

- the high-head injection system,
- the accumulator system,
- the low-head injection system.

The *high-head injection system* is designed to supply coolant to the core in the event of small and medium-size breaks until the reactor pressure is low

enough for the low-head injection system to replace the lost coolant. During large pipe breaks the high-head injection system is not sufficient to replace the lost coolant, but the reactor pressure is reduced so quickly that the low-head injection system can be placed into operation almost immediately. Until the low-head injection system provides full capacity, water is supplied from the accumulator system.

A schematic diagram is shown in Fig. 8.6. During a pipe break, water will escape into the reactor containment and collect in a sump in the containment floor. The high-head injection system first draws water from a storage tank filled with boric acid solution, and this is then pumped into the cold legs of the primary circuit loops. The pumps are identical to the three charging pumps in the chemical and volume control system (cf 5.4.2), one of which is continually in operation for reactor coolant make-up. The other charging pumps are automatically actuated by signals from the reactor protection system, although they can also be started manually.

When the pressure falls below 4 MPa, water is automatically injected into the primary loop from the *accumulator system*. Three accumulators are provided, one for each loop, filled with boric acid solution and pressurized

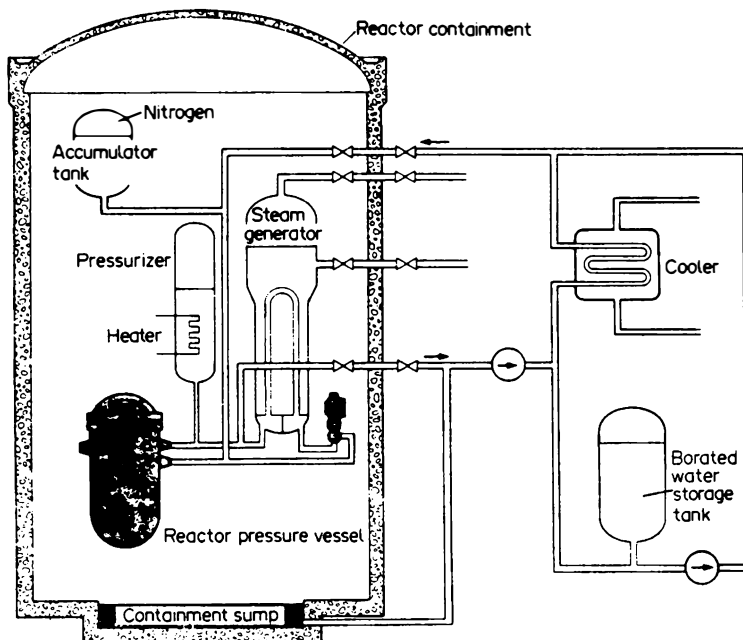


FIG. 8.6. Emergency core cooling systems in a pressurized water reactor.
Courtesy Nuclear Training and Safety Centre, Studsvik

with nitrogen. The accumulators are an example of a passive system which does not require any mechanical or electrical energy to function. As soon as the reactor pressure falls below the accumulator pressure, water is forced into the primary loop.

The *low-head injection system* first draws water from the storage tank. When the tank is nearly empty, the low-head pumps are realigned to recirculate water from the containment sump via heat exchangers. These two pumps and heat exchangers form part of the cooling system which is normally used for decay heat removal after shutdown, known as the residual heat removal system (see 8.2.7). The realignment of the suction lines of the pumps from the storage tank to the containment sump is carried out manually.

The high-head injection system can also draw water indirectly from the containment sump when the storage tank is empty by connecting the suction lines of the charging pumps to the pressure side of the low-head injection system. Thus both the high-head injection system and the low-head injection system have two operating modes. One is called *safety injection* and the other *recirculation*. Realignment is carried out by the reactor operator upon receipt of a signal indicating low liquid level in the storage tank or when the containment sump is at least 45% full.

8.2.6 Containment spray system

The basic purpose of the containment spray system is to cool the containment atmosphere when appropriate. Borated water is pumped via a heat exchanger from the storage tank through spray nozzles in the roof of the containment (Fig. 8.7). The water collects in the containment sump. When the storage tank is empty, water is drawn from the sump and recirculated.

The system has two independent loops. Each loop consists of two pumps and two heat exchangers in parallel trains. Realignment to recirculation is carried out when the operator opens two motor-driven valves in series for each loop. These valves normally isolate the containment sump from the spray system. The operator then closes the valves in the suction lines from the storage tank. The containment spray system not only cools the reactor containment but also provides, during recirculation, redundancy for the low-head injection system for emergency core cooling.

8.2.7 Residual heat removal system

During normal shutdown to “cold” conditions, the steam generators and the turbine condenser are first used to remove heat and lower the pressure. When the pressure falls below 3 MPa, the residual heat removal system is taken into operation and ensures the continued cooling of the shutdown reactor. The pumps in the residual heat removal system then take suction

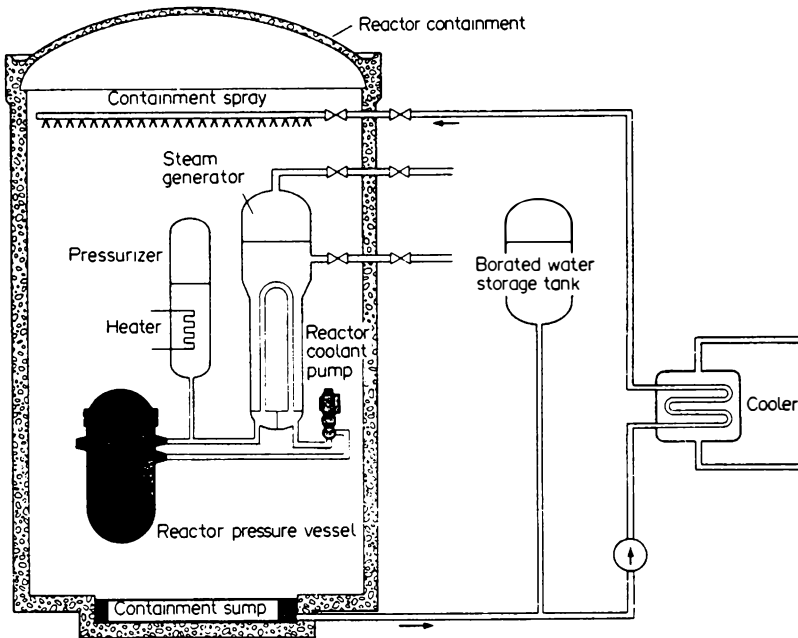


FIG. 8.7 Pressurized water reactor containment spray system schematic.
Courtesy Nuclear Training and Safety Centre, Studsvik

from the reactor coolant system and circulate the water through coolers back to the reactor. The residual heat removal system is not a safety system in the true sense, but its pumps and heat exchangers form part of the low-head injection system for emergency core cooling.

8.2.8 Cooling water systems

During normal operation, most of the waste heat generated by the plant is removed by the reactor coolant system and the turbine condenser and discharged into the sea. A small amount is removed by the *component cooling water system* which cools some of the pumps and heat exchangers in the normal operating systems, such as the main coolant pump bearings and shaft seals (see 5.2.1) and the heat exchangers in the chemical and volume control system. The safety function of the component cooling water system includes the removal of heat from the four heat exchangers in the containment spray system and the two heat exchangers in the residual heat removal system.

The component cooling water system contains three diesel-backed pumps and two heat exchangers. During normal operation, one pump and one heat exchanger ensure the performance of the system. The second pump is on standby and starts automatically if the main pump fails. The third pump serves as back-up and is connected to the second heat exchanger.

The heat exchangers in the component cooling water system are cooled by the *salt water system* to the sea. The salt water system has two redundant trains, each with three diesel-backed pumps and one heat exchanger. There are normally three pumps in operation, two in the first train and one in the second. One pump in each train provides enough water to cool the heat exchangers of the component water cooling system. However, during realignment to the recirculation mode in connection with emergency core cooling and containment spray cooling, two pumps are required in each train.

8.3 Safety Functions

As mentioned previously, there is no precise distinction between operating systems and safety systems. Both types often interact to carry out a particular safety function. It is therefore better to speak of *safety-related systems*. Safety-related systems also include systems which do not directly affect the course of events in an abnormal situation, but whose function is necessary for the systems directly involved. The auxiliary power supply systems and secondary cooling systems are examples of such safety-related systems.

A particular feature of safety-related systems is the very high requirements for availability. This is achieved by designing the systems to incorporate redundancy and diversification so that the failure of one component or subsystem does not jeopardize the function of the whole system. All functions which must be carried out rapidly are automatic. Action which does not need to be carried out rapidly is performed manually, such as the realignment of the residual heat removal system. In the following sections some essential safety functions in boiling and pressurized water reactors are compared.

8.3.1 Reactor coolant make-up

Reactor coolant make-up means supplying the primary system with enough water to ensure satisfactory core cooling under all normal operating conditions and in most abnormal situations, with the exception of a large loss of coolant accident.

In the *boiling water reactor*, make-up water is normally supplied by the feedwater system, which receives water from the condensate system. If the feedwater system is not available, for example, due to malfunction of the

turbine condenser or loss of auxiliary power, the auxiliary feedwater system (8.1.5) will assume the make-up function. Water is then drawn from the containment condensation pool. The pool water is replenished by condensing steam from the reactor.

In the *pressurized water reactor*, the make-up function is carried out by the chemical and volume control system (5.4.2). Charging pumps draw water from storage tanks containing deionized water and boric acid. The water and boric acid are mixed to obtain the desired boron concentration in the reactor coolant system.

8.3.2 Emergency core cooling

In the event of a pipe break or large leak in the primary system, the make-up function is not sufficient to replace the lost coolant. Scram and emergency core cooling are therefore initiated. The reactor is isolated by closing the containment isolation valves in all systems not used for emergency core cooling. The emergency cooling systems cool the core and condense and cool the steam escaping into the containment.

During small pipe breaks in *boiling water reactors*, the auxiliary feedwater system (8.1.5) is used for core cooling, and the containment spray system (8.1.7) for containment cooling. If the water level in the reactor vessel cannot be maintained, automatic depressurization is initiated, after which the low-pressure injection system (8.1.6) is used. When a large pipe break occurs, the pressure rapidly falls below 1.5 MPa and the low-pressure injection system begins to pump water into the reactor. Figure 8.8 is a schematic diagram of the systems employed during emergency core cooling, with system numbers used for Swedish boiling water reactors.

Emergency core cooling in the *pressurized water reactor* was described in section 8.2.5. A schematic diagram of the emergency core cooling systems, with system acronyms for U.S. pressurized water reactors, is shown in Fig. 8.9. These acronyms are also used in Sweden.

8.3.3 Residual heat removal

The purpose of the residual heat removal system is to remove the decay heat generated by the fission products after the nuclear chain reaction has ceased (see 3.4.5).

In the *boiling water reactor*, residual heat removal is normally effected by carrying steam from the reactor to the turbine condenser and the main cooling water system. The condensate is returned to the reactor via the condensate and feedwater systems. At temperatures below 188°C, the shut-down cooling system (8.1.8) is taken into operation. Another cooling route, used when the main condenser is unavailable, is via the pressure relief system (8.1.3) to the condensation pool in the reactor containment. The

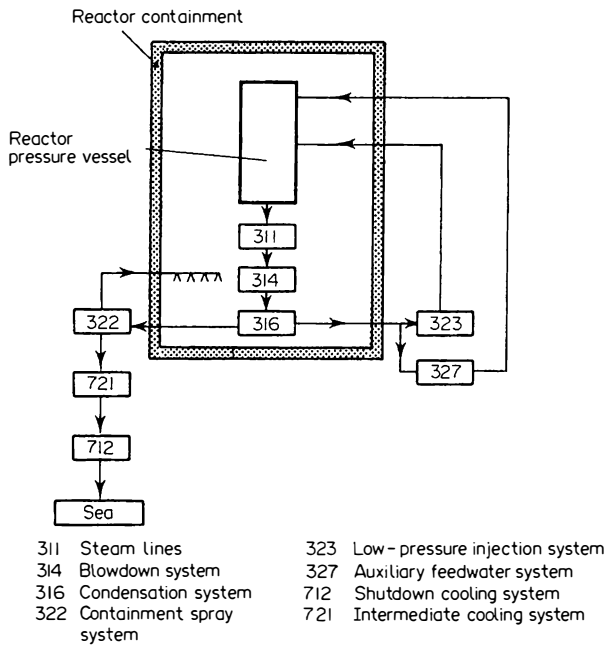


FIG. 8.8. Emergency core cooling in a boiling water reactor

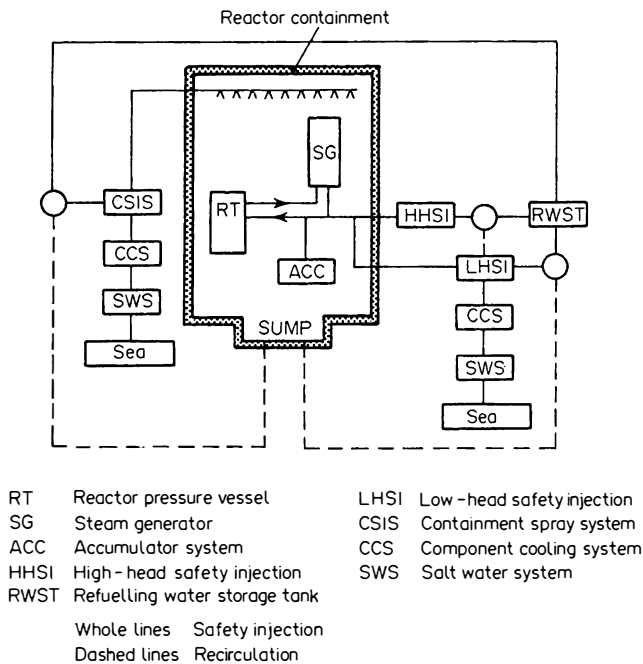


FIG. 8.9. Emergency core cooling in a pressurized water reactor

condensation pool is cooled by the containment cooling system (8.1.7) from which the decay heat is removed by the diesel-backed cooling systems to the sea.

When the turbine condenser is unavailable as a heat sink, the excess steam is discharged from the reactor into the condensation pool in order to maintain a constant reactor pressure. Make-up coolant is supplied by the main feedwater system (by the auxiliary feedwater system in external pump reactors). In F3/OIII, make-up coolant is supplied from a special tank (cf 8.1.9). The water then has a temperature of about 170°C and contributes, along with the decay heat, to heating the condensation pool water.

The capacity of the pool cooling system depends on the difference in temperature between the water in the condensation pool and the ultimate heat sink, the sea. Therefore, the capacity is low before the pool water is heated. Figure 8.10 shows how the supplied heat power and the cooling power vary with time. The difference between the heat supplied and the heat removed is stored in the pool. The stored heat decreases as the decay heat decreases and the pool temperature and the cooling power increases. After about 4 hours the cooling power is greater than the heat power supplied and the pool temperature falls with the decreasing decay heat.

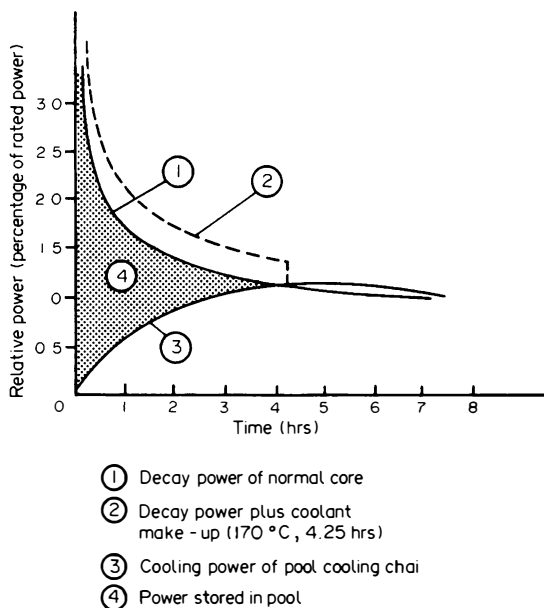


FIG. 8.10. Decay power and cooling power in the condensation pool of a boiling water reactor with internal recirculation pumps. From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

The normal residual heat removal in *pressurized water reactors* is described in section 8.2.7. The same pumps and heat exchangers used during normal residual heat removal are also used in the low-head injection system for emergency core cooling in the recirculation mode. Another cooling route is via the containment spray system (see Fig. 8.9).

8.4 Data for Safety Systems

The description of safety systems and safety functions is summarized with a presentation of design data for boiling water reactors (Table 8.1) and for pressurized water reactors (Table 8.2). Some differences in data can be noted in different generations of reactors.

TABLE 8.1 *Data for safety systems in Swedish boiling water reactors*

System	Unit	OI	OII	F1	F3
SHUTDOWN SYSTEMS					
<i>Control rod system</i>					
Number of control rods		112	109	161	169
Number of control rod groups		28	17	18	18
<i>Boron system</i>					
Number of loops	—	1	1	2	2
Volume of storage tank	m ³	5	7	25	2 × 11
Pump capacity	kg/s	2 × 2.5	2 × 3.5	2 × 2.5	2 × 2.5
PRESSURE RELIEF SYSTEM					
Number of safety, pressure relief and control valves		16	22	13	18
Number of safety valves	—	12	13	1	8
Opening pressure	MPa	8.5	8.5	8.5	8.0–8.35
Capacity per valve at opening pressure	kg/s	66.5	66.5	86.1	123
Number of pressure relief valves	—	4	7	10	8
Opening pressure					
electrically controlled	MPa	7.4–7.55	7.2–7.7	7.4	7.4
impulse controlled	MPa	8	8	8	8–8.5
Capacity at nominal reactor pressure	kg/s	55	5 × 55	70	107.6
Number of control valves	—	—	2 × 23	2	2
CONDENSATION SYSTEM					
Pool volume at normal water level	m ³	1843	1924	2980	3166
Blowdown pipes					
number		96	96	40	24
submergence	m	1.5	3.0	7.0	5.0
inner diameter	m	0.6	0.6	0.6	0.6
Vacuum breakers					
number		—	7	10	8

9

Deterministic Safety Analysis

Safety analysis is the study of how the reactor behaves during fault conditions. Safety analysis is a step in the design process and an essential part of the safety assessment in the licensing process. Plant safety is continuously monitored during operation and recurrently analysed in order to maintain and, if needed, raise the level of safety.

Safety analysis is carried out in two different ways which complement each other. *Deterministic* safety analysis means that the behaviour of the plant after an assumed initial event or malfunction is studied with calculational models which describe the physical processes in the main reactor systems. The aim of this type of analysis is to verify that permissible values of essential plant variables are not exceeded. *Probabilistic* safety analysis concentrates on identifying event sequences which can lead to core melting and on studying the reliability of the safety systems. The aim of this type of analysis is to indicate weak points in the overall safety design and to provide a basis for improving safety.

This chapter describes the main features of the deterministic analysis of events *within* the design basis, i.e. of primary system and reactor containment behaviour after malfunction of the normal operating and control systems when the required safety systems are available as intended. The deterministic analysis of events *beyond* the design basis, i.e. when essential safety systems are *not* available as intended, is treated in Chapter 11.

9.1 Type of Events

Events important to safety include all circumstances with significant deviation from the normal values of essential primary system variables, such as pressure, temperature, heat flux, coolant flow and coolant density. These events can be initiated by component failure or by human error. They can also be caused by extraneous events such as fire or earthquake. For the purpose of analysis, abnormal events are usually grouped into three main categories:

—*LOCA* (Loss-of-Coolant-Accident), i.e. events caused by a pipe break or leakage in the primary system;

- transients*, a general term for all events (except LOCA) leading to imbalance between the rate of heat release and heat removal in the reactor;
- external events*, i.e. earthquake, fire, flooding, lightning, explosions, etc.

The classification is largely historical, resulting from the importance accorded in the U.S. safety philosophy to a large LOCA, i.e. a postulated large pipe break in the main coolant system as the initiating event in the design basis accident for the emergency core cooling system and reactor containment.

9.1.1 LOCA

A LOCA is caused by a pipe break or leak in the primary system of such magnitude that the capacity of the make-up systems is insufficient to replace the lost coolant. This results in reactor scram, closure of containment isolation valves and initiation of emergency core cooling. The course of events is briefly as follows:

1. A break occurs in the primary system and water escapes at high pressure and temperature into the reactor containment.
2. The emergency core cooling systems supply water to keep the core sufficiently cooled.
3. Radioactive substances which may be released from the core are retained within the containment.
4. The containment spray system cools the containment and removes radioactive substances from the containment atmosphere.

If the safety systems operate as intended, the core cooling will be maintained and the fuel will remain mechanically intact. The release of fission products from the fuel will be small and the offsite consequences negligible.

A LOCA can be initiated in several ways, e.g. through a pipe break in the primary system, the failure of a pressure relief valve to close, or a tube rupture in a steam generator (PWR).

Regarding the size of the break, a distinction is made between large, medium and small LOCA. The event progression is different in these cases, as described in sections 9.4 and 9.5. For boiling water reactors, the break is said to be internal or external, depending on whether it occurs inside or outside the containment.

9.1.2 Transients

Most transients are controlled by the normal operating and control systems without interruption of reactor operation. In certain cases, the reactor power must be quickly reduced to prevent core overheating. This type of

transient is the main object of safety analysis. Events involving abnormal increase in reactor power, decrease in coolant flow or increase in reactor pressure belong to this category. Safety analysis also applies to the shutdown reactor, since the core can overheat if the fission product decay heat is not efficiently removed.

Transients of importance to safety can be roughly classified according to the anticipated frequency:

- transients which are expected to occur sometime during an operating year;
- transients which are expected to occur sometime during the lifetime of the reactor.

The first category includes transients caused by a single equipment failure or single operator error, such as malfunction of the feedwater system, temporary loss of offsite power, turbine trip, inadvertent reactor isolation. The more unusual transients include those initiated by large reactivity insertion, long-duration loss of power or several simultaneous system failures.

9.1.3 Design basis accidents

Design basis accidents are a special category of events which are not expected to occur at all during the reactor lifetime but which are postulated as a basis for the design of the safety systems. Examples of design basis accidents (DBAs) are:

- large LOCA, initiated by a double-ended break of the largest main coolant pipeline (DBA for the emergency core cooling system and reactor containment);
- large RIA (Reactivity Induced Accident), a transient with rapid reactivity insertion (DBA for the reactor shutdown system);
- transient with high reactor pressure (DBA for the pressure relief system);
- extreme external events such as earthquakes, strong winds, flooding, etc. (DBAs for buildings and structures).

The analysis of design basis accidents and the validation of the analysis are important areas in the assessment of safety.

9.1.4 Event classification

It is not possible to analyse all conceivable types of events. For the purpose of analysis, the events may be grouped according to their expected frequency, for example as shown in Table 9.1.

According to this classification, only events in categories H_2 to H_5 are of

importance to safety. Examples of such events are given in Table 9.2.

Events in category H₂ to H₄ are examined in sections 9.4 to 9.7 below. Events in category H₅ are analysed in Chapters 10 and 11.

TABLE 9.1 *Event classification for safety analysis*

Event	Frequency (per year)	Designation
Disturbances controlled by normal operating and control systems without interruption of operations	> 10	H ₁
Anticipated, moderately frequent events which may result in safety chain actuation	10 ⁻¹ –10	H ₂
Anticipated, infrequent events resulting in safety chain actuation	10 ⁻³ –10 ⁻¹	H ₃
Improbable events postulated for safety system design	10 ⁻⁵ –10 ⁻³	H ₄
Very improbable events not included in the design bases	< 10 ⁻⁵	H ₅

TABLE 9.2 *Examples of events of importance to safety*

Category	Event
H ₂	Load rejection Turbine trip Uncontrolled boron injection (PWR) Inadvertent reactor isolation (BWR)
H ₃	Small LOCA Loss of reactor coolant pumps (PWR) Reactor isolation with loss of offsite power (BWR)
H ₄	Main recirculation line break (DBA-LOCA) Main steam line break (PWR) Reactor isolation without scram (BWR)
H ₅	Reactor vessel rupture LOCA without emergency core cooling Transients without reactor shutdown

9.2 Criteria

The basic approach of deterministic safety analysis is to specify bounding values of essential plant variables and to show by analysis that the criteria are met for typical initial events. In this section some of the criteria are discussed.

9.2.1 Emergency core cooling

In order to assess the efficiency of emergency core cooling, the U.S. Atomic Energy Commission (AEC) established criteria which are also applied in other countries. Since full-scale experimental verification of large LOCA is not feasible, the criteria are based on the *calculated* course of events for the worst conceivable case. The criteria are general and do not differentiate between boiling water reactors and pressurized water reactors. The criteria are specified in five points (901):

1. The calculated maximum fuel rod clad temperature shall not exceed 2200°F (1204°C).
2. The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4. The calculated changes in core geometry shall be such that the core remains amenable for cooling.
5. After any calculated successful initial operation of the emergency core cooling system, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Requirements are also specified for the methods of calculation. The aim is to support the calculations as far as possible with experimental data on separate effects, and to ensure that the calculations provide conservative results.

When the emergency core cooling criteria were established in the early 1970s, safety design was mainly directed at mitigating the consequences of a large LOCA. The criteria resulted in limitations on heat loads in the core during normal operation. The requirements on capacity and availability of the emergency core cooling systems were tightened. Large experiments were launched to validate the analytical methods. The safety of other types of LOCA which develop more slowly and which may require manual intervention by the reactor operator were not given the same attention.

9.2.2 Heat loads

A general criterion for transients is that the *critical heat flux* at the fuel cladding surface shall not be exceeded anywhere in the reactor. At critical

heat flux, the clad temperature rises rapidly (see 3.4.3), possibly resulting in clad damage.

The margin to critical heat flux is usually defined differently for the pressurized water reactor and the boiling water reactor. This is related to the experimental correlations used for the critical heat flux. For PWRs, the ratio of the critical heat flux and the local surface heat flux is determined. This ratio is called *DNBR* (Departure from Nucleate Boiling Ratio). For BWRs, the ratio of the fuel assembly power causing critical heat flux at the real coolant flow rate and the real fuel assembly power in a particular coolant channel is determined. This ratio is known as *CPR* (Critical Power Ratio).

In order to take into account uncertainties in the experimental correlations and in the thermohydraulic calculations, one of the following two procedures is specified (902):

- (a) The DNBR or CPR shall be determined so that with 95% probability at 95% confidence level the hottest fuel rod does not exceed the critical heat flux.
- (b) A minimum value of DNBR or CPR shall be determined so that at least 99.9% of the fuel rods will not run the risk of reaching critical heat flux.

In practice, pressurized water reactors are designed so that the minimum DNBR is greater than 1.50 at steady-state operation and greater than 1.30 during transients.

The minimum CPR for boiling water reactors is usually greater than 1.30 at steady-state operation and greater than 1.06 to 1.07 during transients.

9.2.3 Fuel enthalpy

A condition for avoiding fuel-coolant interaction (cf 3.4.7) is that the energy deposition in the fuel during a power excursion be limited. Since some of the energy deposited is due to delayed fissions, fuel rod damage during a power burst is better correlated with the fuel enthalpy than with total energy. The fuel enthalpy is 10–25% less than the total energy. The criterion, as formulated by the U.S. Atomic Energy Commission (903), is that the radial average fuel enthalpy is not greater than 280 cal/g UO_2 (1172 J/g UO_2) at any axial location in any fuel rod.

9.2.4 Pressure relief

For Swedish boiling water reactors, the total capacity of the pressure relief system shall be sufficient to prevent the system pressure from exceeding the reference value established in the Swedish pressure vessel code, i.e. 10% over and above the design pressure of the reactor vessel. This requirement

applies even if scram fails during maximum pressure transients, such as transients involving isolation of the reactor from the turbine.

9.2.5 Reactor scram

Transients of importance to safety can generally be defined as events which initiate reactor scram. The control rods are automatically actuated by electrical signals and are inserted in groups (cf 8.1.2). Events which may cause large forces on core structures, e.g. in connection with large pipe breaks, must not deform the geometry of the core and prevent the insertion of the control rods. For the purpose of safety analysis, all scram groups except the most reactive one, are assumed to enter the core during scram. The failed scram group is assumed to remain in a completely withdrawn position. Hence, a safety margin is obtained for the unlikely case of a control rod getting stuck in the withdrawn position. An additional margin is obtained by the requirement that the calculation shall result in a safe reactivity deficit, usually less than 1%, with all scram groups inserted except the most reactive one.

9.3 Analytical Methods

Deterministic safety analysis is the study of selected LOCA and transients using calculational models which provide the time history of essential plant variables after the initiating event. The purpose is to verify the safety design, to show that the licensing requirements are fulfilled, and to make realistic safety assessments for actual or anticipated events. Essential variables include clad temperature, rod surface heat flux and reactor pressure as well as temperature and pressure in the reactor containment. Thermohydraulic calculational models are set up based on mass, energy and momentum balance. Since the models are only more or less accurate approximations of reality, their validity must be tested in realistic experiments.

The calculational models are incorporated in computer codes for LOCA analysis, transient analysis and containment analysis.

9.3.1 LOCA analysis

Computer codes for LOCA analysis describe the thermohydraulic processes during loss of coolant in the primary system. The primary system is divided into a number of control volumes which are linked through one or several flow paths. The fluid may contain one or several phases (steam, water, gas). The computer code solves the equations for the conservation of mass, energy and momentum of the fluid in one-dimensional geometry. Due to the rapid non-linear processes during large LOCA, the numerical

solution becomes complex. Since certain basic phenomena and mechanisms are insufficiently known, two different types of models are used:

- licensing models* including conservative assumptions prescribed by the safety authorities;
- realistic models* using best-estimates of insufficiently known phenomena and mechanisms.

The licensing models generally predict peak clad temperatures several hundred degrees higher than the realistic models in the analysis of large LOCA. Large-scale integral experiments have confirmed that the licensing models are conservative.

9.3.2 Transient analysis

In transient analysis, the feedback between reactivity (reactor power) and thermohydraulics (heat transport) is generally of importance. Hence, besides thermohydraulic models, the computer codes also contain models for reactor kinetics and control system performance. Moreover, the thermohydraulic processes are generally much slower than during large LOCA and can be described by simpler models. However, certain transients require a detailed spatial description of events in the reactor core. Transient models are often also used for the analysis of small and medium LOCA. “Shutdown transients” represent a special class of transients where the energy and mass balance in the shutdown reactor are studied.

9.3.3 Containment analysis

In order to predict pressure and temperature in the reactor containment during a LOCA, special calculational models and computer codes are used. The containment is divided into a suitable number of compartments assumed to contain a gaseous and liquid phase. The gaseous phase may contain non-condensable gases and superheated or saturated steam including water droplets. The liquid phase consists of subcooled or saturated water and possibly air and steam bubbles. The mass and energy conservation equations for each phase and component in each compartment are solved. The mass flow between the compartments is calculated using momentum equations. The conditions in the condensation pool during blowdown are of particular interest in boiling water reactors.

9.4 LOCA in Boiling Water Reactors

A LOCA is initiated by a break or leak in the primary system. It is practical to distinguish between breaks occurring above and below the upper

edge of the core (“top breaks” and “bottom breaks”) as well as between large and small breaks. Large breaks are generally characterized by a rapid drop in reactor pressure so that the low-pressure injection system can deliver water to the reactor. During small breaks or leaks, the capacity of the normal make-up systems is sufficient to replace the lost coolant and to maintain the water level in the reactor vessel. Whether the pipe break takes place inside or outside the reactor containment is also of importance. The flow from external breaks can be limited by closure of the isolation valves in the corresponding pipelines, while the break flow from internal pipes cannot be shut off in this way.

9.4.1 Main recirculation line break

In external pump reactors (Fig. 4.5), a break in a main recirculation line connected to the bottom of the reactor vessel constitutes a design basis accident. In modern internal pump reactors, large bottom breaks cannot occur since the external recirculation loops have been eliminated. In these reactors, the main recirculation pump casings and control rod drive mechanisms connected to the bottom of the reactor are equipped with flow restrictors so that the outflow in case of a break is strongly limited.

The course of events during a postulated large LOCA in a reactor with *external recirculation* is initiated by a double-ended (“guillotine”) break in a recirculation line (650 mm in diameter) near the inlet nozzle in the bottom of the reactor vessel. The initial break flow is estimated at 20,000 kg/s. Immediately after the break, scram and reactor isolation are actuated. Off-site power is postulated to be unavailable when the turbine generator ceases to supply power. The thermohydraulic processes are roughly divided into a blowdown phase and an emergency core cooling phase (Fig. 9.1).

Shortly after the break, the coolant flow reverses in the core. Dryout occurs within 2 seconds. The cooling deteriorates severely and the clad temperature begins to rise. After a short period of steam cooling, temporary rewet is caused by the downward flow of a two-phase mixture of steam and water generated by intensive boiling of the water in the gaps between the coolant channels. Once the downcomer has emptied after about 15 seconds, steam escapes through the downcomer and the break, causing the reactor pressure to drop rapidly. After about 30 seconds the pressure in the reactor vessel and containment equalizes and the flow through the core stagnates.

The low-pressure injection system is expected to be in operation after about 20 seconds, when water is sprayed over the core. Water will gradually wet the walls of the fuel channel and then the clad walls. The spray cooling causes the clad temperature to pass a maximum after a few minutes. It is this peak clad temperature which must be shown to be lower than 1204°C in the analysis, using licensing calculational methods.

After about 30 minutes, the clad temperature has fallen to a low level. A

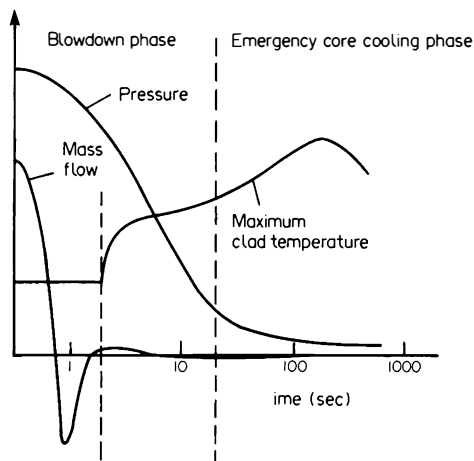


FIG. 9.1. Schematic diagrams of a large LOCA in a boiling water reactor with external recirculation

decay heat removal period is then initiated, during which it is sufficient to replace the water boiled away by the fission product decay heat. The reactor vessel must be refilled for the core to be eventually accessible. After large pipe breaks, this can only be achieved by flooding the entire reactor containment.

Within the first few seconds after the break, large reaction forces appear on the reactor vessel and internals due to the escaping water. The pressure of steam and gas in the drywell will force water and gas through the blowdown pipes into the condensation pool. This causes the pool water to swell, which results in large dynamic loads in the wetwell. The reactor vessel and internals, pipelines and containment are designed to withstand these loads.

Pool water cooling is automatically initiated after the break and is achieved by spraying the compression chamber above the pool with water and by recirculating the water via coolers (cf 8.1.7). Spraying of the drywell is initiated manually. The spray water tends to limit the pressure and temperature in the containment atmosphere by steam condensation. However, the manually initiated drywell spray is not credited in the analysis until at least 30 minutes after the initial event (cf 7.1).

9.4.2 Main steam line break

In boiling water reactors with *internal recirculation* an assumed guillotine break in a main steam line inside the reactor containment is representative of a large LOCA. The steam flow causes an increase in the pressure and temperature of the containment, which initiates reactor scram and closure of the steam line isolation valves. The reactor vessel pressure decreases

rapidly, causing the water in the reactor vessel to swell and reach the steam outlet nozzles. The character of the break flow then changes from steam to a two-phase mixture of steam and water. When the water inventory in the reactor vessel diminishes, the break flow again changes to steam flow. After a few minutes, the pressures in the vessel and containment equalize and the blowdown phase ends. A depressurization phase is then initiated by the actuation of the containment spray system. The spraying of the compression chamber and cooling of the condensation pool is automatically initiated when the break occurs, and is assumed to start after 50 seconds. The spraying of the drywell is manually initiated and is assumed to start after 30 minutes.

Calculations show that, provided the main recirculation pumps keep running for at least 5 seconds after the break, the core remains well cooled during the whole blowdown phase due to the effective heat transfer to the flashing mixture of steam and water. The peak fuel clad temperature is kept only slightly above saturation temperature. When the pressure difference between the reactor vessel and the wetwell compression chamber is approximately 1.2 MPa, the low-pressure coolant injection system becomes operable and starts refilling the reactor vessel. Figure 9.2 shows the calculated relative water level and pressure in the reactor vessel.

The relative (collapsed) water level is defined by:

$$\phi = (\text{water volume in the reactor vessel})/(\text{water volume below the upper edge of the core}).$$

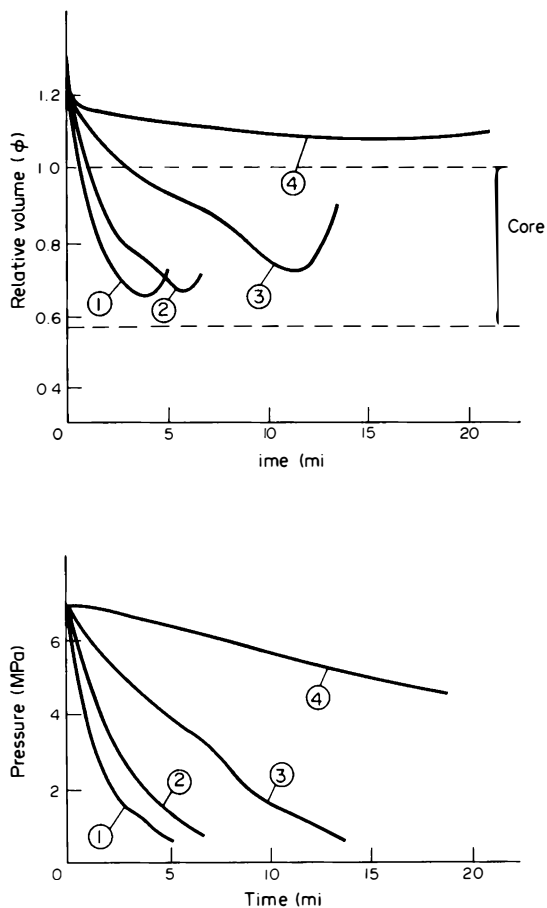
This definition means that the core will be covered by water if $\phi > 1$. The core can also be well cooled if $\phi < 1$, since there may be a two-phase (swell) level above the upper edge of the core.

In Fig. 9.2 only two out of four subsystems of the auxiliary feedwater system and of the low-pressure coolant injection system are assumed to be available. Loss of offsite power is assumed to occur simultaneously with the pipe break. The detailed analysis shows that the LOCA criteria (9.2.1) are met with a considerable margin (904).

9.4.3 Small and medium breaks

For small and medium breaks, the steam flow (top breaks) or water flow (bottom breaks) leads to an increase of the reactor containment temperature, which initiates closure of the isolation valves, reactor scram and opening of the pressure relief valves. The continued process depends on the type of break as well as on the particular reactor type. The following description applies to internal pump reactors of the Forsmark 3 type (903).

For *small top breaks* with a steam flow < 80 kg/s, the water level in the reactor vessel can be maintained by one or two auxiliary feedwater



Curve	Break area (% of max area)	Initial break flow rate at 7 MPa (kg/s)
①	100	950
②	60	570
③	20	190
④	5	48

FIG. 9.2. Calculated water level and pressure in the reactor vessel after steam line breaks in Forsmark 3. From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

subsystems. Each of the four subsystems has a capacity of 22.5 kg/s and draws water from the condensation pool. The cold auxiliary feedwater and the escaping steam cause the reactor pressure and the break flow to decrease. For small steam flows, the depressurization is very slow (Fig. 9.3, curve 1). The decay heat generates steam which discharges through the break. During larger break flows (Fig. 9.3, curve 2), the pressure decreases more rapidly. The decay heat then produces a smaller part of the steam, the major part originating from stored energy in the reactor coolant and reactor internals (cf Table 3.4).

During *medium top breaks* with a steam flow < 500 kg/s, there is a rapid drop in reactor pressure, causing the reactor coolant to swell. The flow decreases in proportion to the drop in pressure. When the water level falls below a preset value, automatic depressurization is initiated. This is followed by the start-up of the low pressure injection system which keeps the core covered with water. If the initial break flow is greater than about 300 kg/s, the pressure drops so rapidly that automatic depressurization is not important. At break flows less than 300 kg/s, the auxiliary feedwater system (three loops) is sufficient to keep the core covered with water for most of time (as shown in Fig. 9.3, curve 3).

During *small bottom breaks* with a liquid flow < 45 kg/s, the water level in the reactor vessel can be maintained by the auxiliary feedwater system. However, it must compensate for the break flow as well as for the steam generated by the residual heat. With an initial break flow of 45 kg/s and an auxiliary feedwater supply of 45 kg/s (two loops), the level in the reactor first falls, since steam is discharged to keep the pressure constant. After a short time, the steam discharge and the break flow decrease so that the two auxiliary feedwater loops can restore the normal water level in the reactor vessel. The water level is at all times above the upper edge of the core.

In Swedish internal recirculation boiling water reactors, 45 kg/s represents the largest break flow that can conceivably be obtained in a bottom break. However, in the safety analysis of Forsmark 3, a bottom break of 80 cm² is postulated, which corresponds to an initial liquid flow of about 500 kg/s. The capacity of the auxiliary feedwater system is then insufficient to compensate for the lost coolant. If the main feedwater system is unavailable, the pressure must be rapidly decreased so that the low-pressure injection system can be used. Calculations show that automatic depressurization is initiated after about 1 minute and that the pressure decreases to 1.2 MPa after about 5 minutes when the low-pressure injection system (LPIS) can start to reflood the core. The water level then rises relatively rapidly (Fig. 9.4). Assuming that two (of four) LPIS subsystems are in operation, the maximum clad temperature is achieved after about 6 minutes. While some core uncover and heat-up occurs in this case, the peak clad temperature stays well below permissible levels.

In general, the course of events after a top break is characterized by a

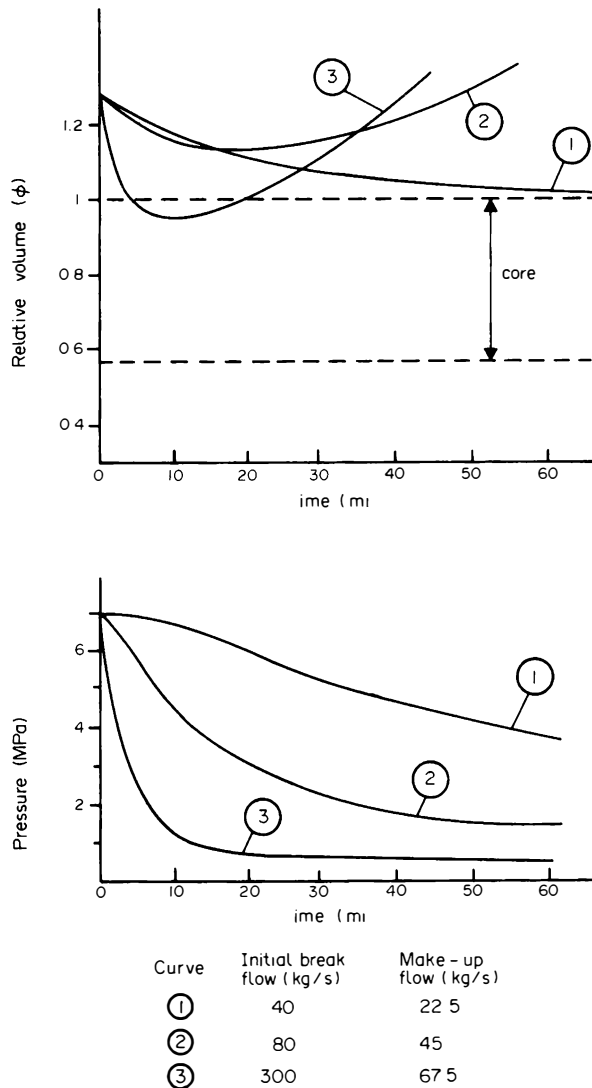
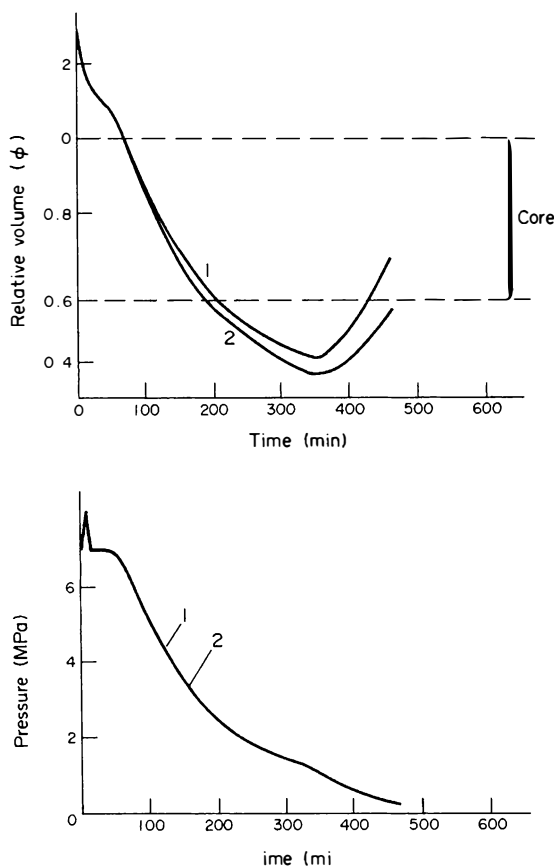


FIG. 9.3. Calculated water level and pressure during small and medium top breaks in Forsmark 3. Adapted from *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985.

relatively rapid decrease of the reactor pressure and a slow decrease of the water level. A bottom break typically leads to a decrease in the water level while the pressure is maintained. A break at an intermediate level, such as in a feedwater or emergency core cooling line, results in behaviour which



Curve	Automatic depressurization	No. of low-pressure core cooling circuits	No. of auxiliary feedwater circuits
1	Yes	3	3
2	Yes	2	2

FIG. 9.4. Calculated water level and pressure after a postulated 80 cm² bottom break in Forsmark 3. The maximum break flow is 500 kg/s. Adapted from *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asca-Atom and ES-Konsult AB, 1985

is somewhere between those described above. At first the response is similar to that of a bottom break with a rapid drop in water level while pressure is maintained. Once the nozzle through which the water is escaping has been uncovered, the continued outflow occurs in the steam phase. The pressure then decreases in the same way as for a steam line break.

Calculations for Forsmark 3 show that for a *feedwater line break* with a

maximum break flow of 2400 kg/s the peak clad temperature will only slightly exceed the saturation temperature, if two (of four) auxiliary feed-water subsystems and two (of four) low-pressure injection systems are assumed to operate (904).

For a *low-pressure injection line break*, assuming the same emergency core cooling efficiency as in the previous case, the calculations predict that the top of the core will be temporarily uncovered, before the reactor pressure has decreased sufficiently for the low-pressure injection system to start operation and reflood the core. The temporary core cooling deficiency will cause a minor heat-up of the core with a peak clad temperature of less than 600°C.

The characteristic variation of the reactor pressure and water level can be used to diagnose the type of LOCA from the control room where only the event symptoms can be observed. A difficulty lies in the fact that the indicated water level can deviate essentially from the real level, for example during rapid depressurization or when the main recirculation pumps are in operation.

9.5 LOCA in Pressurized Water Reactors

When analysing LOCA in pressurized water reactors, it is useful to differentiate between large LOCA, which are characterized by a break flow area corresponding to a diameter of at least 250 mm, medium LOCA (80–250 mm) and small LOCA (10–80 mm). In order to replace the lost coolant, one or more emergency core cooling systems, i.e. high-head safety injection, accumulators and low-head safety injection are used (8.2.5). The high- and low-pressure systems are actuated by a signal indicating safety injection, while the accumulators start to supply water as soon as the reactor pressure drops to below about 4 MPa. Once the injection phase is terminated, manual realignment to recirculation for long-term decay heat removal is carried out.

9.5.1 Large LOCA

The design basis accident is initiated by an assumed guillotine break in an inlet coolant pipe (“cold leg”) in a main coolant loop. The sequence of events can be divided into four phases:

- Blowdown*, characterized by rapid depressurization and intense break flow for 20–40 seconds.
- Refill*, which occurs when the break flow stagnates and the supplied water begins to fill the reactor vessel. During this period the core is filled with steam, and cooling deteriorates, causing the clad temperature to rise rapidly.

- Reflood*, which is defined as starting when the water level reaches the lower edge of the core. During this period, the maximum clad temperature is reached, 1–2 minutes after the initial break.
- Long-term cooling* which starts when the clad temperature has dropped to normal values. Long-term cooling continues as long as necessary for the core to be accessible for the removal of fuel, after which repair and maintenance work can be started.

The break initiates reactor scram and safety injection on a signal indicating low pressure in the pressurizer or high pressure in the containment. Within 10–25 seconds, the pressure is low enough for the accumulators to inject water. The low-head safety injection system begins to pump water into the reactor after 20–40 seconds. The accumulator tanks are emptied after about 50–100 seconds. The low-head safety injection system continues to supply water until the storage tank with borated water is almost empty. This is predicted to occur after about 20 minutes. The reactor operator must then realign the low-head safety injection system to recirculate water from the containment sump via heat exchangers in the residual heat removal system (Fig. 8.9).

A schematic diagram of the system pressure and water level in the reactor pressure vessel is shown in Fig. 9.5. During the blowdown phase, the pressure falls rapidly at first, until saturation pressure is attained, when the water begins to boil violently and the break flow is limited. The blowdown phase ceases after about 15 seconds when the pressure levels in the primary system and the reactor containment are equalized at 0.4–0.5 MPa and the flow ceases. Prior to this the accumulators are actuated.

During the blowdown phase some of the injected water can be prevented from reaching the core by a reverse flow in the downcomer, i.e. the annulus between the reactor vessel and the moderator tank (see Fig. 5.1). This is known as *bypass*. Part of the injected water then escapes directly through the break.

The vessel is refilled and the core reflooded first by water from the accumulators and then from the low-head safety injection system. During the refill and reflood phases there is no bypass, but the water meets resistance from the steam in the core which must be forced away before the water level can rise. This *steam blockage* is most severe when the break is located between the main coolant pump and the steam generator, since the flow resistance for the steam which has to be forced away is then at its greatest.

Figure 9.5 also shows the maximum clad temperature for the hottest fuel rod, calculated with a licensing model, i.e. with conservative assumptions. The critical heat flux is reached very rapidly during the blowdown phase. When the water starts to boil, the rod is effectively cooled (“quenched”) by a violent flow of water and steam, and the clad temperature passes a maximum. When the core starts to uncover, cooling deteriorates again until

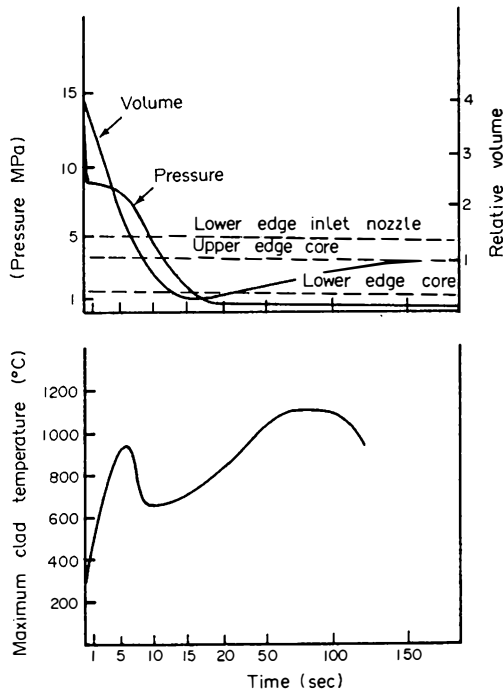


FIG. 9.5. Calculated water level, pressure and clad temperature (licensing model) for DBA-LOCA in a pressurized water reactor.

the rods are rewetted during the reflood phase and the clad temperature passes a second maximum.

Experiments in the LOFT reactor in the USA have shown that rewet occurs already in the blowdown phase if the main coolant pumps are in operation (906). However, according to the licensing requirements, loss of power to the main coolant pumps is assumed to occur at the moment of break. Therefore, no credit is allowed for rewetting during the blowdown phase in current licensing calculations.

9.5.2 Small and medium LOCA

In contrast to the large LOCA where the reactor vessel is rapidly emptied and refilled, small and medium LOCA are characterized by a slower drop in the water level which results in core uncover only if make-up water is unavailable or as a consequence of operator error. In typical cases, reactor isolation, scram and safety injection are initiated within 20–60 seconds (depending on the size of the break) in response to signals indicating high

containment pressure, low reactor pressure or low water level in the pressurizer. The main coolant pumps are stopped and the auxiliary feedwater system automatically taken into operation.

The core is cooled by natural circulation, first in the water phase and then, as the pressure falls to saturation level, in a two-phase mixture of steam and water. If and when phase separation occurs and the water level falls below the outlet nozzles of the reactor vessel (see Fig. 5.1), steam escapes to the steam generators and condenses there. The condensate flows back to the reactor vessel in the opposite direction ("reflux condenser mode"). Cooling is very effective in this case. The different flow regimes have been demonstrated in large-scale thermohydraulic experiments.

The pressure falls at such a rate that the accumulators start to inject after about 10–15 minutes. The pressure is eventually stabilized at about 1 MPa. The low-head injection system can then pump water into the primary circuit. The pumping continues until the storage tank begins to empty. The operator then has plenty of time to realign the low-head injection system for recirculation. The break flow ceases when the pressures in the primary system and the reactor containment have equalized.

During *small LOCA*, break area $< 50 \text{ cm}^2$, the pressure falls more slowly than in the previous case, stabilizing at a higher pressure than that at which the low-head safety injection system begins to operate. The reactor operator must then reduce the temperature and pressure in order to use the low-head safety injection system. This is normally achieved with the help of the steam generators, the auxiliary feedwater system and by opening the relief valves on the secondary side. Alternatively, the operator can manually break the isolation of a loop in the main feedwater system and use the turbine condenser as a heat sink.

The phenomenological difference between small and medium LOCA is that in the latter the break flow is sufficiently large to remove the decay heat generated in the core. During a small LOCA, an additional heat sink is required, namely discharging steam on the secondary side or dumping steam to the turbine condenser. An alternative method of reducing the reactor pressure is to open and close the electrically driven pressure relief valves in the pressurizer. What is in fact a small LOCA is then transformed into a medium LOCA.

A schematic diagram of levels and pressures at different break sizes is presented in Fig. 9.6. In all cases, two (of four) high-head safety injection and four (of eight) borated water storage tanks are assumed to be available. The accumulators and low-head safety injection system are not credited. The calculations refer to a 1300 MWel PWR of West German (KWU) design, but are also valid, in principle, for other types of pressurized water reactors.

With break areas smaller than about 50 cm^2 , the level in the reactor vessel stays above the outlet nozzles for the main coolant. The time during which

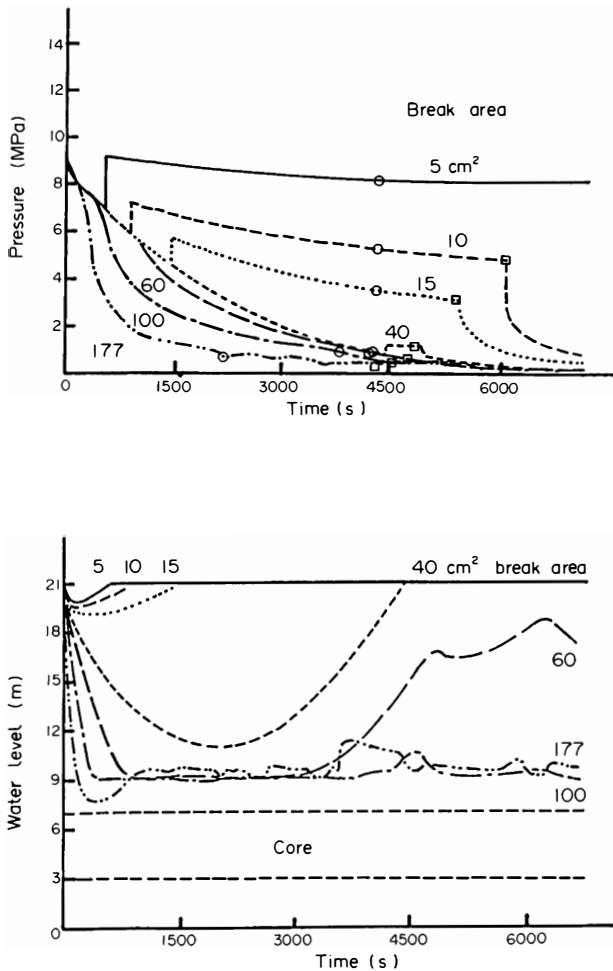


FIG. 9.6. Calculated pressure and water level for small and medium LOCA in pressurized water reactors. From D Hein and H Watzinger, *Small Break LOCAs. Analysis, Control and Experimental Results*, Paper IAEA-CN-39/30, International Conference on Current Nuclear Power Plant Safety Issues, Stockholm, 20–24 October 1980

the level drops decreases with the break area. The water flow from (half) the high-head safety injection system is sufficient to compensate for the break flow and to refill the primary system. The system pressure first falls rapidly to the saturation point, after which it follows the saturation line until the reactor vessel is completely filled. At this point there is a sudden pressure increase corresponding to the head of the high-head pumps.

For medium break areas ($> 50 \text{ cm}^2$) the level rapidly falls to the level of the outlet nozzles and to an even lower level for areas $> 100 \text{ cm}^2$ with a risk

of core uncover for large break areas. The system pressure follows the saturation line and the two high-head pumps are not sufficient to refill the reactor system.

The circles in Fig. 9.6 indicate the points in time when the temperature in the primary system reaches 175°C, corresponding to the pressure at which the low-head pumps begin to supply flow. At this point the low-head safety injection system can pump water into the reactor if the system pressure is lower than the maximum pump head.

The squares in Fig. 9.6 indicate the point in time when the storage tanks are emptied. Well in advance of this, the operator should have realigned to recirculation with the low-head safety injection system, or with the high-head safety injection system if the pressure is still high. If the switch-over fails in the latter case, the pressure will rapidly drop to the saturation pressure which corresponds to the reactor coolant temperature. The low-head safety injection system can then supply water to the primary system and the reactor containment.

Certain break and leak locations can cause an abnormal water distribution in the primary system. If a pressure relief valve at the top on the pressurizer gets stuck in the open position and no mitigating action is taken, the pressurizer will fill up with water (“go solid”) within a few minutes. At the same time there may be free surfaces elsewhere in the system, for example in the reactor vessel. Since the pressurized water reactor has no direct indication of the water level in the reactor vessel, the operator may be led, by the rising level in the pressurizer, to think that the system is being overfilled. This occurred during the initial stage of the accident at Three Mile Island-2, further described in section 13.5.

9.6 Transients in Boiling Water Reactors

Transient is the overall term used for an abnormal event—with the exception of LOCA—occurring during power operation or after shutdown. The reactor is designed to control such events without exceeding the bounding values of essential plant variables such as clad temperature, rod surface heat flux and reactor pressure. Deterministic safety analysis is concerned with predicting the processes involved and verifying that the safety requirements are satisfied. Some typical transients which can occur as a result of malfunction of the normal operating and control systems are described in this section. The description largely refers to internal recirculation pump reactors of the Forsmark 3 type. The detailed course of events can differ in other types of boiling water reactors due to differences in the emergency core cooling system design and function.

9.6.1 Malfunction of the reactivity control system

The withdrawal of control rods is normally achieved with the help of the plant computer in accordance with a predetermined sequence. The participation of the operator is limited to commands for control rod insertion or withdrawal. The position of the control rods in the core is presented on a video display unit in the control room. At power levels above about 50% of full power, an interlock unit prevents a control rod from being withdrawn by more than 5% of its length. Calculations show that a 5% movement of any control rod results in an insignificant transient.

At low power during reactor start-up, malfunction of the control rod operating system or operator error can cause the reactivity and power level to increase more rapidly than intended. This event is known as *uncontrolled withdrawal* of control rods and classified as a category H₂ plant condition (cf 9.1.4). In the limiting case, the most reactive control rod group is assumed to be withdrawn at maximum rate in the just critical reactor at hot standby conditions. Calculations show that the reactivity transient will be limited by the Doppler effect. No fuel damage would occur even if scram were to fail.

A hypothetical event, category H₄, where a control rod with a large reactivity worth is suddenly ejected from a fully inserted position is known as the *control rod drop accident*. The potential causes of a control rod drop accident are (see Fig. 4.2):

- Failure to connect the piston tube to the control rod shaft upon control rod drive service. If the control rod gets stuck in its fully inserted position and the piston tube is subsequently withdrawn, the stuck rod could fall by gravity.
- Inoperative drive piston tube latches. If this occurs when the piston tube and the drive nut are separated shortly after scram, then both the control rod and the piston tube will drop.
- Fracture of the control rod drive casing. This event might result in both the control rod and piston tube being ejected from the core.

Several design and administrative measures have been taken to prevent a control rod drop accident. Nevertheless, it is postulated to occur and is analysed in safety reports as the limiting reactivity insertion accident (RIA).

In the analysis of the control rod drop accident for Forsmark 3, the initial conditions were assumed to be those corresponding to hot critical standby with full recirculation flow and 50% of the control rods inserted. The maximum worth control rod is then assumed to drop. Calculations show that the reactor goes prompt critical within a second (905). The initial power increase is limited by the Doppler effect. The IRM detectors (4.5.1) actuate scram after about a second. The transient is terminated after about 5 seconds by the combined action of the inherent Doppler effect and the automatic scram.

The peak fuel pellet enthalpy is estimated at 266 cal/g UO_2 , to be compared with the bounding value of 280 cal/g UO_2 (see 9.2.3). The number of fuel channels around the dropped control rod for which the calculated critical power ratio (cf 9.2.2) is temporarily less than 1, i.e. for which dryout is expected to occur, is 24 as compared to a total of 700 channels. However, the individual fuel rod will not experience dryout until the enthalpy exceeds 170 cal/g UO_2 . The number of fuel rods predicted to reach this value is 128 or 0.3% of the total number of rods.

With other initial conditions, the transient will be less severe. At power operation the control rod worth will be less and the feedback effects stronger.

9.6.2 Malfunction of the main recirculation system

The main recirculation flow is regulated by the power control system (4.5.4). Failure of the power regulator leading to an increase in pump speed and an *increase in mass flow*, will result in an increase of the reactor power. Scram and pump coast-down are initiated in response to a signal indicating high neutron flux. Analyses show a safe margin to dryout. This transient is classified as category H_2 .

Failure of the power regulator, leading to a reduction of pump speed and hence a *decrease in mass flow*, results in an insignificant transient. The reactor power will stabilize at a level which corresponds to the lower pump speed without the actuation of scram.

Trip or seizure of a main recirculation pump results in a sudden decrease of the coolant flow which is partly compensated for by the power regulator attempting to increase the speed of the other pumps. In Forsmark type reactors, it is most likely that *two* pumps will be tripped, due to the loss of a busbar feeding two pumps. This results in a moderate reduction in the coolant flow and does not lead to scram. *Trip of all recirculation pumps* is initiated by the loss of auxiliary power and is classified as a category H_3 or H_4 event depending on the particular circumstances. Total loss of auxiliary power implies simultaneous loss of both the turbine-generator and offsite power. Safe plant shutdown is not impaired, since all equipment required for safe shutdown is fed from the emergency diesel-generator powered buses.

The loss of auxiliary power causes the recirculation pumps to coast down which immediately affects the core cooling conditions. The coolant flow is reduced to a minimum in a few seconds. The void content in the core increases and reduces the neutron flux and the fission power, due to the negative void coefficient of reactivity (cf 3.3.4). However, because of the thermal inertia of the fuel rod, the change of the surface heat flux is delayed relative to the change of the coolant flow and the fission power. Therefore, the dryout margin drops sharply within 2–3 seconds. As the surface heat flux

decreases, the dryout margin rapidly increases again. Typical calculational results are shown in Fig. 9.7

The pump trip transient will be plant-specific with regard to initial conditions, pump inertia, steam separator pressure drop, etc. The calculated

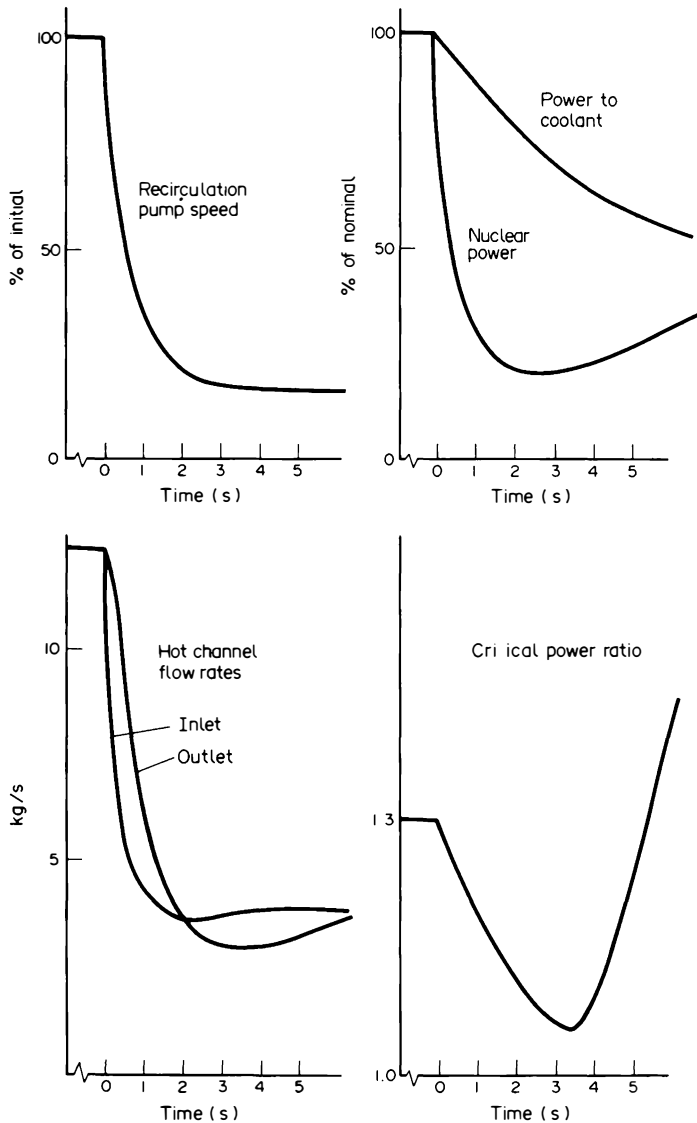


FIG. 9.7. Pump trip transient in an internal pump boiling water reactor. From O Nylund et al, *Post Dryout in Connection with BWR Main Circulation Pump Trip*, Paper to the European Two-Phase Flow Group Meeting, Munich, 10–13 June 1986

minimum critical power ratio (MCPR) may even fall below 1 in some cases. The dryout conditions are, however, expected to exist only for a short time before rewetting occurs. Due to the short duration of dryout and the relatively low peak clad temperatures, it is unlikely that the integrity of the fuel rods will be adversely affected.

In general, pump trip transients are of less interest in external pump reactors. This is due to the larger inertia of the rotating parts in the external recirculation pumps. The coast-down will be slower than for internal recirculation pumps which means that the thermal margins will be larger. This has been clearly demonstrated by experiments in the FIX loop at Studsvik, Sweden (see Fig. 15.2).

9.6.3 Malfunction of the feedwater system

Failure of the feedwater control system, inadvertent closure of an isolation valve in a feedwater line, feedwater pump trip or loss of auxiliary power causes partial or complete *loss of feedwater*. This leads to low water level in the reactor vessel, which initiates fast runback of the main recirculation pumps, reactor scram, and start-up of the auxiliary feedwater pumps.

Failure of the feedwater control system resulting in a *flow increase* halts the feedwater flow when a high water level set-point in the reactor vessel is reached. This actuates closure of the main steam line and feedwater line isolation valves, scram and opening of the pressure relief valves. The water level is then regulated by the auxiliary feedwater system.

An increase in feedwater flow or a drop in feedwater temperature leads to *increased subcooling* of the reactor coolant and hence to increased reactivity. Moderate changes, such as through the unintentional start-up of an auxiliary feedwater pump or through loss of a feedwater preheater, will not have any significant effect on the reactor and will not result in scram.

Disturbances of the feedwater supply are relatively common occurrences (category H₂). Detailed analyses of the above-mentioned and other cases show that if the safety systems operate as intended, the pressure in the reactor is kept within acceptable limits and the core will remain covered and cooled during the entire transient.

9.6.4 Malfunction affecting the steam flow

A sudden change in the live steam flow to the turbine will affect the reactor in two ways:

- a decrease of the steam flow results in an increase of the reactor pressure and a decrease on the void content in the core due to steam compaction and condensation (since the saturation enthalpy increases with pressure);

—an increase of the steam flow results in a reactivity decrease, a decrease in the water inventory and a pressure drop which can cause steam flashing and level swelling.

Both kinds of events are characterized by rapid transients in the reactor.

The steam flow to the turbine will be completely interrupted upon closure of the main steam line isolation valves (MSIV). The reactor pressure increases since the steam flow through the safety/relief valves is delayed. The reactor power increases due to the positive pressure coefficient of reactivity (cf 3.3.4). A transient of this kind is known as a *pressure transient*. In pressure transients it is important to counteract the void collapse by rapidly reducing the speed of the main recirculation pumps.

Inadvertent MSIV closure is considered as an H_2 event if offsite power is available throughout the transient and as an H_3 event if offsite power is not available. The signals that initiate MSIV closure also actuate reactor scram, fast recirculation pump runback, and pressure relief valve opening. Calculations show (904) that the maximum allowable pressure (cf 9.2.4) will not be exceeded. The minimum critical power ratio (MCPR) may temporarily fall below 1 for the hottest channel(s) shortly after closure of the MSIV valves. Dryout conditions will, however, exist only for a few seconds before the affected fuel rods are rewetted. No fuel damage is expected to occur.

Results of a sample calculation for Forsmark 3 are shown in Fig. 9.8. The initial reactor power is assumed to be 102% and the coolant flow 90% of their nominal values, when steam blockage occurs at time zero. Figure 9.8 refers to the average fuel channel and shows the rapid power peaking and the delayed heat transfer to the coolant. The maximum pressure, 8.42 MPa, occurs 3.5 seconds after the initial event and is well below the bounding value, 9.35 MPa.

Figure 9.9 shows the calculated MCPR for the hot channel. The “local” MCPR is defined as the factor by which the channel power should be multiplied to attain dryout in the actual position. Dryout conditions are obtained in the middle upper part of the hot channel. The corresponding time to dryout and rewet as well as the clad temperature are shown in Fig. 9.10.

In external pump reactors, the pump speed reduction will not be as fast as in the internal pump reactor, because of the larger pump inertia. This results in a higher power peak and, in spite of the slower decrease of the coolant flow, dryout will occur for a lower initial (stationary) channel power, i.e. for a higher initial critical power ratio. This is illustrated in Fig. 9.11.

It is interesting to note that the thermal margins are larger in the internal pump reactor than in the external pump reactor for pressure transients, whilst the opposite is true of pump trip transients (cf 9.6.2).

The *design basis pressure transients* for the reactor coolant pressure boundary comprise a set of initiating events including MSIV closure in combination with reduced capacity of the pressure relief system or failure of the

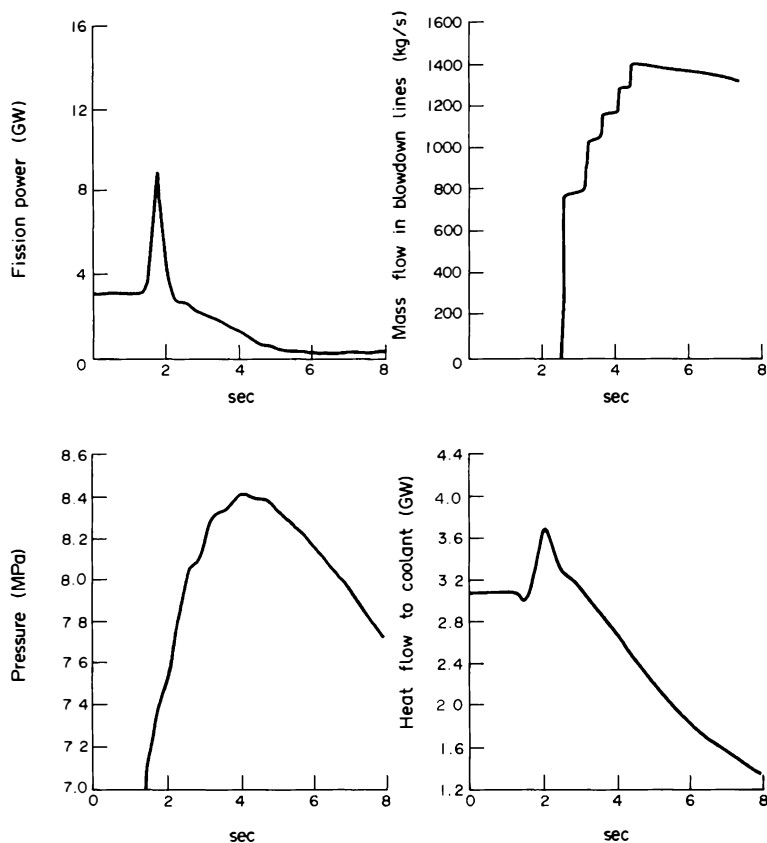


FIG. 9.8. Pressure transient upon MSIV closure in Forsmark 3

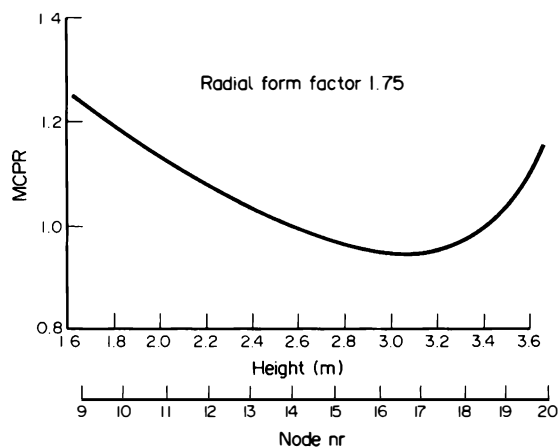


FIG. 9.9. Minimum critical power ratio (MCPR) at various axial positions in the hot channel

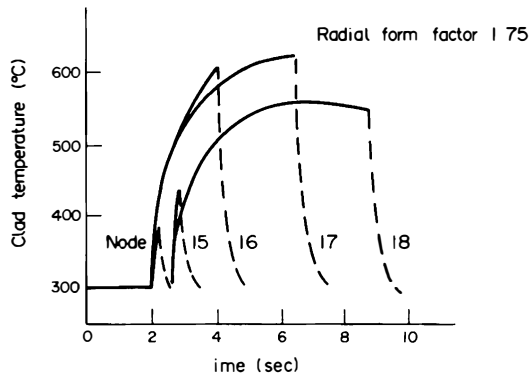


FIG. 9.10. Clad temperature versus time for various positions in the hot channel

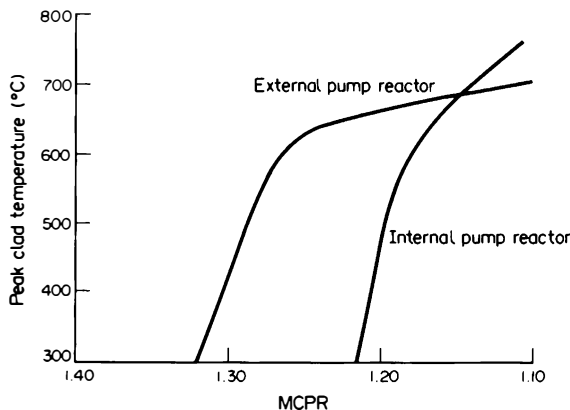


FIG. 9.11. Calculated maximum clad temperature versus initial minimum critical power ratio (IMCPR) during pressure transients in external and internal pump BWRs

reactor shutdown system. These transients are classified as H_4 events. The shutdown system is assumed to fail either by failure of the hydraulic scram or, if scram is effective, by failure of the fast recirculation pump runback. In the former case, which is an example of a class of transients called *anticipated transients without scram* (ATWS), the reactor power is assumed to be reduced initially by fast recirculation pump runback and eventually by fine-motion control rod insertion. Calculations show that the maximum allowable reactor pressure will not be exceeded.

Malfunction of the turbine system can also result in rapid reduction and even complete interruption of the steam flow to the turbine. The most severe pressure transient is obtained when the turbine stop valves close and the bypass valves fail to open. This event is known as *turbine trip without*

bypass and is classified as an H₂ event if offsite power is available during the transient and as an H₃ event if offsite power is not available. The reactor response is similar to that obtained after MSIV closure.

If the turbine pressure regulator erroneously demands pressure reduction, this will result in *increased steam flow* due to opening of the governor valves and possibly also the bypass valves. The reduced pressure causes a reactivity decrease and swelling of the water level in the reactor, thereby initiating reactor scram, pump coast-down, closure of the steam line isolation valves and opening of the pressure relief valves. The pressure decrease and level swelling will cease once the main steam lines are isolated.

If a pressure relief valve should open spuriously during operation, steam will be discharged into the condensation pool. The sudden increase of the steam flow will be counteracted by the pressure and power control systems. The inadvertent opening of more than two valves cannot occur as a result of a single failure in the electrical equipment.

9.6.5 Malfunction of the residual heat removal system

When the turbine condenser is unavailable as a heat sink, the excess steam is normally discharged into the condensation pool in order to maintain a constant reactor pressure of 7 MPa. The condensation pool is cooled by the safety-grade sea water cooling system (cf 8.1.8). Make-up coolant is taken from the feedwater system or the auxiliary feedwater system. Controlled depressurization is carried out by the pressure relief system, until the shutdown cooling system can take over the cooling function. If this system is unavailable, heat is removed via the condensation pool. This may be performed over a long period of time.

The discharge of decay heat and stored energy in the fuel causes an initial rise in temperature in the condensation pool. The capacity of the pool cooling system is proportional to the temperature difference between the pool and the sea and is therefore low initially. After a few hours, the cooling power is greater than the decay power (cf Fig. 8.10). The pool temperature reaches a maximum and decreases as the decay power decreases.

Some calculated results for Forsmark 3 are shown in Fig. 9.12. The maximum temperature, 54°C, is reached after about 4 hours. If only two of four cooling subsystems are operating, it takes about 12 hours to reach the maximum temperature, 72°C. In the event on total unavailability of the pool cooling systems, the water temperature rises to 100°C in about 7 hours. In this case, cold water must be supplied from other sources.

Figure 9.12 also illustrates pool cooling with 1–4 hours' delay. If the cooling systems are realigned before the temperature reaches its maximum in the reference case (100% cooling power without delay), i.e. as long as the decay power is greater than the cooling power, the temperature increase will only be a few degrees larger than in the reference case.

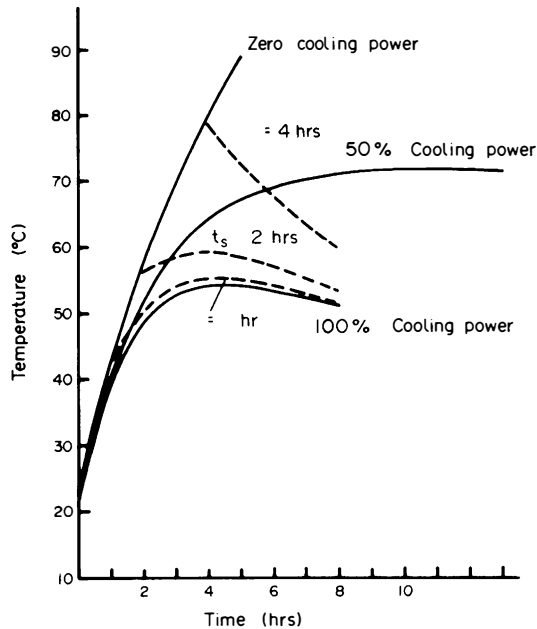


FIG. 9.12. Temperature in the condensation pool during insufficient decay heat removal. From *Handbook of Process Relations during Disturbances in Swedish Boiling Water Reactors*, AB Asea-Atom and ES-Konsult AB, 1985

9.6.6 Malfunction of the auxiliary power supply system

During normal operation the plant's auxiliary power supply network is connected to the main generator via station transformers (see 4.6.2). During reactor scram or *turbine trip*, the generator is disconnected from the external grid and the onsite auxiliary power supply grid. Auxiliary power is then supplied from the external grid via either the main transformer and the station transformers or the start-up transformers (see 4.6).

During *load rejection* due to malfunction of the external grid, the turbine speed increases rapidly. The turbine regulator then closes the governor valves and opens the bypass valves. At the same time the generator is disconnected from the external grid and the reactor power regulator decreases the main recirculation pump speed to about 20% of full speed which corresponds to about 60% of full power. The turbine governor valves are opened again and enough steam is supplied to the turbine for *house load operation*. The pressure increase in the reactor is normally not high enough to initiate scram. However, the processes in the turbine plant can lead to turbine trip and unsuccessful transition to house load operation. Load rejection transients belong to category H₂.

If the external grid is lost and the transition to house load operation is

unsuccessful, the plant is connected to the start-up grid which supplies auxiliary power to the start-up transformers (Fig. 4.12). Failure of this connection results in *loss of auxiliary power*, category H₃. Power supply to the feedwater pumps and main recirculation pumps is then interrupted. The loss of the cooling water pumps to the turbine condenser leads to a deterioration of the condenser vacuum, which causes dump blockage.

During loss of auxiliary power, the power supply for the operation of safety-related systems is obtained from the diesel-motor driven emergency power supply system. Equipment requiring a continuous power supply or whose operation cannot be delayed until the start-up of the diesel system, is fed from independent battery grids. Complete *station blackout*, i.e. loss of the external grid, turbine generator, start-up grid and diesel generators, is considered as a category H₄ event. The likelihood of an extended blackout is very small.

9.7 Transients in Pressurized Water Reactors

In this section some typical transients in pressurized water reactors are reviewed. In order to facilitate the comparison with boiling water reactors, the description is structured in the same way as in the previous section.

9.7.1 Malfunction of the reactivity control system

Malfunction of the reactivity control system is generally classified as an H₂ event, i.e. an occurrence of moderate frequency which in the worst case leads to scram but which allows more or less immediate restart. These events are not expected to result in fuel damage or reactor system overpressure.

Uncontrolled withdrawal of control rods at power operation results in an increase of the heat rate in the core. Since the rate of heat removal remains constant, the coolant temperature will increase. Unless terminated, this power mismatch and resultant coolant temperature rise will eventually result in DNB. Therefore, the reactor protection systems will initiate scram in response to signals indicating high neutron flux, high temperature increase over the core, high pressure or high water level in the pressurizer. The conditions for scram are set so that the margin to critical heat flux is at least 30%, i.e. $DNBR > 1.30$, which gives a safe margin to clad damage.

Figure 9.13 shows the calculated neutron flux, reactor pressure, coolant temperature and DNBR during a reactivity transient caused by the uncontrolled withdrawal of two control rod banks at full power. The rate of reactivity insertion is 75 pcm/sec. Scram is initiated after 1.9 seconds in response to a signal indicating high neutron flux. Since this time is short in relation to the time constant of the fuel and the moderator, the temperature change in the moderator will be small. The minimum DNBR during the transient is estimated at 1.37.

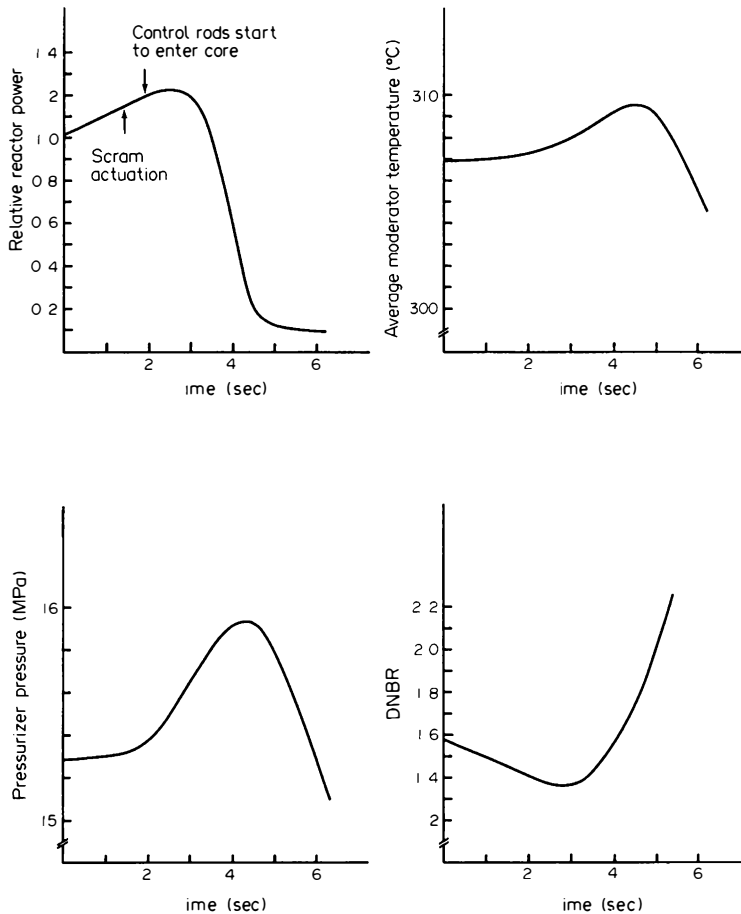


FIG. 9.13. Uncontrolled withdrawal of control rods from full power in a pressurized water reactor. The transient is terminated by reactor scram. From *Ringhals 3/4 Final Safety Analysis Report*, Swedish State Power Board, 1984

Uncontrolled withdrawal of control rods during the *start-up procedure* can lead to a superprompt transient. Since the reactor is initially slightly subcritical and essentially at zero power, enough reactivity can be inserted to exceed prompt critical before the power level rises to a high enough level to cause scram. The transient is terminated by the prompt negative Doppler effect as illustrated in Fig. 9.14. Although the peak power is nearly ten times full power, the power burst is so narrow that the energy release in the fuel is not sufficient to cause damage.

The mechanical failure of a control rod mechanism housing could result in the ejection of a rod cluster control assembly and drive shaft. This *control rod ejection* accident is classified as an H_4 event. It leads to a rapid reactivity

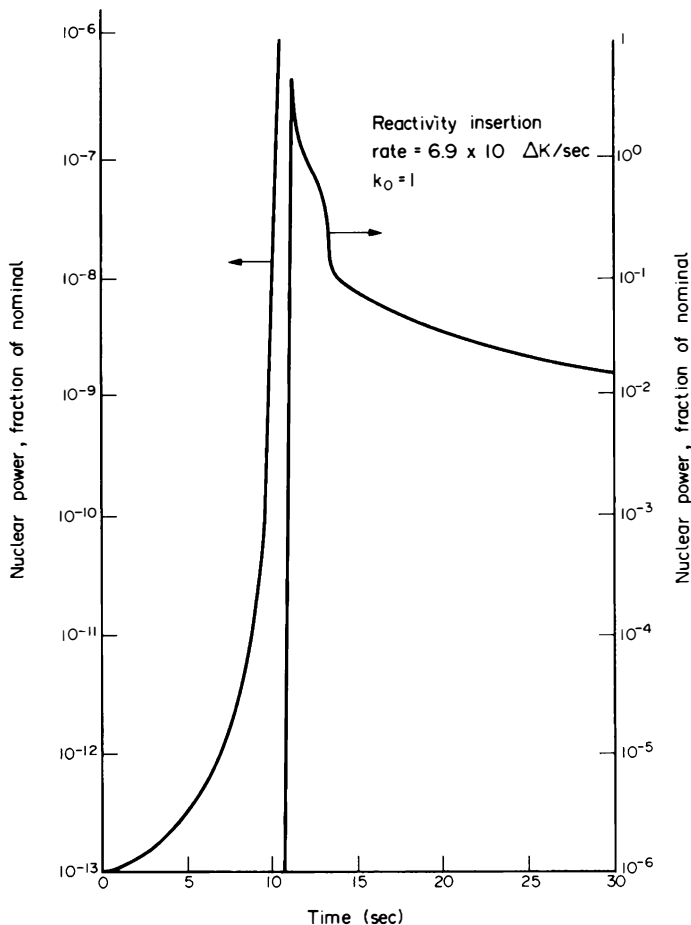


FIG. 9.14. Uncontrolled withdrawal of control rods from a subcritical condition in a pressurized water reactor. The transient is terminated by the Doppler effect. From *Ringhals 3/4 Final Safety Analysis Report*, Swedish State Power Board, 1984

insertion together with an adverse power distribution, and possibly to localized fuel rod damage. The transient will be terminated by the combined action of the Doppler effect and scram. The relevant criterion is that the fuel pellet enthalpy during the power burst should not exceed 280 cal/g UO_2 .

The rod ejection transient analysis is performed in two stages, first an average core calculation and then a hot region calculation, and for various hot zero power and full power cases, ejected rod worths and Doppler reactivity coefficients. The results indicate that safety limits for fuel damage are not exceeded.

During the *uncontrolled insertion* of a control rod, which can occur if the power supply to the control rod drive mechanism is lost, the reactor power decreases and the form factor (3.3.2) increases. If no countermeasure is adopted, the power control system will seek to increase power which will then lead to a reduction in the margin for critical heat flux. A “dropped” control rod therefore actuates reduction of the turbine power and blockage of automatic control rod withdrawal.

Slow reactivity control is normally carried out by the chemical and volume control system (5.4.2) which is manually controlled from the control room. During *inadvertent dilution*, the boron concentration in the reactor coolant decreases which increases reactivity. At power operation, scram is initiated in response to a signal indicating high power and high moderator temperature. In order to prevent dilution, the manual procedures are carefully regulated. The amount of unborated water which can be delivered to the reactor is limited, as is the make-up rate, so that the operator has sufficient time to correct the situation in a safe and orderly manner.

9.7.2 Malfunction of the reactor coolant system

Swedish pressurized water reactors have three main coolant loops (5.2.1). The immediate consequence of a *decrease in the coolant flow*, e.g. due to loss of power to a main coolant pump or a mechanical failure of the pump, will be an increase in the coolant temperature. If the reactor power is not rapidly decreased, the critical heat flux may be exceeded. Scram is therefore initiated by a signal indicating reduced coolant flow. Calculations show that if scram is actuated once the coolant flow has fallen to about 80% of full flow, the DNBR will not be below the minimum permissible 1.30. Events resulting in the partial loss of coolant flow are classified as category H₂.

The simultaneous loss of power to all main coolant pumps is the most severe case of coolant flow reduction. This event belongs to category H₃. Scram is initiated by signals indicating a reduced coolant flow and a large temperature increase over the core. The pressure increase causes the relief valves in the pressurizer to open. The coolant flow is initially maintained by the inertia of the coolant and the rotating parts of the main coolant pumps and then by natural circulation. The DNBR is calculated not to fall below 1.30 during the transient.

Operating the reactor with an *inactive loop* will result in reverse flow in the inactive loop since there are no check valves or isolation valves in the loops. If the reactor is operated at (reduced) power with an inactive loop, the coolant temperature will be lower in the inactive loop than in the other loops. During restart of the inactive coolant pump, “cold” water will be supplied to the reactor and result in a sudden reactivity increase. An analysis of this transient shows that the corresponding power increase will not initiate scram and that the DNB margin is satisfactory.

9.7.3 Malfunction of the feedwater system

A *reduction* of the feedwater flow, e.g. due to pump trip, will result in reactor scram in response to a signal indicating low water level in the steam generators. The auxiliary feedwater system (8.2.4) will start automatically. Steam will be dumped to the turbine condenser. If steam bypass is not possible, the steam will be discharged through the safety valves in the main steam lines.

Calculations show that failure of the main feedwater system under the circumstances described above will lead to an initial rise in the coolant temperature and in the water level in the pressurizer. However, the pressurizer will not be filled up so that no coolant will be lost. Although the water level in the steam generators will fall, it will not be enough to prevent decay heat removal. Hence, no fuel damage will occur.

Inadvertent *increase* of the feedwater flow will lead to an overpower transient which will be terminated by reactor scram without the DNBR falling below the safe limit. This also applies to transients resulting from sudden reduction in the feedwater inlet temperature. Events initiated by disturbances in the feedwater supply belong to category H₂.

9.7.4 Malfunction affecting the steam flow

A small increase of the steam flow and decrease of the steam pressure is interpreted by the reactor power regulator as an increase of the load demand. The power regulator will therefore seek to increase the reactor power.

A large increase of the steam flow occurs in the event of a main steam line break (category H₄). Safety injection, scram, closure of the main steam line isolation valves and start-up of the auxiliary feedwater pumps are then initiated. The progression of the transient will depend on whether the break has occurred inside or outside the reactor containment.

In the event of a break *outside* the reactor containment, the break is isolated by closure of the isolation valves. Since the main feedwater system is disconnected when safety injection is actuated, the decay heat is first removed by safety injection and the discharge of steam through the pressurizer safety valves, and in the long run by the auxiliary feedwater system and the discharge of steam through the steam line safety valves. Safety injection with borated water guarantees that the reactor will not become critical when the reactor coolant temperature falls, even if a scram group should fail.

In the event of a break *inside* the reactor containment, the reactor operator must stop the supply of feedwater to the steam generator affected by the break. Otherwise, the reactor power will be transferred to the containment through the damaged steam generator and result in high pressure and high

temperature in the containment. Once the damaged steam generator is isolated, the long-term decay heat removal takes place through the undamaged steam generators, the auxiliary feedwater system and the pressure relief valves on the secondary side.

The inadvertent opening of a safety valve on the secondary side is equivalent to a (small) steam line break. The same applies to a safety valve getting stuck in the open position. These events are classified as category H_2 and are overcome without scram.

9.7.5 Malfunction of the turbine system

Failure in the turbine system or in the external grid can result in the disconnection of the plant from the grid, i.e. *load rejection* (category H_2). The steam flow to the turbine is then intercepted by the closure of the turbine stop valves. At the same time the bypass valves are opened for steam dump directly to the condenser. The reactor power and turbine control valves are regulated so that a power level corresponding to the needs of the plant is reached (house load operation). If the transition to house load operation fails, reactor scram is initiated, usually in response to a signal indicating high water level in the steam generators.

During *turbine trip without steam bypass* (category H_2), reactor scram is initiated in response to signals from the turbine oil system pressure. The temperature and pressure increase in the primary system, which also actuates scram. The reactor pressure is relieved by the opening of the pressurizer safety valves. The pressure increases on the secondary side until the steam line safety valves open. The decay heat is removed by the discharged steam. Diagrams of the transient are shown in Fig. 9.15. The calculated DNB ratio is greater than 1.30 during the entire transient. A similar transient is obtained after malfunction of the turbine regulator causing the control valves to close inadvertently at full power.

The *inadvertent opening* of control valves or bypass valves results in an increase in the steam flow and a mismatch between the power supplied by the reactor and the power delivered to the turbine. Although this transient resembles that obtained during a steam line break, it generally does not involve scram or decrease in the DNB margin.

9.7.6 Loss of auxiliary power

During loss of auxiliary power and unsuccessful transition to house load operation (category H_2), scram and start-up of the diesel generators for power supply to safety-related equipment is initiated. Once the main coolant pumps stop functioning, the coolant flow through the core is maintained by natural circulation. Since the main condenser is not available due to the loss of condenser vacuum resulting from the loss of power, the decay heat

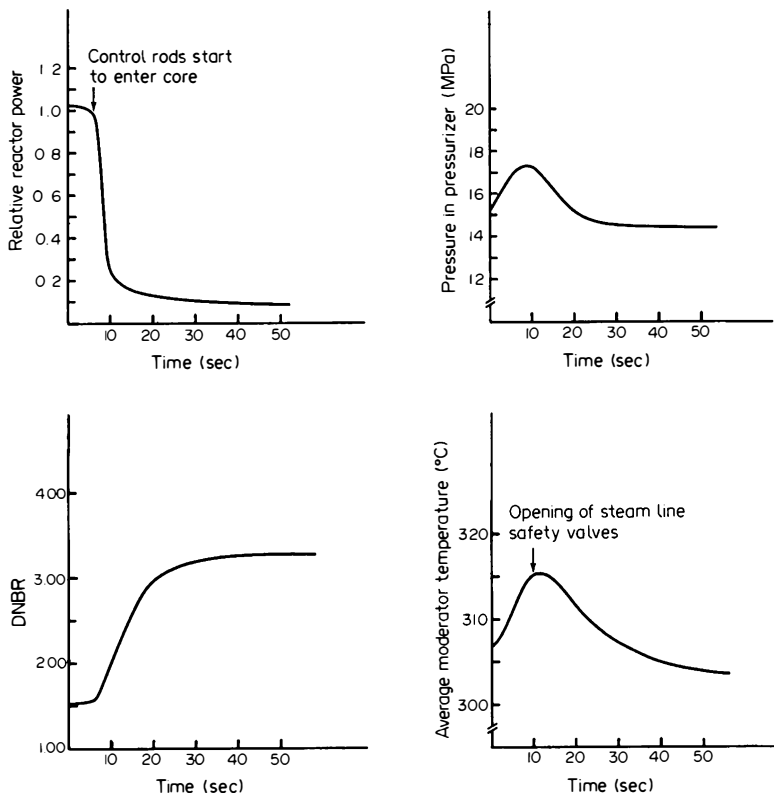


FIG. 9.15. Temperature and pressure during turbine trip without bypass in a pressurized water reactor. From *Ringhals 3/4 Final Safety Analysis Report*, Swedish State Power Board, 1984

is removed via the auxiliary feedwater system and the discharge of steam through the main steam line safety valves. Calculations show that DNBR will stay above 1.30 during the transient and that the set-point pressure for the pressurizer relief valves will not be reached. The reactor operator then reduces the pressure and temperature in the primary system until the residual heat removal system can be used. If the auxiliary feedwater system is not available, the decay heat can be removed by the injection of make-up coolant by the charging pumps and the discharge of steam through the pressurizer relief valves.

9.8 External Events

Deterministic safety analysis is mainly concerned with "internal" events anticipated or postulated to occur as a result of reactor faults, i.e. malfunction of the reactor's normal operating and control systems. The effect of

external events on the plant must also be considered, however. These events may be caused by natural phenomena such as strong wind, lightning, snow and ice, flooding or earthquake, or may be man-made such as aircraft crash, chemical explosion, sabotage, terrorist action and wartime action. External events are also usually taken to include fire and flooding in the plant.

9.8.1 Design requirements

The occurrence and extent of external events varies depending on the location of the power plant. Therefore, the requirements for protection against such events will be plant-specific. In the USA there are design criteria for extreme wind, ambient temperature, precipitation and water level, explosion and earthquake. In the Federal Republic of Germany, the reactor containment must be designed to withstand the impact of aircraft crashes.

In Great Britain, specific criteria were developed for the Sizewell B plant (906). The aim was to set the criteria so that the combination of the probability of the external hazard and the probability of subsequent failure to control the reactor would be consistent with the general criteria for the risk of a large uncontrolled release of radioactive substances (cf 7.1.2).

In Sweden, the Nuclear Power Inspectorate established general criteria for external events in the licensing of Forsmark 3 and Oskarshamn III. The meteorological, hydrological and seismological conditions are classified as "normal" and "extreme". Normal events comprise the worst events which can be assumed to occur during the lifetime of the plant. These events may supposedly occur at any time, i.e. during all operating conditions considered in the plant design. The design shall be such that normal events do not have any significant effect on the operation of the plant.

Extreme external events comprise the worst conditions which are physically possible at the site. If the probability of an extreme event is less than 10^{-5} per year, its effects need not be considered in the design process. Extreme external events shall be assumed to occur only during normal reactor operation. With the simultaneous occurrence of a single failure in a required component, the normal shutdown and cooling of the plant to the cold subcritical state as well as the maintenance of the reactor in this condition shall be possible.

According to the classification in section 9.1.4, normal external events belong to category H_3 and extreme external events to category H_4 .

External events which originate on the site include fire, missiles, dropped loads, and failure of pressurized systems which could result in pipe whip, jet impingement and local flooding. Design requirements for these events cannot, in general, be approached in the same way as those for natural phenomena.

9.8.2 Earthquake

Before a reactor is built, the seismic conditions at the site are determined. An earthquake is characterized by the maximum ground acceleration, the frequency spectrum and the duration. Its effect on the reactor plant is analysed with the methods of structural mechanics, usually by approximating the plant structure by a system of elastically connected mass nodes. The natural frequencies and vibratory modes of the plant are of special interest. If the natural frequencies are close to strong frequencies in the earthquake spectrum, the plant response will be amplified and lead to severe loads on plant structures.

In typical cases, the site has a low natural frequency and a high damping factor. The reactor containment has a medium frequency and damping while the primary system, which is anchored in the containment and therefore affected by ground movements via the containment, has a relatively small mass, a high natural frequency and a small damping. The response of the plant to an earthquake would therefore consist of the rapid shaking of the primary system superimposed on a displacement of the reactor containment with a frequency of about one period per second, which in turn would be superimposed on a slower rocking of the whole system in the ground.

The U.S. seismic criteria define normal values and extreme values for ground accelerations etc. These values are specific for each site. It must be shown that the plant can withstand an earthquake according to the normal values without incurring any damage, and an earthquake according to the extreme values without damage to essential safety-related equipment and without release of radioactive substances to the environment. The extreme values specify a design basis accident, known as the Safe Shutdown Earthquake (SSE).

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10

Probabilistic Safety Analysis

In deterministic safety analysis, the physical processes in the reactor are studied during fault conditions caused by malfunction of the reactor's normal operating and control systems. The safety systems are assumed to function according to the design intent. The analysis is not concerned with the probability of the fault conditions, nor with the possibility that the safety systems might not function as intended. If the safety systems do not operate effectively, the core may overheat, resulting in more or less severe core damage. At worst, the entire core or a large part of it will melt. In this chapter, core damage, core overheating and core melting are used synonymously to denote degraded core conditions.

Core damage results in the suspension of operations, which means costs for outage and repair. Excessive release of radionuclides to the environment could result. It is therefore important to estimate the probability of core damage and the consequences for the plant and the environment. This is the objective of probabilistic safety analysis. Probabilistic safety analysis had its breakthrough in the mid-1970s through the Reactor Safety Study in the USA. Since then it has been increasingly used for safety assessment as a complement to deterministic safety analysis.

10.1 Scope of Analysis

Probabilistic safety analysis, PSA, known as probabilistic risk analysis (PRA) in the USA, comprises several stages which characterize the level of scope (1001). The first stage, *PRA level 1*, focuses on estimating the core damage frequency, i.e. the probability of core damage per year of reactor operation. This includes the following steps:

- identification of accident sequences leading to core damage;
- analysis of the performance and reliability of the safety systems;
- quantification of accident-sequence probabilities.

The second stage comprises the analysis of the physical processes during core melt accidents:

- study of the core meltdown process and the release of radioactive substances in the reactor vessel;
- analysis of the behaviour of the core melt and the released radionuclides in the reactor containment;
- study of the containment response to severe accident conditions;
- estimation of the radioactive release to the environment.

Risk analysis comprising the first and second stages is called *PRA level 2*.

The dispersion of radioactive substances in the environment and the consequences to life, health and property are studied in the third stage which includes the prediction of:

- the concentration of the radionuclides at different times and distances from the nuclear power plant;
- the resulting radiation doses and effects on the general public;
- the probability distribution of major consequences.

Risk analysis comprising the first, second and third stages is known as *PRA level 3*. A complete risk analysis must consider all kinds of events which can result in core damage, i.e. also the effects of external hazards such as fire, flooding and earthquake.

The first complete risk analysis was the Reactor Safety Study. A similar study was later carried out in the Federal Republic of Germany. In these studies, accident analyses of selected plants and offsite consequence analyses of “average” sites were carried out. It was considered possible to apply the results to the general safety assessment of nuclear power plants with boiling water reactors or pressurized water reactors. This type of study is termed *generic*.

Plant-specific safety studies were later carried out in several countries including Sweden. Although the Swedish studies have so far been limited to PRA levels 1 and 2, separate studies of offsite consequences have also been carried out. These and other plant-specific studies show that the results cannot easily be generalized.

10.2 Reliability Technology

PRA level 1 is based on the *systematic reliability analysis* of systems and components of importance to event sequences which can lead to core damage. The event tree–fault tree methodology is generally used. Special attention is given to the performance and interaction of the safety systems, including operator action. This section describes the main characteristics of the systematic reliability analysis.

10.2.1 Event trees

The basic requirement for avoiding core overheating is that the core remains covered with water and cooled. Under fault conditions, the following *safety functions* are required to ensure adequate core cooling:

- the nuclear chain reaction must be interrupted sufficiently fast;
- water must be supplied to the core in sufficient quantity;
- decay heat must be removed at a sufficient rate.

For identifying potential *core damage sequences*, an initiating event is first specified. It is then investigated, for each possible sequence of events, whether the basic safety functions are satisfied or not. In order to proceed systematically and have a clear picture of the various sequences, *event trees* are used. The trunk of the tree represents the *initiating event* and the branches the success or failure of the basic safety functions. The tip of each branch represents a plant state as a result of the initiating event and a particular combination of subsequent events. The event tree is constructed by induction, i.e. from cause to effect.

Figure 10.1 is an example of a simplified event tree. The initiating event is a pipe break in the reactor coolant system. It is then indicated whether or not reactor isolation, emergency core cooling and residual heat removal are available. At each branching point, the upper branch represents the

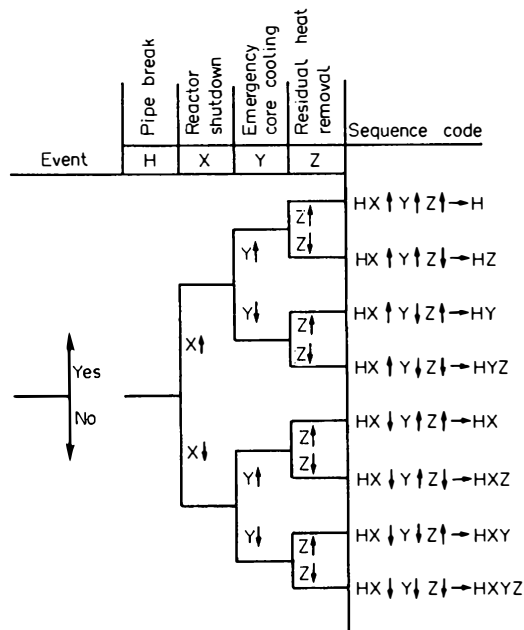


FIG. 10.1. Simplified event tree

success of the particular safety function, and the lower branch represents the failure of the system to fulfil its function. When a safety function is successful, it is indicated in the diagram by a letter and an upward arrow, e.g. $X \uparrow$. Similarly, $X \downarrow$ means that the particular safety function has failed. A sequence of events is represented by the appropriate combination of letters such as $H X \uparrow Y \downarrow Z \uparrow$, where H is the initiating event. An abbreviated system where only the failed safety functions are represented (without the downward arrow) is usually used. Consequently, $H X \uparrow Y \downarrow Z \uparrow$ is equivalent to $H Y$.

If the number of safety functions affecting the accident sequence is taken to be n , the number of branches will be 2^n . In general, many branches can be eliminated as being of no significance to the end result. A *reduced event tree* is then obtained. If H in Fig. 10.1 represents a small or medium pipe break and reactor shutdown (X) fails, it is immaterial if emergency core cooling or residual heat removal is successful or not, since the sequence will still lead to core overheating (Fig. 10.2).

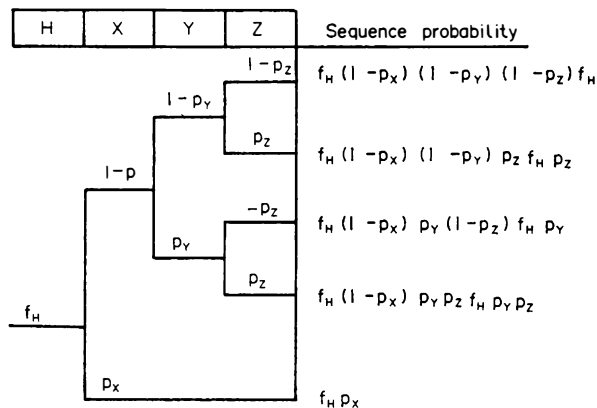


FIG. 10.2 Reduced event tree

Using the reduced event tree, the calculation of the *core damage frequency* can be illustrated. If the frequency of the initiating event is f_H and the failure probabilities of the system functions X, Y, Z are p_X , p_Y , p_Z , the core damage frequency is obtained by multiplication of the failure probabilities (if they are mutually independent) and the frequency of the initiating event. (Note that by definition a probability is a number between 0 and 1, while a frequency, expressed for example as an expected number of events per year, can be greater than 1.) Since the failure probabilities of vital safety functions are low, p_X , p_Y and p_Z represent small numbers. The complementary probabilities, $1 - p_X$ etc., that the particular function will succeed, can then be approximately set equal to 1 in the multiplication.

The simplified event trees in Figs. 10.1 and 10.2 also illustrate a practical, if not a fundamental, complication of the event tree methodology. The description is binary and static. The possibility that system functions are partially or temporarily available is not represented. Intermittent availability is quite possible in situations affected by human action. Obviously, event trees would become very complex if all such possibilities were to be taken into account.

In principle, a very large number of initiating events are conceivable. They can be roughly classified as LOCAs or transients as described in Chapter 9. Within these broad categories, sequences with similar initiating events are grouped together. The groups are characterized by the fact that the same safety function is needed to avoid core overheating. In this way the number of event trees is reduced to a manageable amount.

The criterion for core overheating is usually that the clad temperature exceeds 1200°C (cf 9.2.1). The term *core meltdown* is often used synonymously with core overheating, even if a clad temperature in excess of 1200°C is not necessarily equivalent to a molten core (the melting point of uranium dioxide is 2800°C). An event sequence is assumed to involve either total core meltdown or no core melting. The possibility of limited core damage or partial core meltdown is not explicitly considered. This assumption is conservative and is prompted by the difficulty of predicting the processes occurring in an overheated core.

10.2.2 Function analysis

As described in Chapter 8, a particular safety function can generally be accomplished by several identical systems (redundancy) or by different systems (diversification). In certain cases, interaction between systems is necessary, and may involve action by the reactor operator. Systems which are needed quickly are actuated automatically, while systems required at a later stage can be manually initiated. The aim of function analysis is to determine how and when the required functions can and need to be performed.

The establishment of *system requirements* or “success criteria” i.e. the minimal configuration of (redundant and diversified) systems for the successful performance of a particular safety function, as well as the *interdependence* between systems is of particular concern. In the latter case, a distinction is usually made between front-line systems and support systems (cf 8.3).

The relationship between front-line systems and support systems can be illustrated by a matrix (Fig. 10.3). The diagram shows the interdependence between the emergency core cooling systems and the auxiliary systems in a pressurized water reactor (Ringhals 2). The auxiliary electric systems (AC and DC) are each subdivided into four buses. The high-head injection sys-

		FRONT-LINE SYSTEMS (Components)				
		High-head systems (Pumps) Train			Low-head systems (Pumps) Train	
		1	2	3	1	2
SUPPORT SYSTEMS	Bus					
6.6 kV AC (diesel-backed)	A	x			x	
	B					x
	C		x			
	D			x		
110 V DC (battery-backed)	A	x			x	
	B					x
	C		x			
	D			x		
Component cooling system					x	x
Salt water system		x	x	x		

FIG. 10.3. Interdependence between front-line systems and support systems in Ringhals 2. Adapted from *Ringhals 2 Safety Study*, Swedish State Power Board, 1983

tem consists of three redundant trains and the low-head injection system of two trains. The pumps require 6.6 kV AC power for operation and 110 V DC power for start-up. The component cooling water system and the salt water system are necessary for heat removal from the safety injection systems. The secondary cooling system pumps also depend on electric power for operation.

An example of system requirements for emergency core cooling and residual heat removal in the event of a large LOCA in Ringhals 2 is shown in Fig. 10.4. The table illustrates the high degree of redundancy implemented for these essential safety functions.

10.2.3 Fault trees

The failure of a safety function can be caused by equipment failure, an erroneous manoeuvre or an external event. The purpose of fault tree analysis is to illustrate those combinations of faults which result in functional failure. Fault trees are constructed by deduction (from effect to cause). The undesirable event, or *top event* (the tree is drawn upside down), is the starting-point for the analysis. The top event is successively broken down

EMERGENCY CORE COOLING	RESIDUAL HEAT REMOVAL
<i>either</i>	<i>either</i>
1 (of 3) pump in low-head system	1 (of 3) low-head pump
2 (of 3) effective accumulators	1 (of 3) pump in component cooling system
1 (of 4) pump in containment spray system	1 (of 6) pump in salt water system
	1 (of 3) cooler in low-head system
<i>or</i>	<i>or</i>
2 (of 3) low-head pumps	1 (of 3) low-head pump
1 (of 3) accumulator	1 (of 3) pump in component cooling system
1 (of 4) pump in containment spray system	1 (of 6) pump in salt water system
	2 (of 4) pumps and coolers in containment spray system

FIG. 10.4. Alternative system requirements for a large LOCA in a pressurized water reactor. Adapted from *Ringhals 2 Safety Study*, Swedish State Power Board, 1983

into *basic events* which are interrelated by the branches of the tree in a coherent diagram.

Fault trees are constructed on three levels:

- function fault tree*, where the top event represents the failure of a safety function and the basic events comprise system failures. The function fault tree is the link between fault tree and event tree analysis;
- system fault tree*, where the top event is a failure of a system function and the basic events are failures in components such as pumps, valves, fans, etc.
- component fault tree*, where the top event is a component failure and the basic events represent failures such as mechanical failure, loss of power supply, leakage, inadvertent manoeuvres, etc.

By successive decomposition, safety function failures can be traced back to basic failure events whose probability can be determined by experiment or operating experience. The probabilities are combined through the fault tree logic to obtain the failure probability for the particular safety function.

The principle of a function fault tree is illustrated in Fig. 10.5. Systems A and B are assumed to each fulfil the same function, while systems C and D each fulfil another function. This means that both A and B must fail for the first function to fail and both C and D must fail for the second function to fail. This is illustrated by the use of “and” gates. Moreover, it is assumed that both functions are needed to fulfil the particular safety function. Hence, if either the first or the second (or both) fails, the safety function will fail. This is illustrated by the “or” gate.

If the failure probability of the individual systems is represented by p_A , p_B etc., the failure of the safety function F will be

$$P_F = p_A p_B + p_C p_D$$

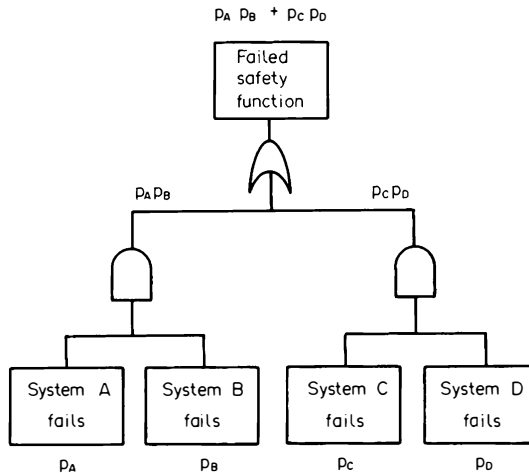


FIG. 10.5. Simplified function fault tree

if the systems are mutually independent. If there are dependences, e.g. a common power supply, the failure probability for the safety function will be larger (see 10.2.5).

The failure probability of a safety function can be reduced by the principle of redundancy. In Fig. 10.5, A and B may represent redundant systems in a “1 of 2” configuration. Important safety functions are often carried out by “2 of 4” systems. This means that the system consists of four subsystems, two of which are sufficient for the required safety function. The fault tree for such a system, broken down into trains, is shown in Fig. 10.6.

If the subsystems are identical and the failure probability of the individual subsystem is p , the failure probability of the safety function will equal the probability that at least three subsystems fail, i.e.

probability that three systems fail and one system succeeds $4p^3(1-p)$	probability that four + systems fail $+ p^4 = 4p^3 - 3p^4$

It is easily seen that the availability of a “2 of 4” system is better than that of a “1 of 2” system if $p < 1/3$.

System fault trees are constructed for each system in the function fault tree, and component fault trees are constructed for each component in the system fault tree. The construction of system fault trees can be simplified by using “standard fault trees” for components, since the same components are included in several systems. Figure 10.7 is an example of a fault tree for a motor-driven pump. In addition to the symbols defined in Fig. 10.5, the circles designate basic events, which do not require further decomposition

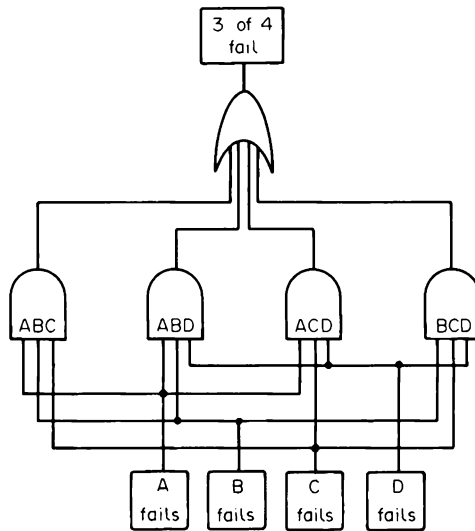


FIG. 10.6. Fault tree for a "2 of 4" system. At least three of the four subsystems must fail for the system function to fail

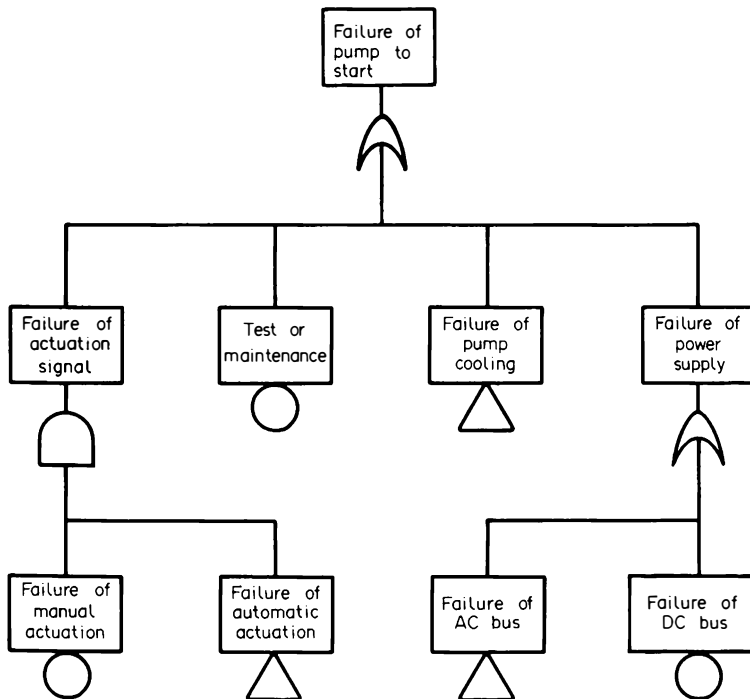


FIG. 10.7. Simplified fault tree for a motor-driven pump

since their failure probabilities can be obtained directly. The triangles indicate transfers from other fault trees common to several fault trees.

When constructing a fault tree of the kind illustrated in Fig. 10.7, several *failure modes* must be represented, such as the failure of a component to start when required or the failure of a component during operation. Failure to start can be caused by spurious malfunction, faulty signals or manoeuvres. A component can also be unavailable due to testing or maintenance.

Because of the large number of components and failure modes, the system fault trees tend to become very complex. There is no generally accepted method of fault tree construction. The failure logic is sometimes ambiguous and completeness cannot be guaranteed. Considerable attention must be paid to dependences and common cause failures.

Each fault tree represents a large number of combinations of basic events leading to the top event. Such a combination is called a *cut set*. There are special computer codes for fault tree analysis which produce the least number of required combinations ("minimal cut sets") and the resulting probabilities. A minimal cut set is such that if a particular basic event is eliminated from the set, the remaining combination of basic events will no longer represent a cut set.

10.2.4 Reliability data

There are two types of failure probabilities in fault tree analysis:

- the probability that a component will fail while in operation;
- the probability that a component on standby is in a failed state at the time of demand.

If the failure occurs randomly, the first probability can be written

$$p(t) = \lambda t$$

if λt is $\ll 1$. The expression gives the failure probability of the component during the time interval 0 to t . λ is called the *failure rate*. If the probability for non-availability on demand is represented by q , the total probability of functional failure will be

$$q + \lambda t$$

The *failure probability per demand*, q , can be obtained experimentally from the observed number of start-up failures in a (large) number of trials.

Faults in components on standby are mainly discovered during routine testing. The probability of faults during the period between two tests is on average $\lambda T/2$, where T is the time between tests. The contribution to unavailability due to repair of a redundant component can be set equal to λt_R where t_R is the average repair time.

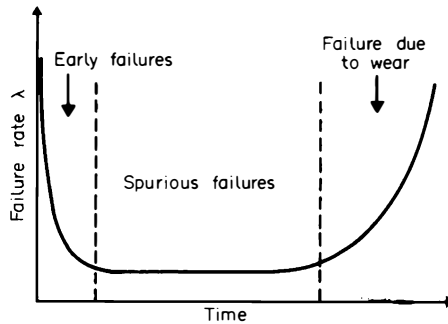


FIG. 10.8. Typical failure rate curve for technical components ("the bathtub curve")

In typical cases, the failure rate varies with time as shown in Fig. 10.8. Most components are designed, tested and used so that they are at stage 2, i.e. with a constant (low) failure rate. This is achieved through careful quality control and testing which eliminates components with high initial failure rate. At the other end of the scale, the failure rate increases due to wear and ageing. The components are therefore replaced before this stage is reached.

Failure statistics from Swedish nuclear power plants are centrally stored. A common data base of failure rates has been compiled by processing and supplementing the raw data (1002). Generic failure rates, such as those in Table 10.1, can be updated for plant-specific analyses by incorporating operating experience from the plant itself. In this way the data uncertainties are reduced.

10.2.5 Dependent failures

A distinction is made between *independent* failures which occur at random and *dependent* failures, which are correlated. Fault tree analysis that only considers independent failures would give misleadingly low failure probabilities. There are several types of dependences. Dependence may imply that the failure of a support system results in the unavailability of several other systems, or that identical components fail due to a common cause. It is practical to consider two groups of dependent failures:

- failure due to functional dependence,
- common cause failure (CCF).

Examples of systems and functions which can cause the first type of failure are: auxiliary power systems, component cooling systems, salt water systems, ventilation systems, control signals and human error. The depen-

TABLE 10.1 *Typical failure data for components in Swedish boiling water reactors*

Component	Failure	Failure probability per 10 ³ demand	Failure rate per 10 ⁶ hours
Centrifugal pump	Inadvertent trip		30
Piston pump, on standby	Failure to start	4	
Isolation valve, motor-operated	Failure to change position	7	
	Failed/erroneous indication	0.9	
	Inadvertent/erroneous indication	0.9	
Check valve	Failure to close	3	
	Failed/erroneous indication	33	
	Inadvertent/erroneous indication	23	
Safety valve	Inadvertent opening		1.3
	Failure of main valve to open	0.78	
	Failure of pilot valve to open	8.3	
	Failure of main valve to re-close	2.4	
	Failure of pilot valve to re-close	1.2	
Control rods	Failure of hydraulic scram	0.028	
	Failure of fine-motion control rod insertion	0.66	
Diesel generator	Failure to start	7.7	
	Inadvertent trip		5500
Battery	Failure of power supply on demand	13	

Source: *The T-book. Reliability Data for Components in Swedish Power Reactors*, Report KS 85-05, Nuclear Safety Board of the Swedish Utilities, 1985

dences are explicitly considered in the function analysis and represented in the function fault trees.

The second type of failure concerns components and systems without direct functional dependence, for example:

- failure due to external events, such as fire, earthquake, onsite or offsite flooding etc.;
- failure caused by propagation, when a primary failure causes a secondary failure. An example: jet impingement as a result of a large pipe break in the reactor coolant system can damage equipment in the reactor containment;

—failure in identical components through manufacturing faults, environmental effects (e.g. corrosion), normal wear, erroneous calibration, etc.

External events are usually not explicitly treated at PRA level 1 but are only dealt with through the effect they may have due to the location of certain safety-related equipment in common rooms. Failure modes due to propagation can be identified and quantified in the system fault trees.

Failures in identical components can have a number of causes which are difficult to represent in a fault tree. They are therefore modelled using special methods. In the *beta-factor method*, the minimal cut set probabilities are modified with regard to dependent failures in the identical components. In the simplest case of two redundant components the resulting failure probability takes the form:

$$p^2 + \beta p$$

where p is the individual failure probability and β is a measure of the dependency. Similar expressions are obtained for three or more identical components.

The beta-factor can be estimated from operating statistics by the identification of failures occurring simultaneously in several identical components and which have not been modelled in the fault tree. A beta-factor estimate is then obtained from the ratio of the number of simultaneous failures and the total number of failures for the particular component. The beta factor is usually in the interval 0.01 to 0.1. This means that the contribution from dependent failures will dominate the total failure probability for low values of the independent failure probabilities ($p < 0.01$).

10.2.6 Human reliability

Human error can affect an accident sequence in two ways:

- erroneous action during routine conditions, e.g. in testing or maintenance;
- erroneous or omitted action during the course of an abnormal event.

The first type of error is characterized by manual action contrary to established rules and procedures and is therefore often called *procedural error*. Examples include systematic miscalibration of instruments and erroneous base-setting of components. Such errors are generally included in the failure statistics reported from the plants. They can therefore be directly quantified in the component fault tree.

The second type, *errors of commission or omission*, can be modelled in event trees or system fault trees although there is no direct basis of experience for the quantification. Special analyses, such as the construction of

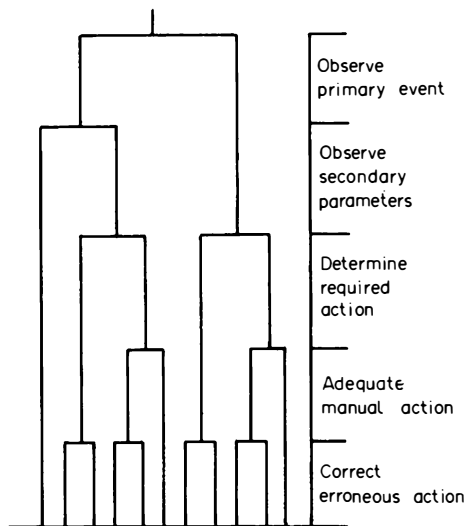


FIG. 10.9. Operator-action tree. From *Ringhals 1 Safety Study*, Swedish State Power Board, 1984

operator-action trees (Fig. 10.9), are therefore required. This method involves three main steps: observing the abnormal event, diagnosing the problem, and taking corrective action. The probability of a step omitted can be estimated by analysing the human ability to carry out a sequence of tasks according to given instructions. The time available for the operator to carry out the tasks and the stress he may experience are taken into account.

10.2.7 Quantification

As indicated in the event tree (Fig. 10.2), it is formally simple to calculate the frequency of an accident sequence if the frequency of the initiating event and the failure probabilities of the safety functions are known. The latter are obtained by fault tree analyses which are successively broken down into a number of basic events for which the probabilities can be directly assigned or estimated from operating data.

In general, fault trees become very complex even for relatively simple systems. In order to calculate the probability for the top event, computer codes are used which model the logical structure of the tree. The input data are point estimates of the basic event probabilities. Dependences are introduced at the function level, and human action at the event sequence level if the action is unique for the particular sequence. If not, human action is introduced at lower levels in the fault tree hierarchy.

The process for quantifying core damage sequences is illustrated in Fig.

10.10. The dependences between front-line systems and common support systems must be considered at the sequence level so as not to underestimate the sequence probability. Since event trees generally contain both available and unavailable functions, cut sets which are mutually exclusive must be eliminated so as not to *overestimate* the sequence probability. For example, in Fig. 10.10 function X may presuppose that auxiliary power is available, while a cut set for function Y assumes that auxiliary power is unavailable, which is not possible at the same time.

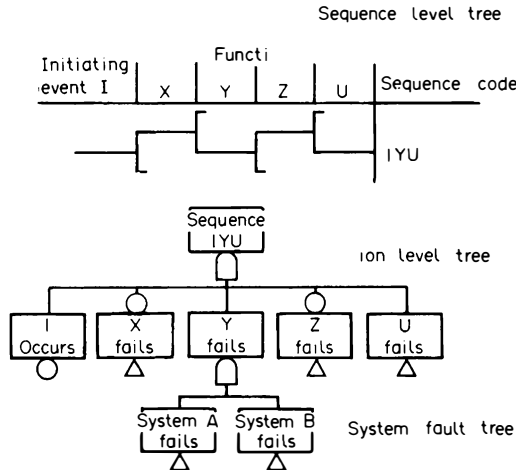


FIG. 10.10. Logic for the quantification of core damage frequencies. From *Ringhals 1 Safety Study*, Swedish State Power Board, 1984

The quantification provides numerical values of the frequencies for the various sequences. The total core damage frequency is then obtained by summing the frequencies for the individual sequences. Alternatively, all sequence level trees for a given initiating event may be totalled *before* quantification (1010). Cut sets which exclude each other are then automatically eliminated, which simplifies the quantification. However, information on the individual sequence probabilities is then lost.

In general, it is found that the contribution from a few sequences will dominate the total core damage frequency. For these sequences it is of interest to determine the contributions from various basic events. This is achieved through *sensitivity analysis*. The input data are then varied and the effects on the end result are examined. The effects of uncertain data, such as human error frequencies and common cause failure probabilities, are often studied in this way.

It is useful to estimate the relative importance of a component to the unavailability of a particular system or to a (dominant) core damage

sequence. The importance of a particular component can be determined as the ratio of all cut sets to which the component contributes and the total amount of cut sets in the particular system (sequence). The ratio is a measure of the sensitivity of the system (sequence) to the particular component.

10.2.8 Uncertainties

The probability of a basic event, such as a component failure, is characterized by a *distribution function*, which can, in principle, be determined by experiment. The distribution function can be expressed as a mean value and a standard deviation, or as a median value with upper and lower confidence bounds. Mean values of basic event probabilities are usually used as input for fault tree quantification. The sequence probabilities then also become mean values.

The uncertainty of the input data propagates through the fault trees and event trees into a resulting uncertainty for the sequence probability. The total uncertainty can be estimated using statistical methods and the algebraic expressions for the sequence probabilities. There are special computer codes for such calculations.

Another type of uncertainty which is more difficult to quantify is due to fault tree modelling, e.g. of common cause failures and human error. *Incompleteness*, i.e. the omission of relevant failure modes, also belongs to this type of uncertainty. The change of material properties with time ("ageing") is an example of phenomena which are difficult to represent. Reliability studies must therefore be continually updated.

10.3 Plant Analyses

The first systematic safety study using probabilistic methods was carried out for the U.S. Atomic Energy Commission by a group under the direction of Norman C Rasmussen and published in 1975 (1004). This study, known as the Reactor Safety Study, served as the reference for a series of subsequent studies. This section provides a brief summary of the first part of the Reactor Safety Study, namely the estimation of core damage frequencies. Some of the results from a similar study conducted in West Germany and from some plant-specific Swedish studies are then presented. Finally, a comparison is made between results for boiling water reactors and pressurized water reactors. The effects of external events are discussed in section 10.5.

10.3.1 The Reactor Safety Study

The Reactor Safety study was made for a pressurized water reactor, *Surry-1*, with 788 MW electric output, supplied by Westinghouse in 1972, and a 1065 MWel boiling water reactor, *Peach Bottom-2*, designed by General

Electric and commissioned in 1974. These two reactors were typical of the state-of-the-art of reactor technology at the end of the 1960s.

More than a thousand event sequences were studied using the event tree – fault tree methodology. Dominant sequences were subjected to detailed quantitative analysis. The total core damage frequency was estimated at 6×10^{-5} per reactor year for the pressurized water reactor and 3×10^{-5} for the boiling water reactor. These values fall within each other's confidence bounds. The Reactor Safety Study therefore gives a common frequency of 5×10^{-5} per reactor year for both types of reactors. The upper confidence bound is estimated at 3×10^{-4} per reactor year which means that the core damage frequency is lower than this value with a probability of 95%.

The dominant accident sequences for *pressurized water reactors* are summarized in Table 10.2. The table indicates that small LOCA, i.e. sequences which are initiated by small pipe breaks or primary system leakage make the largest contribution, 17×10^{-6} per reactor year or 17 per million years (PMY), to the total core damage frequency. This results from the fact that the initiating event frequency is substantially greater and the safety function failure probability not essentially lower than those for large breaks. The failure probability is dominated by human error, particularly in the switch-over from the safety injection mode to the recirculation mode of emergency core cooling operation (cf 8.2.5).

A failure mode of relatively high frequency which was “discovered” in the Reactor Safety Study was the *interfacing systems LOCA* (“V-LOCA”), estimated to have a frequency of 4 PMY. The V-LOCA is caused by failure

TABLE 10.2. Dominant core damage sequences for a U.S. pressurized water reactor (Surry 1) according to the Reactor Safety Study (1004). The frequencies and probabilities are median values

Event	Frequency (per year)	Failed safety function	Safety function failure probability	Core damage frequency (PMY) ^a
Large LOCA	1×10^{-4}	Safety injection	1×10^{-2}	1
		Recirculation	2×10^{-2}	2
Medium LOCA	3×10^{-4}	Safety injection	1×10^{-2}	3
		Recirculation	1×10^{-7}	3
Small LOCA		Coolant make-up	6×10^{-3}	6
		Recirculation	9×10^{-3}	9
		Containment spray	2×10^{-3}	2
Interfacing systems				
LOCA	4×10^{-6}	Low-head safety injection	1	4
Loss of auxiliary power	2×10^{-1}	Decay heat removal	3×10^{-1}	6
Unsuccessful reactor scram	4×10^{-4}	Closure of isolation valve	2.5×10^{-3}	1

^a 1 PMY (per million years) = 10^{-6} per year of reactor operation.

of the check valves which isolate the low-head injection core cooling system from the reactor's main coolant system. For this event to occur, two check valves connected in series must fail. The low-head injection system will then be subjected to a pressure for which it has not been designed, which will almost certainly lead to failure of the system. This results in a medium LOCA without an operable low-head injection system for emergency core cooling. It has been possible to considerably reduce the probability of this failure mode by simple measures such as more frequent inspection of the check valves.

Transients initiated by loss of offsite power make a significant contribution to the core damage frequency. Loss of power results in feedwater pump trip, and if the auxiliary feedwater system also fails, the steam generators will boil dry within about an hour. Blowing steam through the safety valves on the pressurizer then leads to loss of coolant and the uncover and meltdown of the core within 2–3 hours. If emergency power is available from the diesel generators, the containment spray system will ensure that containment integrity is not threatened, until offsite power is recovered.

The core damage frequency for loss of power transients is calculated as follows. U.S. experience indicates that loss of offsite power occurs about 0.2 times per year. The probability that the auxiliary feedwater system is not available is estimated at 1.5×10^{-4} per demand. If offsite power is recovered within 1 hour, the main feedwater system can be used for decay heat removal. The probability of offsite power not being recovered within that time is estimated at 2×10^{-1} . The resulting frequency becomes $0.2 \times 1.5 \times 10^{-4} \times 0.2 = 6 \times 10^{-6} = 6 \text{ PMY}$. If electric power cannot be recovered within about 3 hours the containment will fail due to overpressure, releasing a large amount of radioactive substances.

The dominant core damage sequences for *boiling water reactors* are presented in Table 10.3. The availability of the boiling water reactor emer-

TABLE 10.3. *Dominant core damage sequences in a U.S. boiling water reactor (Peach Bottom-2) according to the Reactor Safety Study (1004). The frequencies and probabilities are median values*

Event	Frequency (per year)	Failed safety function	Safety function failure probability	Core damage frequency (PMY)
Large LOCA	1×10^{-4}	Emergency core cooling	1×10^{-3}	0.1
Medium LOCA	3×10^{-4}	Emergency core cooling	7×10^{-3}	2
Small LOCA	1×10^{-3}	Coolant make-up	2×10^{-4}	0.2
		Decay heat removal	1×10^{-4}	0.1
Arbitrary transient	10	Decay heat removal	1.6×10^{-6}	16
Anticipated transient without scram	1.3×10^{-4}	Reactor shutdown	1×10^{-1}	13
Loss of main feedwater system	3	Coolant make-up	1.3×10^{-7}	0.4

gency core cooling systems is judged to be better than that of the pressurized water reactor. Hence, LOCA is found to contribute less to the total core damage frequency than in PWRs. Instead, the total core damage frequency is dominated by transients with *inadequate residual heat removal*. Residual heat removal is necessary at reactor scram regardless of the cause for scram. Since it is assumed that scram occurs ten times per reactor year, and the unavailability of the residual heat removal system is estimated at 1.6×10^{-6} per demand, the expected core damage frequency is 16 PMY

Anticipated transients without scram (ATWS) are estimated to have a relatively high frequency for the reference BWR. Reactor shutdown can be achieved in two ways, by reactor scram or by a combination of recirculation pump runback and operator action, either actuation of the boron injection system or manual insertion of the control rods. The median value for scram failure is estimated at 1.3×10^{-4} per demand with an uncertainty factor of 3. The probability of failure to shut down the reactor by alternative means is estimated at 0.1. The core damage frequency is therefore 13 PMY with an uncertainty factor of 4.

The Reactor Safety Study was a pioneer effort in the application of systematic reliability analysis to nuclear power plants, which lent new possibilities to quantitative safety analysis. When assessing the results, it must be borne in mind that they refer to two specific reactors built around 1970. The study is therefore of limited relevance to other and newer reactors where improvements in safety design have been implemented, partly as a consequence of probabilistic safety studies.

10.3.2 The German Safety Study

In 1979 a safety study conducted under the direction of A Birkhofer was published in the Federal Republic of Germany (1005). The study used event tree–fault tree methodology for the analysis of a German pressurized water reactor, Biblis B with 1300 MW electric output, commissioned in 1976. There are several design differences between this reactor and the reference PWR in the Reactor Safety Study, Surry-1, but the results and conclusions in the first part of the study, the estimation of core damage frequencies, are largely the same. The dominant core damage sequences are summarized in Table 10.4. It can be seen that small LOCA makes the largest contribution, followed by loss of offsite power. The mean value of the total core damage frequency is estimated at 90 PMY. The corresponding median value is 40 PMY, which can be compared with the 60 PMY reported by the Reactor Safety Study. The total core damage frequency was estimated to lie in the region of 10–300 PMY with 90% confidence.

The contributions to the dominant sequences from various failure sources are shown in Table 10.5. The largest contribution, about two-thirds, originates from human error, mainly in connection with the manual realignment

TABLE 10.4. Dominant core damage sequences for Biblis B according to the German Safety Study (1005). The frequencies and probabilities are mean values

Event	Frequency (per year)	Safety function failure probability	Core damage frequency (PMY)
Large LOCA	2.7×10^{-4}	1.7×10^{-3}	0.5
Medium LOCA	8×10^{-4}	2.3×10^{-3}	2
Small LOCA	2.7×10^{-3}	2.1×10^{-2}	57
Loss of offsite power	1×10^{-1}	1.3×10^{-4}	13
Loss of main feedwater system	8×10^{-1}	4×10^{-6}	3
Loss of auxiliary power with failure of pressure relief valve to reclose	2.7×10^{-4}	2.6×10^{-2}	7
Failure of pressure relief valve to reclose	1×10^{-3}	2×10^{-3}	2
Anticipated transient without scram	3×10^{-5}	3×10^{-2}	1

TABLE 10.5. Contributions to the core damage frequency of Biblis B from various failure sources according to the German Safety Study (1005)

Event	Core damage frequency (PMY)	IF ^a %	CCF ^a %	HE ^a %	IF+ CCF %	IF+ HE %	CCF+ HE %	IF+ CCF+ +HE %
Large LOCA	0.5	73	15	12				
Medium LOCA	2	62	11	27				
Small LOCA	57	13	1	85				
Loss of auxiliary power	13	26			29	27	18	
Failure of pressure relief valve to re- close during loss of auxiliary power	7	33			26	4		37
Total	80	18%	1%	63%	7%	5%	3%	3%

^aIF = Independent failure of technical equipment. CCF = Common cause failure of technical equipment. HE = Human error.

during change-over to the recirculation emergency core cooling mode, particularly during small LOCA. During large LOCA, the largest failure source is unsuccessful safety injection from the accumulators, whereas functional failures in the high-head injection core cooling system make a dominant contribution during medium LOCA.

During a loss of offsite power transient, human error is of no significance since all countermeasures are initiated automatically. If feedwater is unavailable due to failure of the auxiliary feedwater system or common cause failure in the diesel generators, auxiliary feedwater can be drawn from the sister unit, Biblis A, by manual realignment. Therefore, common cause

failure alone makes no contribution in this case and only does so in combination with other failure sources. Sequences to which common cause failures contribute, represent about 15% of the total core damage frequency.

10.3.3 Forsmark 3

In 1977 Asea-Atom carried out a safety study of Forsmark 3, which was then under construction. The study was based on event tree–fault tree methodology and was set up as a comparison between Forsmark 3 and the reference BWR plant in the Reactor Safety Study, Peach Bottom-2. The results of the study are summarized in Fig. 10.11. The diagram indicates that the total core damage frequency for Forsmark 3 was estimated at about one-eighth of that of Peach Bottom-2. Several factors were considered to contribute to this result:

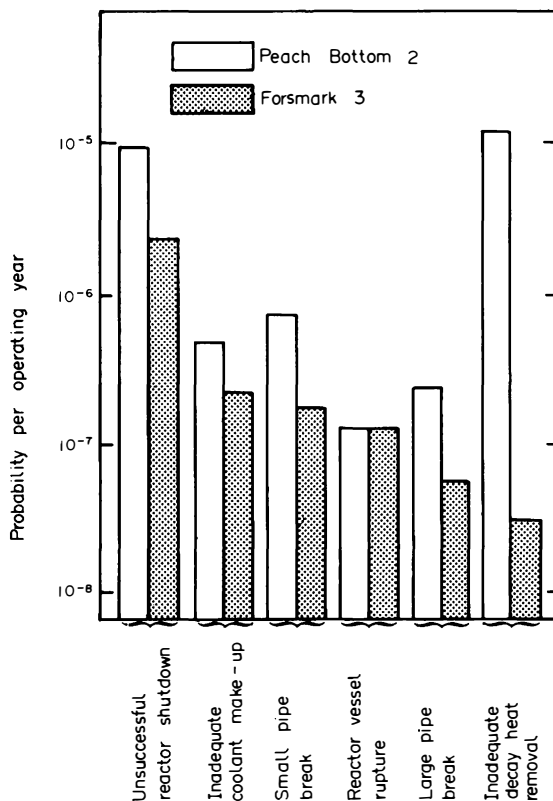


FIG. 10.11. Comparison of core damage frequencies in Forsmark 3 and Peach Bottom-2 according to the 1977 study (1006)

- Improved redundancy and consistent segregation of subsystems in Forsmark 3.
- Control rod insertion can be effected hydraulically (scram) or electro-mechanically (screw). The latter possibility is not available in the U.S. plant.
- The various reactor units at Forsmark have no safety-related common functions or shared areas, in contrast to the situation at Peach Bottom.
- The external grid of Forsmark 3 is considered “stronger” than that of Peach Bottom-2 because the start-up grid at Forsmark, acting as a back-up for the main grid, is connected to gas turbine-driven generators (cf 4.6.1).
- The Swedish 30-minute rule implies that no action is required by the operator within the first half-hour after a large pipe break. This rule also reduces the need for operator action in other cases.

An updated safety study of Forsmark 3 was reported in 1985. The total core damage frequency is estimated at 7 PMY, i.e. about the same value as in the earlier study. However, the distribution of dominant sequences is different (Table 10.6), as are the dominant contributors to the core damage sequences. Transients with inadequate reactor coolant make-up represent more than 80% of the sequences, while LOCA events only represent 0.5% of the total core damage frequency.

Insufficient coolant make-up involves loss of the feedwater system, failure of the auxiliary feedwater system and the failure to connect the low-head injection system, due to failure of depressurizing the main coolant system or failure of the low-head injection system itself. The most probable sequence in Table 10.6 is dominated by common cause failure in the auxiliary feedwater system in combination with failure of the manually initiated depressurization.

TABLE 10.6. *Dominant core damage sequences in Forsmark 3 according to the 1985 study (1007). The frequencies and probabilities are mean values*

Event	Frequency (per year)	Failed safety function	Core damage frequency (PMY)
Loss of feedwater	0.25	Coolant make-up	4.3
Loss of feedwater after another primary event	3.3	Decay heat removal	0.62
Loss of auxiliary power	0.13	Coolant make-up	0.60
Reactor vessel failure	2.7×10^{-7}		0.27
Manual or automatic scram	3.5	Coolant make-up	0.06
Loss of main heat sink	1.5	Reactor shutdown	0.06
Medium LOCA	3.8×10^{-4}	Reactor shutdown	0.014
Small LOCA	5.6×10^{-2}	Decay heat removal	0.010
Large LOCA	1.0×10^{-4}	Reactor shutdown	0.007

10.3.4 Oskarshamn I

Oskarshamn I is the oldest Swedish unit. It has an Asea-Atom boiling water reactor designed according to the safety philosophy of the mid-1960s. During construction, certain safety-related problems for the reactor's auxiliary power supply system became apparent. Extensive modification of the electric and control equipment was carried out in order to improve the segregation of the electric systems. The experience from this work was then used in the design of subsequent plants in Ringhals and Barsebäck.

The safety design of Oskarshamn I remains valid, even in the light of newer, more stringent requirements. The auxiliary power supply system has shown a high reliability. Nevertheless, reliability analyses conducted in the mid-1970s revealed certain weaknesses in the power supply system. They related to the fact that there was shared equipment for the redundant subsystems, which could cause loss of power as a result of fire or explosion.

The complete physical segregation of the subsystems could not be achieved without thorough plant modification. This was carried out during 1978–80 and involved the installation of a new power supply system, completely separated from the old one. The new system supplies power to all components and systems required for the safe shutdown of the reactor, i.e.:

- the pressure relief valves, so that the reactor pressure can be regulated;
- the reactor coolant make-up system, so that the core can be kept covered and cooled;
- the containment spray system, so that the containment can be cooled and the decay heat removed.

A new separate building was installed, which houses a reserve control room from which all essential safety functions can be operated and monitored. The power supply in the new building is subdivided into two complete trains located in separate fire cells. The new system can fulfil its function even if the entire old power supply and control building becomes inoperable as a result of fire or explosion.

A probabilistic analysis was conducted in order to estimate the probability of fire or other events in the central or reserve control room, leading to failure of core and containment cooling, and to identify the components and systems which contribute to this probability. The study included an assessment of the initiating event frequencies and a fault tree analysis of all systems for pressure regulation, reactor coolant make-up and decay heat removal.

The results are summarized in Table 10.7. The core damage frequency in the event of fire in the central control room is estimated at 4 PMY, to which inadequate containment cooling contributes 75% and inadequate reactor

TABLE 10.7. Core damage frequencies for fire in the power supply section of Oskarshamn 1 (1008)

Initiating event	Frequency (per year)	Safety function failure probability	Core damage frequency (PMY)
Fire in the central power supply section	1×10^{-3}	4×10^{-3}	4
Fire in RKB ^a (loss of one sub, offsite power available)	1×10^{-3}	1×10^{-4}	0.1
Fire in RKB (loss of both subs, offsite power available)	1×10^{-4}	5×10^{-3}	0.2
Fire in RKB (loss of both subs, loss of offsite power)	1×10^{-6}	4×10^{-2}	0.04

^a RKB = Reserve control building.

coolant make-up 25%. However, since failure of the containment cooling does not lead to high pressure in the containment until after 10–15 hours, there are good possibilities for mitigative measures to avoid containment failure.

The dominant sequence for fire in the reserve control building is initiated by the failure of both onsite power supply buses, but with offsite power still available. Inadequate coolant make-up then contributes to the core damage frequency with about 50% and failure to maintain the reactor pressure with about 25%.

The conclusion of the reliability analysis is that the modification of the electrical section reduced the core damage frequency due to fire or similar events by at least a factor of 100.

The possibility of core damage from pipe breaks in the primary system has also been studied (1008). For top breaks (cf 9.4.3), the core can always be refilled to ensure cooling. For large bottom breaks, the core cannot be refilled and must be cooled by spray water from the low-head injection system. For medium breaks, automatic depressurization must be initiated to enable the low-head injection system to operate. For a break flow rate of less than 100 kg/s the feedwater system is adequate and for break flows less than 30 kg/s the auxiliary feedwater system is sufficient to keep the core covered. In the event of a pipe break, reactor scram and reactor isolation are, of course, initiated.

The results are summarized in Table 10.8. The dominant sequence is a small LOCA, followed by medium LOCA, while large and very small LOCA result in lower core damage frequencies. For small breaks, the feedwater system maintains the water level in the reactor. The feedwater system draws water from the turbine condenser. The condenser inventory lasts for at least 30 minutes. Within this time, manual realignment of a make-up system to the condenser is required to maintain the feedwater capacity at 100 kg/s. Unsuccessful realignment is the dominant failure source. For

TABLE 10.8. *Core damage frequencies during LOCA in Oskarshamn I according to the 1982 safety study (1008)*

Initiating event	Break flow rate (kg/s)	Frequency (per year)	Dominant failed safety function	Safety function failure probability	Core damage frequency (PMY)
Large break	2000–16,000	5×10^{-5}	Emergency core cooling	2.6×10^{-3}	0.1
Medium break	100–2000	1×10^{-4}	Automatic depressurization	1.3×10^{-2}	1.3
Small break	30–100	5×10^{-4}	Coolant make-up	7×10^{-3}	3.5
Very small break	5–30	1×10^{-3}	Coolant make-up	1.3×10^{-7}	0.1

medium breaks, failure of automatic depressurization, rendering the low-pressure spray inoperable, makes the largest contributions to the core damage frequency.

10.3.5 Ringhals 1

Ringhals 1 (750 MWe, commissioned 1975) is the second in the series of Swedish boiling water reactors. The design of Ringhals 1 differs from that of Oskarshamn 1 in certain respects. The turbine plant has two turbo-generators, each with its condenser and feedwater system. This makes it possible to have one turbine shut down for maintenance while the other remains in operation. It also results in a reduction of the number of potential core damage transients due to malfunction of the turbine and feedwater systems.

The auxiliary feedwater system has a steam-driven pump which is independent of the power supply. The emergency core cooling system consists of two redundant, completely segregated loops, each with a steam-driven high-head pump and an electrically driven low-head pump in series. Core spray is therefore available at full reactor pressure. The pressure relief system has twenty safety valves discharging directly into the drywell, ten blow-down valves discharging into the condensation pool, and two pressure regulation valves. The system has a capacity corresponding to 140% of full nominal steam flow.

A reliability study was conducted from 1980 to 1983 using event tree–fault tree methodology (1003). Potential core damage sequences were grouped according to the type of initiating event. The definition of LOCA was based on the expected break flow as follows:

- A Large LOCA, break flow > 1200 kg/s.
- S₁ Medium LOCA, break flow 35–1200 kg/s.
- S₂ Small LOCA, break flow < 35 kg/s.

Transients were grouped into the following categories:

- T_M Reactor shutdown with all essential normal operating systems initially available. This includes inadvertent reactor scram and scheduled outages.
- T_T Loss of the main heat sink, the turbine condenser.
- T_F Loss of the main feedwater system, with the special case, T_F^1 , partial loss of feedwater.
- T_E Loss of main offsite power (400 kV), leading to the failure of both the main heat sink and the feedwater system.

Anticipated transients without scram were considered in the event tree analysis but not as a separate group of initiating events. Loss of feedwater (T_F) was treated as a subset of T_E and inadvertent reactor isolation as a subset of T_M .

Event trees were constructed for all groups of LOCA and transients. The event tree for the shutdown transient T_M is shown in Fig. 10.12. It also

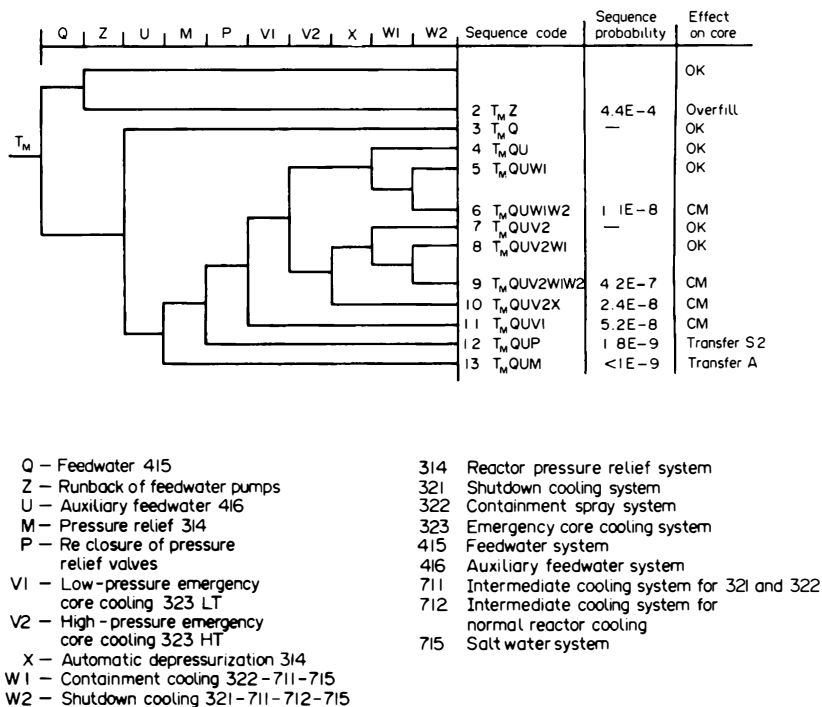


FIG. 10.12. Event tree for a shutdown transient with at least one turbine condenser available. From *Ringhals 1 Safety Study*, Swedish State Power Board, 1984

serves to define the essential system functions along with the established event codes and system numbers.

Normal reactor shutdown requires that at least one feedwater system for coolant make-up and at least one turbine condenser for decay heat removal should be available. The feedwater supply is adjusted to the reduced steam production in the shutdown reactor to avoid reactor vessel overfill.

The lower part of the tree comprises sequences where the main feedwater system is not operable. The auxiliary feedwater system then assumes the coolant make-up function. If this system is not available, the relief valves must first open to avoid overpressure in the reactor vessel and pipelines, and then close to prevent too large a pressure drop and loss of coolant. If pressure is maintained as intended, the high-head injection system will supply coolant make-up. Automatic depressurization may be necessary for introducing the low-head injection system. The containment spray system or the shutdown cooling water system is then used for decay heat removal.

The event trees are successively broken down into function fault trees, system fault trees and component fault trees as described in section 10.2. The initiating event frequencies for LOCA were adopted from the Reactor Safety Study. For transients, empirical scram data from Ringhals 1 were used. The failure rates for basic events were taken from operating statistics as far as possible.

The results are summarized in Table 10.9, showing the dominant core damage sequences in order of importance. Frequencies and probabilities are point-estimated mean values.

The mean value of the total core damage frequency is estimated at 2.5 PMY. The largest contribution comes from a medium LOCA with failure of condensation pool cooling, i.e. a functional failure in one of the systems in the cooling train 322–711–715 (see Fig. 10.12). The temperature of the condensation pool then reaches 95°C after 4 hours, resulting in pump cavitation. Medium LOCA with failure of the low-head injection system (S_2V_1) or faulty back-flushing of the strainers (see Fig. 4.6) of the emergency core cooling system (323) and the condensation pool cooling system (322) also make relatively large contributions to the core damage frequency.

The next important sequence is a large LOCA with the same functional failures as in the previous case, i.e. sequences *AW*, *AV* and *AY*. For transients, loss of offsite power makes the largest contribution, 0.21 PMY in total. It should be noted that the contribution from transients is significantly lower than that from LOCA, while the reverse was the case for the boiling water reactor analysed in the Reactor Safety Study. This is partly due to the fact that Ringhals 1 has two turbines and two feedwater systems which reduces the number of transients, particularly those caused by loss of feedwater. Another reason is that transients with loss of condensation pool cooling are predicted to make a relatively small contribution in the Swedish study.

A newly installed reactor coolant make-up system which serves as a

TABLE 10.9. Dominant core damage frequencies for Ringhals 1 according to the 1984 safety study (1003). Frequencies and probabilities are point-estimated mean values

Event	Frequency (per year)	Failed safety function ^a	Safety function failure probability	Core damage frequency (PMY)
Medium loss of coolant	12.5E-4	W	5.2E-4	0.65
		V ₁	3.4E-4	0.43
		Y	1.4E-4	0.18
Large internal pipe break	3E-4	W	6.9E-4	0.21
		V ₁	3.4E-4	0.10
		Y	3.0E-4	0.09
Reactor vessel rupture	2.7E-7			0.27
External pipe break	9E-4	Reactor isolation	2.0E-4	0.19
Loss of auxiliary power	0.9	UV ₁ Q'	2.9E-8	0.071
		UV ₂ WQ"	5.6E-8	0.050
		UVQ"	1.3E-8	0.012
		CM	3.1E-8	0.028
		CH	3.0E-8	0.027
		CK	2.1E-8	0.019
Partial loss of feedwater	1.0	CH	6.0E-8	0.060
		CK	2.1E-8	0.021
		CL	1.0E-8	0.010

U = Auxiliary feedwater

V₁ = Low pressure safety injectionV₂ = High pressure safety injection

W = Decay heat removal

Y = Backflushing of strainers

C = Automatic scram

Q' = Restart of feedwater system within 30 minutes

Q" = Restart of feedwater system within 4 hours

M = Pressure relief

H = Operator action

K = Fine-motion control rod insertion

L = Control rods

backup to the auxiliary feedwater system effectively reduced the estimated core damage frequency. Coolant make-up can then take place at all reactor pressures with a flow of up to 20 kg/s. If offsite power is lost, the pump of the make-up system is powered by a dedicated diesel generator.

A closer analysis shows that loss of all make-up systems, i.e. sequences containing UVQ or UXQ, are included in sequences representing 28% of the total core damage frequency. The event Q is dominated by failure to manually restart the main feedwater system after loss of offsite power. The dominant contribution to U is a functional failure of the steam-driven auxiliary feedwater pumps. Event V is caused by several kinds of failures of the emergency core cooling system which are both independent and common cause. Event X is failure of automatic depressurization.

Inadequate decay heat removal is a factor in sequences representing 39% of the total core damage frequency. The dominant failure source was identified as a valve failure in the intermediate cooling system which is a part of the main cooling chain to the sea, serving the containment spray system and the main shutdown cooling water system (cf 8.3.3).

Unsuccessful or incorrectly performed back-flushing of the suction strainers in the condensation pool enters into sequences representing 12% of the total core damage frequency. The dominant failure source is operator error.

Anticipated transients without scram (ATWS) represent about 11% of the total core frequency. The relatively low ATWS contribution is mainly due to the large pressure relief capacity and the several alternatives of reactor shutdown. Inadvertently closed valves in the reactor vessel level measuring system are a dominant failure source. In the event of partial loss of feedwater, no scram signal is then obtained for low water level in the vessel.

The reason that a small LOCA does not contribute significantly to the core damage frequency is partly a matter of definition. At break flows < 35 kg/s any of the available make-up systems is sufficient to keep the core covered. A small LOCA is therefore analogous to a transient with a low frequency of occurrence, resulting in a negligible contribution to the core damage frequency.

10.3.6 Ringhals 2

Ringhals 2 (800 MWe, commissioned 1975) was the first pressurized water reactor plant in Sweden. Ringhals 2 has three reactor coolant loops and, similar to Ringhals 1, two turbines, each with its own condenser and feedwater system. The most important operating and safety systems are described in Chapters 5 and 8.

A probabilistic safety study was reported in 1983. The assumed initiating events are limited to those caused by “internal” failures in plant equipment and by human error, as well as by loss of offsite power. The usual classification into LOCA and transients is used. The transients are broadly grouped into events that have occurred frequently in the history of PWR operation, called *anticipated* events, and events that have occurred infrequently or not at all, called *postulated* events (cf 7.4). The frequencies of the anticipated events are obtained from operating experience, while frequencies for postulated events are based on assessment. Event trees are constructed for the following categories of initiating events:

- Large LOCA, break area > 175 cm².
- Medium LOCA, break area 20–175 cm²
- Small LOCA, break area < 20 cm².
- Steam generator tube rupture.

- Transients challenging the pressure relief system.
- General shutdown transients (not challenging the pressure relief system).
- Transients initiated by loss of the main heat sink.
- Transients initiated by loss of offsite power.
- Transients initiated by steam line break.
- Anticipated transients without scram.

The core damage frequency is determined without need for event trees for the following initiating events:

- Loss of cooling during shutdown.
- Interfacing systems LOCA (“V-LOCA”).
- Reactor vessel rupture.

A total of seventy sequences are analysed and quantified. The dominant contributors to the core damage frequency are listed in Table 10.10. The mean value of the total core damage frequency is estimated at 5.2 PMY. The corresponding median value is 3.6 PMY. The upper confidence limit is estimated at 13 PMY and the lower confidence limit at 1.1 PMY.

The dominant sequences are initiated by a small pipe break in the main coolant system with failure to reduce pressure or failure to change over to the recirculation mode. Next in importance are the case of steam generator tube rupture with failure of depressurization and a large LOCA with failure of recirculation. It should be noted that transients are not dominant. This is ascribed to the fact that Ringhals 2 has two feedwater systems and two

TABLE 10.10. *Dominant core damage sequences for Ringhals 2 according to the 1983 safety study (1009). Frequencies and probabilities are point-estimated mean values*

Event	Frequency (per year)	Failed safety function	Safety function failure probability	Core damage frequency (PMY)
Small LOCA	1.1E-2	Depressurization	1E-4	1.1
Small LOCA	1.1E-2	High head recirculation	8.5E-5	0.94
Steam generator tube rupture	9.4E-3	Depressurization	1E-4	0.94
Large LOCA	4.0E-4	Recirculation	2.3E-3	0.92
Medium LOCA	8.2E-4	Recirculation	3.4E-4	0.28
Reactor vessel rupture	2.7E-7			0.27
Small LOCA	1.1E-2	Decay heat removal	2.4E-5	0.27
Large LOCA	4E-4	Safety injection	2.4E-4	0.098
Steam line break in auxiliary system building	4E-4	Break isolation	2.3E-4	0.090
Large LOCA	4E-4	Containment spray	2.2E-4	0.088
Loss of auxiliary power	7E-1	Auxiliary feedwater	3.4E-8	0.024

turbines which makes total loss of feedwater and total loss of main heat sink very improbable.

Small LOCAs contribute more than medium and large LOCAs to the core damage frequency because of their higher initiator frequency due to the large number of small pipes in the plant. Events initiated by the inadvertent opening of a pressure relief valve are also considered as small LOCAs. The dominant sequence is characterized by failure of the reactor operator to reduce the pressure in the primary system and by unsuccessful realignment to high-head recirculation when the storage tanks are empty. In the second dominant sequence, depressurization is successful but the operator fails to connect the low-head recirculation system. The largest failure source is a common cause failure making it impossible to start the low-head pumps.

Loss of offsite power makes a relatively small contribution to the total core damage frequency. Short-term interruption of onsite power can occur as a result of salt storms in the winter-time, but the main offsite grid is not affected, and power can usually be restored within 10 minutes. Long-term loss of offsite power initiates reactor scram and start-up of the diesel generators which feed the plant's 6 kV network.

In the event of station blackout (cf 9.6.6), a LOCA event can result due to failure of the main coolant pump shaft seals (cf 5.2.1). If power is not restored within about 1 hour and the steam-driven auxiliary feedwater pump is not operable, the core will be uncovered within one hour. If the pump is operable, power must be restored within about 3 hours so that safety injection can be carried out and core meltdown avoided.

Anticipated transients without scram do not contribute significantly to the core damage frequency. This relates to the fact that if the scram failure is due to malfunction of the reactor protection system actuating circuits (cf 8.2.1), the operator can initiate scram manually. If the control rods are still not inserted, shutdown can be achieved by using the boron injection system.

Omitted or erroneous operator action contributes significantly to many of the dominant core damage sequences. In order to examine the effects of human error more closely, a sensitivity analysis was performed where the assumed conditions were varied within wide limits. The operator error model used is shown in Fig. 10.13, curve B. The diagram indicates that the probability of operator error is related to the time available for a particular action. The larger the time, the smaller the error probability. For times > 100 minutes, a constant minimum error probability of 10^{-4} per demand is assumed in the base case.

During the sensitivity analysis, both the minimum error probability (curves A and C) and the slope (curve D) were varied. The results are presented in Table 10.11, which shows that if the minimum error probability is increased to 10^{-3} per demand, the total core damage frequency is increased by a factor of 7, while a decrease to 10^{-5} per demand reduces the core damage frequency by only one-third. If an error factor of 10 is applied to

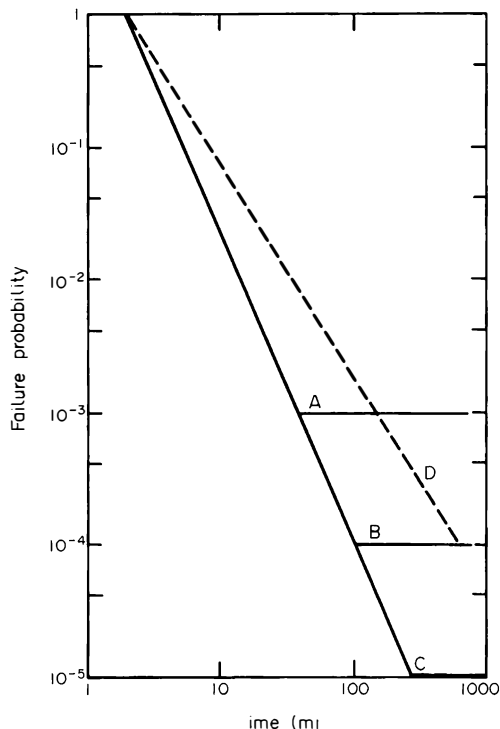


FIG. 10.13. Probability of operator error versus available time. From *Ringhals 2 Safety Study*, Swedish State Power Board, 1983

TABLE 10.11. *The effects of operator error on the total core damage frequency for Ringhals 2*

Probability of operator error	Total core damage frequency (PMY)
Base curve (Fig. 10.13, curve B)	5.1
Base curve with minimum failure probability 10 ⁻³ (A)	40
Base curve with minimum failure probability 10 ⁻⁵ (C)	4.0
New curve with higher failure probability (D)	33

Source: *Ringhals 2 Safety Study*, Swedish State Power Board, June 1983

the base curve, the uncertainty will be 1.1–15 PMY. If the same factor is used on curve A the upper limit will be 1500 PMY, i.e. 1.5 cases of core damage per thousand reactor years. These results show that the core damage frequency is very sensitive to the assumptions for human error.

A sensitivity analysis was also carried out for common cause failures. If

all beta-factors (cf 10.2.5) are zero, i.e. if no common cause failures are assumed to occur, the total core damage frequency is reduced from 5.2 to 4.5 PMY. If instead all beta factors are set equal to 0.1, the frequency increases to 8.1 PMY. This indicates that the assumptions made for common cause failures are not critical for the end result.

10.3.7 Barsebäck 1

The Barsebäck nuclear power station has two practically identical BWR units, each with a net output of 570 MWe (later increased to 595 MWe). Unit 1 started regular operation in July 1975 and Unit 2 in June 1977. A safety study for Unit 1 was completed in 1984 for internal events in the plant, i.e. PRA level 1 (1010). The results are in all essentials also valid for Unit 2.

Initiators were grouped into five LOCA and five transient categories. Event trees were drawn for sequences initiated by large, medium and small pipe breaks and loss of auxiliary power, loss of feedwater, and other events leading to scram. The event trees usually contain general sequences for the basic safety functions: reactor shutdown, pressure relief, coolant make-up and decay heat removal. The general sequences are successively broken down via various failure modes into basic events for which the probability can be determined from operating experience.

Analyses were carried out of both system-related and environment-related dependences. The dependences were ranked into three groups and quantified using the beta-factor method (10.2.5):

- moderate dependence $\beta = 0.1$,
- small dependence $\beta = 0.05$,
- insignificant dependence $\beta = 0.01$.

Three types of human error were considered, namely inadvertent, omitted and erroneous manoeuvres. The probability for unsuccessful manoeuvres was related to the time available for the operator as follows:

<i>Required action</i>	<i>Failure probability</i>
within 0.5 hour	1.0 per demand
within 4 hours	0.1 per demand
within 24 hours	0.01 per demand

The linking of the failure probabilities with the time available is based on the fact that reactor coolant make-up is required within 0.5 hour and condensation pool cooling within 4 hours. For manual reactor shutdown which must be accomplished in a shorter time than 0.5 hour, lower failure probabilities than 1 were assumed, however, depending on the particular case.

TABLE 10.12 *Dominant core damage sequences for Barsebäck 1 according to the 1985 safety study (1010). Frequencies and probabilities are point-estimated mean values*

Event	Frequency (per year)	Failed safety function (cf Table 10.9)	Safety function failure probability	Core damage frequency (PMY)
Large internal pipe break	3.0E-4	Y	2.8E-2	7.8
Medium internal pipe break	9.0E-4	W	2.8E-3	2.5
Unisolated external pipe break	2.0E-6	—	—	2.0
Loss of feedwater	0.8	UV	3.6E-7	0.3
Loss of auxiliary power	0.05	UVQ	5.3E-7	<0.1

Some quantitative results are presented in Table 10.12. The total core damage frequency is estimated at 13 PMY. Some kind of LOCA is responsible for 95% of the frequency. The largest contributor is represented by a large pipe break inside the containment with unsuccessful back-flushing of the suction strainers in the condensation pool. Common cause failure in the decay heat removal chain also contributes. Unisolated external pipe breaks are also estimated to result in a relatively high core damage frequency. This type of break occurs in a suction line to the shutdown reactor cooling system, which—if the isolation valves fail to close—leads to the escape of incoming water into the reactor building without forming a closed circuit.

10.3.8 Comparison of plant safety studies

There is no generally accepted approach to systematic reliability studies. So far, the scope and structure of the studies have varied greatly. In some cases, detailed event trees and reduced fault trees have been used, while in other cases relatively simple event trees have been combined with detailed fault trees. Common cause failure and human error have been treated in different ways. For these reasons, absolute values of core damage frequencies must be treated with caution. In general, frequencies lower than 0.1 PMY should be viewed with scepticism since there is a high probability at this level that a failure mode or failure source has been overlooked. Table 10.13 presents some results of plant safety studies for internal events (“reactor faults”). The uncertainty of the data and analysis as well as the differences in plant design must be borne in mind when comparing the results.

Table 10.14 indicates the estimated uncertainty at a core damage frequency level of 10–100 PMY. The upper bound (95th percentile) implies that the real core damage frequency is lower than this value with a 95% probability. Similarly, the real value is higher than the lower bound (5th percentile) with a 95% probability.

TABLE 10.13 *Estimated total core damage frequencies (mean value) for internal initiators*

Type of reactor	Country	Unit	Power (MW _{el} , net)	Commercial operation	Core damage frequency (PMY)	Ref.
BWR	USA	Peach Bottom-2	1051	1974	8.2	1011
	USA	Grand Gulf-1	1250	1985	29	1011
	S	Barsebäck 1	600	1975	13	1010
	S	Ringhals 1	750	1976	2.5	1003
	S	Forsmark 3	1063	1985	7	1007
PWR	USA	Surry-1	775	1972	26	1011
	USA	Zion-1	1040	1973	150	1011
	D	Biblis B	1240	1977	90	1005
	S	Ringhals 2	800	1975	5	1009
	UK	Sizewell B	1175	^a	4.1	1012

^aUnder construction.TABLE 10.14 *Estimated uncertainties in the core damage frequency*

Unit	Mean value	Upper bound (95th percentile)	Lower bound (5th percentile)
Peach Bottom-2	8.2	24	1.3
Grand Gulf-1	29	100	3.7
Surry-1	26	67	7.1
Biblis B	90	300	10

The tables indicate that although there is substantial variation in individual core damage frequencies, there is no significant difference between the reactor types.

A U.S. reevaluation study (1011) shows that the detailed results are highly plant-specific and depend not only on the particular design configuration but also on the state of development of the PRA methodology. This is illustrated in Fig. 10.14 which compares results for the Reactor Safety Study reference plants. The total core damage frequencies differ considerably, and in particular the contributions of dominant sequences. The differences arise from design changes and modelling improvements that have taken place since the Reactor Safety Study was published in 1975. It should also be noted that the reevaluation study uses mean values to represent frequencies, whereas the Reactor Safety Study generally used median values. The mean values for the Reactor Safety Study are somewhat higher than the values illustrated in Fig. 10.14.

LOCA events are clearly dominant for the Swedish boiling water reactors Ringhals 1 and Barsebäck 1, and the pressurized water reactor Ringhals 2. The small contribution from transients is explained by the differences in plant design between the Swedish and U.S. boiling water reactors as well

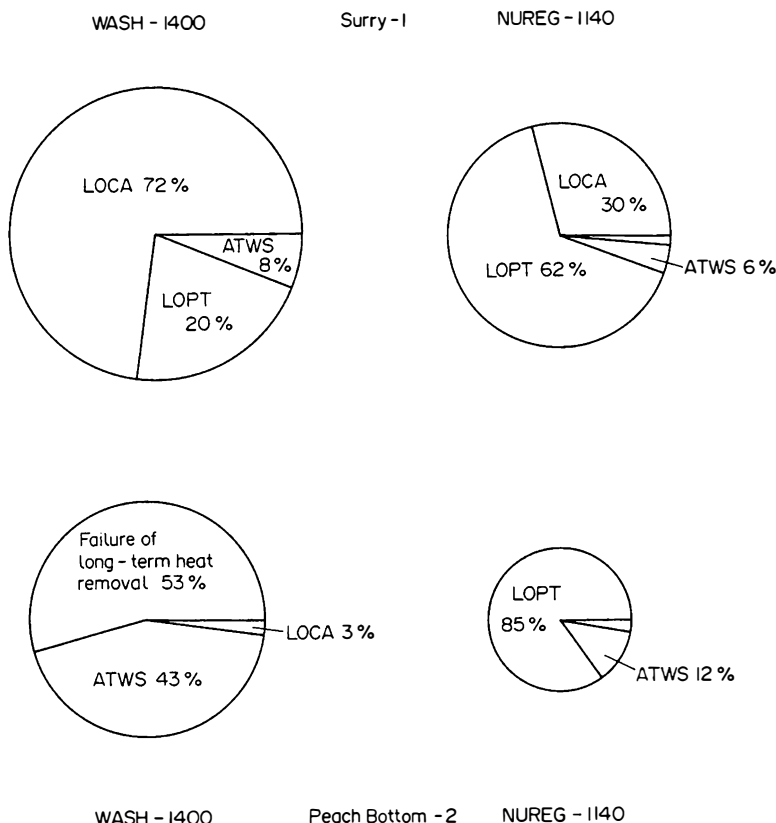


FIG. 10.14. Comparison of core damage frequencies due to internal initiators in Surry-1 and Peach Bottom-2. The area of the circles is proportional to the total core damage frequency. LOPT = loss of power transient

as by the double turbine and feedwater systems (in the Ringhals reactors), the stronger external grid, the larger pressure relief capacity and redundancy for reactor shutdown.

Ringhals 1 and Barsebäck 1 belong to the same reactor generation. Basically the same methods were used in the safety analyses. A valid comparison can therefore be made (Fig. 10.15). In both cases, the contributions from transients are less than 1 PMY. The contribution from LOCA events is greater than 10 PMY for Barsebäck and about six times less for Ringhals. This is mainly due to the fact that a higher probability for unsuccessful suction strainer back-flushing during large LOCA was obtained in the Barsebäck study. Also the probability for an external pipe break in the shutdown cooling system with failure of isolation valve closure was estimated to be higher in Barsebäck 1 than Ringhals 1.

The total core damage frequency for internal events in Forsmark 3 (F3)

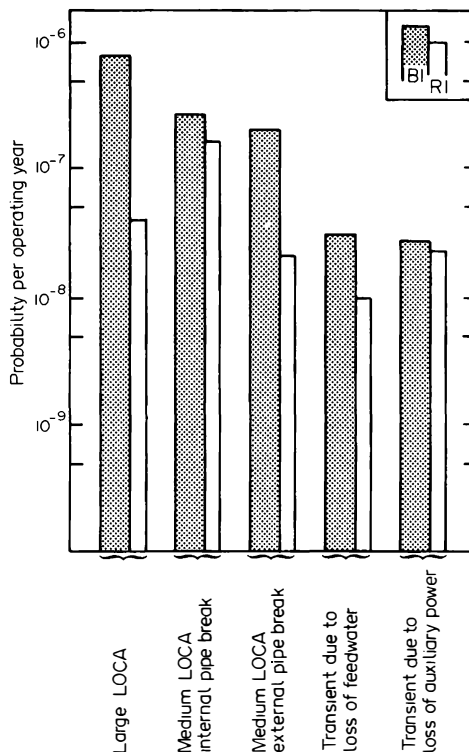


FIG. 10.15. Core damage frequencies for dominant sequences (internal initiators) in Barsebäck 1 and Ringhals 1

was estimated at 7 PMY as compared to 2.5 PMY for Ringhals 1 (R1). With regard to the uncertainty of the analysis, the difference is not significant. However, the absolute values are not directly comparable since more conservative assumptions were used for human error and common cause failure in the F3 study. If similar assumptions as in the R1 study are used, the predicted core damage frequency for F3 becomes 1.3 PMY. The remaining difference is due to differences in plant design, such as:

- R1 has external and F3 internal main recirculation pumps;
- F3 has a more complete and consistent segregation of redundant safety-related equipment;
- R1 cooling systems have a $2 \times 100\%$ capacity as compared to $4 \times 50\%$ in F3;
- R1 reactor protection systems logics has 2-of-3 coupling while that of F3 has 2-of-4;
- R1 has two turbine and feedwater systems as opposed to one in F3;

- R1 has a high-head safety injection system which can operate at full reactor pressure, while the F3 emergency core cooling system requires low pressure for operation;
- R1 has automatic depressurization during transients, in contrast to F3.

A comparison of predicted core damage frequencies is shown in Fig. 10.16. The very low contribution ($< 1\%$) from LOCA events in Forsmark 3 is due to the fact that no large pipes are connected to the reactor vessel below the upper edge of the core. This results in more favourable emergency core cooling conditions than in the external pump reactor. The contribution from transients due to insufficient coolant make-up is about 10 times greater in F3 than in R1. This can be explained by the effects of the special coolant make-up system in R1 (cf 10.3.5) and of the high-head safety injection system and the automatic depressurization function in this reactor.

In general, a large core damage frequency is not synonymous with a large release of radionuclides to the environment. The magnitude and composition of a large release, if any, in connection with severe core damage depends on the interaction of the core melt with reactor containment, which

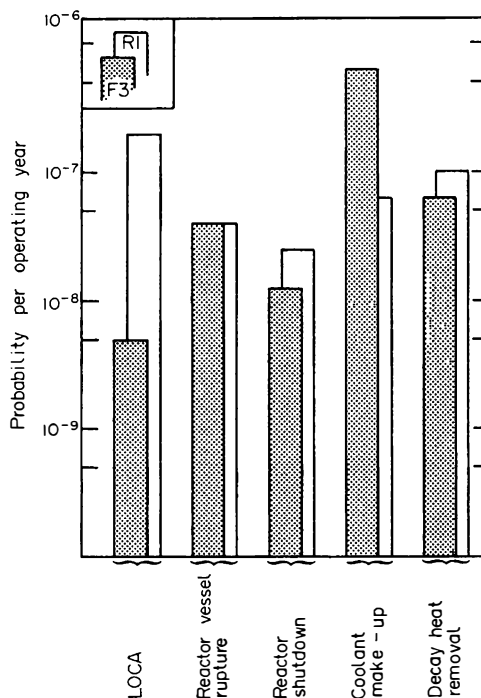


FIG. 10.16. Core damage frequencies (internal initiators) for Forsmark 3 and Ringhals 1, grouped according to (unsuccessful) basic safety function

is determined by the particular accident sequence. These matters are treated in Chapter 11.

10.4 Fracture Probabilities

The plant analyses show that some kind of LOCA makes a dominant contribution to the core damage frequency in many cases. If the reliability of the safety systems is further improved, the core damage frequency approaches a value determined by the probability of reactor pressure vessel rupture. Vessel rupture can be considered as a kind of LOCA where the amount of coolant lost exceeds the capacity of the emergency core cooling systems.

10.4.1 Pipe break

In the Reactor Safety Study, a reactor plant is estimated to contain about 100,000 metres of pipeline. Some of these are *high-energy pipes*, i.e. they are pressurized to at least 2 MPa or have a temperature of at least 100°C during normal operation. In some of the high-energy pipes, a break will result in a LOCA, since they are part of or connected to and pressurized from the main coolant system.

High-energy pipelines are designed with large safety margins and much attention to quality. Nevertheless, the safety requirements specify that pipe breaks should be postulated to occur and the reactor so designed that the consequences can be handled without compromising safety. *Pipe criteria* have been established which determine where and under which conditions pipe breaks shall be assumed to occur. Regarding LOCA, breaks shall be postulated up to a size corresponding to a double-ended break of the largest pipeline in the main coolant system.

The probability of a pipe break as initiator of a LOCA was estimated in the Reactor Safety Study on the basis of nuclear and non-nuclear plant data available at that time (Table 10.15).

TABLE 10.15 *Pipe break probabilities according to the Reactor Safety Study (1004)*

Category	Pipe diameter mm	Failure probability (per operating year)			
		Median (50th percentile)	Upper bound (95th percentile)	Lower bound (5th percentile)	Mean value
Large break	>150	10^{-4}	10^{-5}	10^{-3}	3×10^{-4}
Medium break	50–150	3×10^{-4}	3×10^{-5}	3×10^{-3}	9×10^{-4}
Small break	12–50	10^{-3}	10^{-4}	10^{-2}	3×10^{-3}

Since the statistics are insufficient, the confidence intervals in Table 10.15 are relatively large. However, no reason has so far been found to revise the values of the Reactor Safety Study. These values have therefore been used in most of the subsequent studies.

No large pipe break has yet occurred in the main coolant system of a light water reactor. In December 1986 a large break occurred in secondary side piping in the Surry-2 PWR. The break involved a 1.8–3.6 m long elbow section of a 450 mm diameter, 12.7 mm thick feedwater line leaving a feed-water heater. Inspection revealed that the pipe wall had thinned due to erosion and corrosion during 13.5 years of operation.

Data from non-nuclear plants indicate that the fracture probability for large pipes is less than 4×10^{-4} per reactor year with 99% confidence (1013). For small pipes, there is enough experience from nuclear power plants to validate the mean value, 3×10^{-3} per reactor year, of the Reactor Safety Study.

The pipe break probability can also be estimated by way of probabilistic fracture mechanics (cf 3.5.2). A distinction is made between spontaneous fracture through unstable cracking due to fatigue or corrosion, and indirect fracture caused by external events such as earthquake. The analysis of both types of fracture results in lower fracture probabilities (1014) than those of the Reactor Safety Study. At the same time, leakage probabilities are obtained which are greater than the fracture probabilities by several orders of magnitude.

The fracture mechanics analysis and the increased operating experience indicate that the pipe break probabilities so far used in safety studies are conservative. In addition, the “leak-before-break” principle is confirmed, i.e. the probability of leakage is much greater than the probability of fracture. This means that a large break need never occur since it would be preceded by leakage which can be detected. This principle has led to some relaxation of the safety design requirements for the pressurized water reactor primary system (1015).

10.4.2 Pressure vessel rupture

Reactor pressure vessels are designed and manufactured according to generally accepted standards with large safety margins against rupture (cf 3.5.2). Not only the normal operation of the reactor is taken into consideration, but also the particular stresses that the pressure vessel is exposed to under upset and fault conditions. In addition, changes in the properties of the material during reactor operation are taken into account. Hydrostatic testing of the vessel is conducted before start-up, and inspections are regularly carried out during its lifetime.

However, the possibility of rupture cannot be ruled out completely. In principle, the fracture probability can be estimated in three ways, based on:

- operating experience for reactor pressure vessels;
- accident statistics for conventional pressure vessels;
- probabilistic fracture mechanics.

There is still not enough operating experience from reactor vessels for a meaningful assessment of the fracture probability. This is expected to remain the case until around the turn of the century.

Studies of the experience from conventional pressure vessels have been carried out in West Germany, Great Britain and USA (1016). These studies show that the rupture probability of a non-nuclear vessel is in the interval 10^{-3} – 10^{-4} per pressure vessel and year with 99% confidence. However, it is not possible to apply this experience directly to reactor pressure vessels, since they are manufactured to other, more stringent standards and are subjected to more thorough control before and after start-up.

Experience from non-nuclear pressure vessels shows that the most important cause of rupture is the occurrence of crack-like faults in the material during the manufacturing process. The cracks can grow during operation due to mechanical, thermal or corrosion-assisted fatigue. Many of the factors affecting crack growth are statistically distributed and amenable to analysis using probabilistic fracture mechanics. Such studies have been carried out in several countries including Sweden (1017). The results indicate fracture probabilities in the interval 10^{-6} – 10^{-8} per reactor vessel and operating year.

In the Reactor Safety Study, the probability of reactor vessel rupture was estimated at 10^{-6} per year (median value) with a confidence interval of 10^{-5} – 10^{-7} . The corresponding mean value is 2.7×10^{-6} which has been used in most subsequent safety studies.

10.5 External Events

For a complete assessment of plant safety, the effects of external events must also be considered. External events can be caused by natural phenomena such as earthquake, wind storm, flooding, or human action such as aircraft crashes, chemical explosion, sabotage, or war. Onsite fire and flooding are usually also considered as external events.

Reactors are designed to withstand extreme external events (cf 9.8.1). If, however, essential safety functions should fail simultaneously or as a result of the external event, core damage may occur. The corresponding core damage frequency can be estimated with the same methods as previously described. In addition, special methods are required for characterizing the external event and its effects on the plant. In this section, the probabilistic approach for analysing earthquake, fire and flooding is outlined and some results are presented.

10.5.1 Earthquake

A probabilistic earthquake analysis consists of four steps:

- assessment of the seismic risk at the plant site;
- dynamic analysis of the seismic response of the plant;
- determination of the resistance of components and systems to seismic loadings;
- analysis of relevant core damage sequences using event tree-fault tree methodology.

Models based on known geological and seismic conditions as well as on historical data have been developed to characterize earthquakes. The seismic risk can be expressed as the probability (per year) of exceeding a particular peak ground acceleration (Fig. 10.17). The exceedance frequency decreases rapidly as the ground acceleration increases, and the uncertainty in the probability estimate increases. The curves refer to conditions in Great Britain (1018) which, like Sweden, is located in a region of low seismic activity.

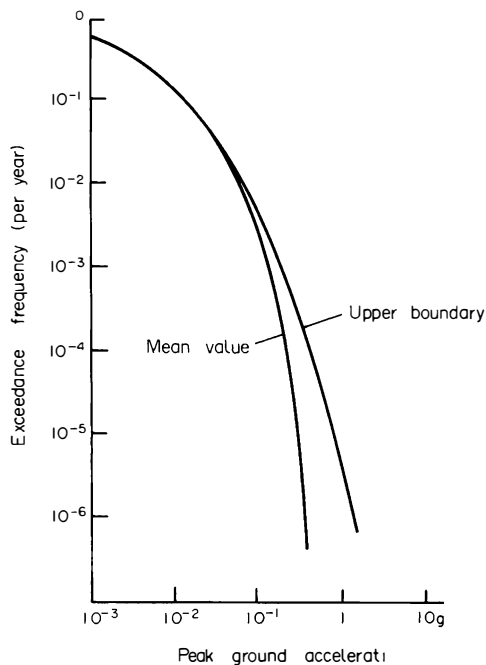


FIG. 10.17. Risk curves for earthquakes, showing the annual acceleration exceedance probability. From S F Hall et al, *Nucl. Energy*, Vol 24, No 4, August 1985

In addition to the peak ground acceleration, an earthquake is characterized by its frequency spectrum and energy content (duration). During the structure mechanics analysis (see 9.8.2), the acceleration and displacement of buildings, systems and components are studied. The approach is usually deterministic, using a standard spectrum for the frequency content and the duration, scaled to a certain peak ground acceleration.

The next step is to determine the *fragility* of the plant components, i.e. their ability to withstand seismic loads. The fragility can be expressed as the probability of failure as a function of the peak ground acceleration. By using these failure probabilities, the relevant fault trees and event trees can be quantified. The result will be a probability distribution for core damage as a function of the peak ground acceleration (Fig. 10.18).

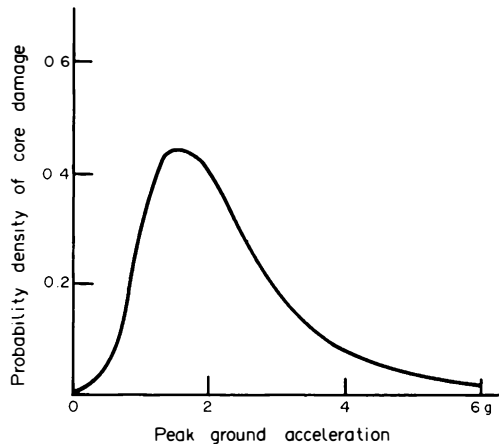


FIG. 10.18. Typical probability density distribution for core damage in the event of an earthquake

By combining information on the acceleration exceedance frequency as in Fig. 10.17, and the core damage probability density as in Fig. 10.18, the expectation value of the core damage frequency as a function of the peak ground acceleration is obtained (Fig. 10.19). As illustrated, the expectation value reaches a level which in the particular case is about 10^{-7} per year on average and about 10^{-5} per year on the upper boundary. The example shows that the uncertainty is great in estimating the contribution of earthquakes to the core damage frequency. The uncertainty is genuine, due to the inherent uncertainty in the frequency of major earthquakes. This fact is disturbing since, although the absolute contribution of earthquakes to the core damage frequency is small, the relative contribution may be great.

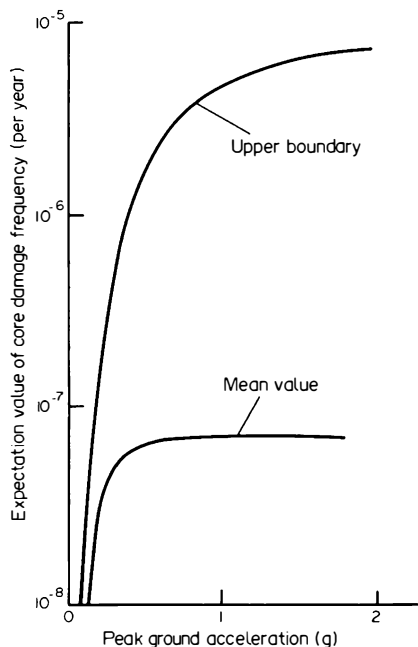


FIG. 10.19. Estimated expectation value of the core damage frequency due to earthquake for UK conditions. From J F Hall et al, *Nucl. Energy*, Vol 24, No 4, August 1985

10.5.2 Fire

Fire is usually considered as an external event even if it originates inside the plant. As for other kinds of external events, the probabilistic safety analysis comprises four steps. Firstly, critical areas where fire may occur and cause damage to safety-related equipment are identified. The fire hazard in these areas is estimated on the basis of historical data. Secondly, ways in which the fire can spread and the effect of fire-fighting measures are assessed. Thirdly, an analysis of the plant design is carried out for investigating the possible damage to plant systems and components as well as for estimating the probability of safety function failure. Finally, fault trees and event trees are quantified and the core damage frequency estimated.

Attempts have been made to estimate the frequency (probability per year) of fire in critical areas, such as the central control room, cable distribution rooms, diesel generator building, reactor containment, turbine building and auxiliary system building (1019). Statistical data on fires which have occurred are used. Unfortunately, there is no suitable scale for characterizing the intensity of a fire (like the Richter seismic scale). It is therefore difficult to establish a relation between frequency and intensity for fire in

nuclear power plants. One approach is to define a series of “typical fires” Since the number of rooms to be analysed is large, a complete fire analysis would be very comprehensive.

10.5.3 Flooding

Onsite flooding includes any unintentional flow outside the reactor containment, from rupture or leakage in the water and steam systems. Flooding analysis is similar to fire analysis. First, the layout of the plant is reviewed, including system design, location of safety-related equipment, etc. A qualitative analysis of the flow paths and the effects on the plant is then carried out. Finally, fault trees and event trees are quantified to determine the contribution of the particular flooding to the core damage frequency.

During flooding—as with fire—electrical faults will occur in the form of earth faults and short-circuits. These faults can cause safety-related equipment to malfunction. It is therefore of great importance that the electrical design of each type of safety-related component is thoroughly analysed.

10.5.4 U.S. studies

Safety studies for U.S. nuclear power plants have shown that external events are, in some cases, major contributors to the core damage frequency and dominant contributors to the environmental effects. For example, a total core damage frequency of 160 PMY was estimated for Indian Point-2 (873 MWe1 PWR), with a 50% contribution from external events (Table 10.16) (1020). The corresponding values for the Zion plant (2×1085 MWe1 PWR) are 67 PMY and 15% according to the utility's safety study (1021).

The Zion plant is situated near Lake Michigan about 60 km north of Chicago in an area which is considered to have a low seismic activity. As with other U.S. plants, Zion is designed to withstand the effects of a postulated “extreme” earthquake. The design basis earthquake for Zion is assumed to

TABLE 10.16 *Contributions to the core damage frequency for Indian Point-2*

Event	Contribution (%)
LOCA	29
Stormwinds	28
Transients	21
Fire	17
Earthquake	5

Source: *Indian Point Probabilistic Safety Study*, Consolidated Edison Company of New York, March 1982

have a horizontal ground acceleration of 0.17 g and a simultaneous vertical acceleration of 0.11 g.

The result of the probabilistic seismic analysis is shown in Fig. 10.20. The mean value is 5.6 PMY, which means that the contribution from earthquakes to the total core damage frequency is 4%. The mean value corresponds to a peak ground acceleration of about 0.5 g. Such strong ground movement is expected to result in loss of offsite power and probably also of onsite power. Leakage will then occur in the reactor coolant pump shaft seals (cf 5.2.1). The consequence will be a small LOCA resulting in core damage and reactor containment overpressure since no auxiliary systems will be available if the power supply is lost and cannot be recovered.

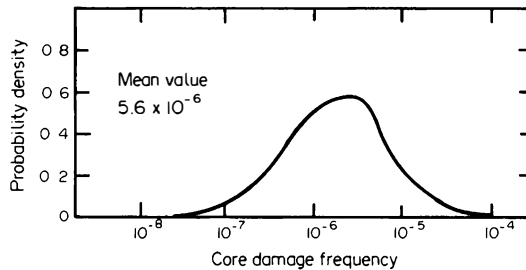


FIG. 10.20. Probability density distribution of the core damage frequency due to earthquake for Zion. From *Zion Probabilistic Safety Study*, Commonwealth Edison Company of Chicago, 1981

The fire analysis for Zion resulted in a contribution to the core damage frequency of 4.6 PMY. The most important initiating event is fire in the room containing logic circuits, relays for automatic control systems, instrumentation, etc. The greatest threat comes from the loss of instrumentation, which forces the reactor operator to safely shut down the reactor without any information on plant conditions. Fire in cable runways was also found to be a relatively large contributor. Such a fire occurred in 1975 at the Browns Ferry plant although core overheating was avoided.

Flooding caused by external pipe break or leakage of service water systems was shown to make a negligible contribution to the core damage frequency in the Zion study.

10.5.5 Swedish studies

With the exception of Forsmark 3 and Oskarshamn III, Swedish nuclear power plants are not designed to withstand seismic events. Consequently, essential plant components have relatively little resistance to seismic loadings. A preliminary analysis for Ringhals 1 showed that earthquake can

contribute significantly to the core damage frequency (1022). There is, however, a considerable uncertainty in estimating the earthquake hazard.

A flooding analysis has been conducted for Ringhals 1 (1023). The plant was shown to be relatively sensitive to flooding because of the large number of electric components that could be damaged and the large number of rooms involved. The sensitivity could be considerably reduced by redistributing vital voltages on several fuses. After these shortcomings were rectified, the contribution of flooding to the core damage frequency was estimated at 3 PMY. This can be compared with the contribution from internal initiators which was estimated at 2.5 PMY (see 10.3.5). The largest contribution comes from flooding in the turbine building due to outflow from the salt water system which incapacitates the decay heat removal system.

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11

Severe Accident Analysis

Two types of severe accidents may occur in nuclear reactors, broadly classified as core melt accidents (CMAs) and core disruptive accidents (CDAs). A CMA results from inadequate core cooling leading to core uncover, heat-up and meltdown in a time scale of hours. A CDA is caused by rapid and large reactivity insertion leading to a power excursion and fuel disintegration in a time scale of seconds. The two types are exemplified by the Three Mile Island and Chernobyl accidents. A CDA is considered practically impossible in a light water power reactor due to inherent reactivity characteristics and engineered safety features.

This chapter is devoted to the analysis of core melt accidents. The meltdown process and the behaviour of the core melt in the reactor vessel and containment are examined. The mechanisms for the release, transport and removal of radionuclides in the plant are described. The function of the reactor containment is analysed for typical meltdown accidents. The chapter concludes with a discussion of the external source terms, i.e. the magnitude and composition of the environmental releases.

11.1 Core Meltdown

A qualitative examination of the core meltdown process in the reactor pressure vessel and containment is presented in this section. The possibilities of steam explosion and hydrogen explosion are discussed. The description is based on the state-of-the-art in the mid-eighties (1101).

11.1.1 *In-vessel behaviour*

If the water level in the reactor vessel drops so that the core is uncovered, the clad temperature will rise rapidly due to the decay heat in the fuel, even if the nuclear chain reaction is interrupted. At about 900°C, the metal–water reaction (cf 3.4.6) between zirconium (in the fuel cladding) and steam begins to produce hydrogen and generate heat. The heat-up of the fuel is accelerated and once the temperature exceeds about 1200°C, the metal–water reaction will be violent and the rate of heat generation greater than that of the decay heat.

The temperature in the uncovered part of the core will increase more and more rapidly. Alloys can be formed between the fuel and the cladding which melt at a lower temperature than the uranium dioxide (melting point 2800°C). If the water level drops quickly, as after a large pipe break with failure of emergency core cooling, it will take about half an hour before parts of the core begin to melt. If the water level sinks more slowly, as during a small LOCA with failure of coolant make-up, it will take several hours before core uncover and meltdown starts.

When the fuel melts, drops of molten fuel will run along the surface of the fuel rods and solidify in the cooler lower regions which have not yet been uncovered. This may block the coolant flow in the fuel channels and accelerate the melting process. It is possible for a bowl of solidified fuel to form, which is supplied with molten fuel and fuel debris from above. The molten fuel will gradually collect at the bottom of the reactor vessel, either because the bowl collapses by its own weight or because molten fuel flows over the edge of the bowl. The greater part of the core may collect on the bottom within half an hour after the onset of melting.

If there is water left in the reactor vessel, a coolable bed of core debris will form at the bottom of the vessel. Calculations show that spherical fragments of core melt with a solidified crust can be cooled if the diameter is less than 10–15 cm. When most of the remaining water has evaporated, the fragments will melt again and form a liquid mass at the bottom of the vessel.

In boiling water reactors and most pressurized water reactors, there are several relatively thin-walled pipe penetrations in the bottom of the reactor vessel. The core melt will probably break through one of these penetrations first and fall by gravity into the space below the vessel. If the reactor pressure is low the outflow of the melt is calculated to take about 2 minutes, during which the diameter of the hole expands to an estimated 300 mm. Any remaining coolant in the lower plenum will evaporate under violent boiling.

The previous scenario is typical of core meltdown at *low pressure*, e.g. during a large loss of coolant accident with failure of the emergency core cooling systems. Core meltdown can also occur at *high pressure* in the reactor. A typical example is the case of station blackout in a pressurized water reactor. This will lead to core uncover, heat up and meltdown within a few hours, if power cannot be restored. The melt will be rapidly ejected at high pressure through failed penetrations in the bottom of the pressure vessel.

11.1.2 Steam explosion

It is well known from metallurgical industry that steam explosions can occur when hot metal or metal oxide falls into water. The melt disintegrates into small particles a thousandth of a millimetre in diameter, with a very large contact surface to the water. If the disintegration and mixing with

water takes place within a few thousandths of a second, spontaneous evaporation can occur, which results in a steam explosion.

If the mixing and heat transfer occur over a more extended period, perhaps tens of seconds instead of milliseconds, a process called a *steam spike* results. Steam spikes are not accompanied by the shock waves and fine fuel fragmentation that are characteristic of steam explosions. A steam spike is not expected to damage the fuel channels or the reactor pressure vessel.

Whether or not steam explosions can occur with enough force to breach the reactor pressure vessel when the core melt falls into water in the lower plenum of the vessel has been a subject for much concern. In the Reactor Safety Study, some ten tons of core melt were assumed to fall into the lower plenum within a very short period of time with instantaneous disintegration and mixing with water, causing a steam explosion with an energy efficiency of at least 10%. A layer of water was also assumed to have formed above the melt–water mixture, and to have been thrown like a piston towards the reactor vessel head with enough force to have blown off the head which then blasted a hole in the reactor containment.

Subsequent investigations have shown the Reactor Safety Study description to be unduly conservative (1102). Firstly, it is difficult to imagine that the entire mass which would fall to the bottom of the vessel would be completely molten. It is more likely that the molten fuel would gradually run down to the bottom of the vessel as described in the previous section. Secondly, for energy-related reasons, it is hardly possible for a 10-ton molten mass to disintegrate and mix with water in the short period of time required to cause a steam explosion. Thirdly, the model of a compact water layer transferring energy to the reactor vessel head is oversimplified. It is difficult to understand how such a layer could arise in the first place. If it did arise, it would break up during the explosion or when it passed through the reactor vessel internals.

The conclusion is that a massive steam explosion, violent enough to rupture the reactor pressure vessel, is physically impossible on the basis of present evidence. Limited steam explosions involving at most a few hundred kilograms of molten fuel cannot be ruled out, however. Explosions of this size would not damage the reactor vessel.

A distinction should be made between the situation where a coherent melt falls by gravity into water and the case where a severe reactivity-induced accident causes fuel disruption and intensive fuel–coolant interaction (cf 3.4.7). In the latter case the molten fuel is finely fragmented when mixing with the water and a powerful steam explosion may result as evidenced by experiment and the Chernobyl accident (see 13.7.4).

11.1.3 Processes in the reactor containment

The reactor vessel might fail about 1 hour after the onset of core melting. A few hundred tons of molten core material then escapes into the reactor containment. The melt will come into contact with the concrete floor under the reactor vessel. In pressurized water reactors, the region under the vessel is known as the reactor cavity (Fig. 5.8). Any water in the cavity will boil away and contribute to pressure build-up in the containment. The melt will then interact with the concrete. The melt may also form coolable fragments under water in the bottom of the cavity if water from the accumulator tanks or the containment sump is available.

First generation Swedish boiling water reactors have a drainage pipe in the floor of the pedestal region below the reactor vessel through which most of the melt would flow thus falling into the condensation pool which occupies the entire bottom region of the containment (Fig. 11.1). The molten fuel would then disintegrate forming fragments which are cooled without net steam formation. Large steam explosions which might damage the containment are considered impossible as mentioned above. If the pool is not effectively cooled, the water will boil off and the steam contribute to the pressure build-up in the containment.

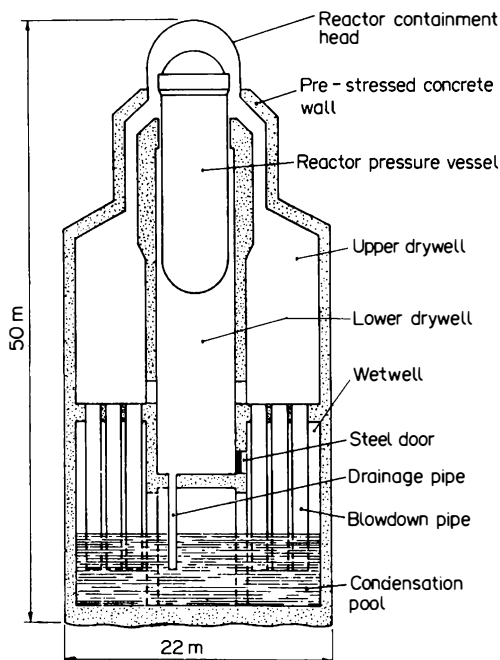


FIG. . Schematic reactor containment of a boiling water reactor of the Barseback type

In Forsmark-type boiling water reactors, the condensation pool forms an annular region close to the walls of the containment (Fig. 4.6). In this case, the core melt would fall onto the floor of the lower drywell, melt through the steel liner and interact with the basemat concrete. Penetration of the steel doors of the air locks (Fig. 4.6), or of any of the numerous inlets in the lower drywell, would also occur. In order to avoid severe melt–concrete interaction, the lower drywell is flooded with condensation pool water, if necessary. Special protection barriers for the weak points are also provided.

When the hot core melt comes into contact with concrete, free and chemically bound water in the concrete will evaporate. The concrete itself will also disintegrate through chemical reactions. Non-condensable gases, particularly hydrogen, and in certain types of concrete carbon dioxide as well, will then be formed. The steam and gases contribute to the pressure build-up in the containment. The melt will erode the walls and base of the containment at an initial rate of a few centimetres per minute. After about an hour, the rate will be considerably reduced due to the drop in temperature of the melt when it mixes with molten concrete, since the chemical reactions between melt and concrete require heat. After about 24 hours, the melt will have solidified although it will continue to erode the floor at a slow rate since its solidification temperature is higher than the melting point of concrete (about 1500°C as compared to 1200°C).

The detailed processes during melt–concrete interaction are still not completely known. It cannot be predicted with certainty whether or not the concrete basemat of the reactor building, which is several metres thick, will be melted through. However, it is evident that the so-called China syndrome, where the melt would successively eat through the ground, is a myth.

11.1.4 Hydrogen explosion

Hydrogen is formed during core meltdown and melt–concrete interaction. The extent of hydrogen formation depends strongly on the prevailing conditions. During meltdown, the availability of steam and the temperature of the cladding determine the metal–water reaction rate. The amount of hydrogen produced may correspond to the reaction of 10–25% of the zirconium in the core. During melt–concrete interaction, temperature and time are the decisive factors. If the interaction continues for a long time, all the zirconium metal in the core melt will react.

The hydrogen generated within the primary system is transported either as a gas or dissolved in the coolant. The gaseous hydrogen may accumulate at high points in the primary system and interfere with the circulation of the coolant, as happened in the Three Mile Island accident (see 13.5.2).

Ultimately, the hydrogen and steam generated within the primary system will be released into the containment and contribute to the pressure build-

up there. The venting hydrogen may ignite and burn in the vicinity of the release point. If the hydrogen does not burn, it will mix with any air, steam or hydrogen already present in the containment. If the mixing is rapid, the hydrogen concentration might rise approximately uniformly over the entire containment volume. If the mixing is slow, a high concentration of hydrogen could develop locally.

If the gas concentration and other conditions are within certain limits and an ignition source is present, combustion will occur. "Hydrogen explosion" is a rather imprecise term which is applied to various forms of combustion. *Deflagration* is a form of combustion, where the flame progresses at subsonic speed relative to the unburned gas which is heated to reaction temperature by thermal conduction from the hot burned gas (1103).

Under certain conditions, combustion takes place extremely rapidly within a shock front moving at supersonic speed into the unburned gas which is heated to combustion temperature by shock wave compression. This process, which is known as *detonation*, can cause high dynamic and static loads on the containment and internals. A global detonation would be quite serious.

In a ternary diagram for mixtures of air, steam and hydrogen a flammability limit can be identified (Fig. 11.2). Combustion is possible for mixtures within the flammability limit. If the concentration falls within the

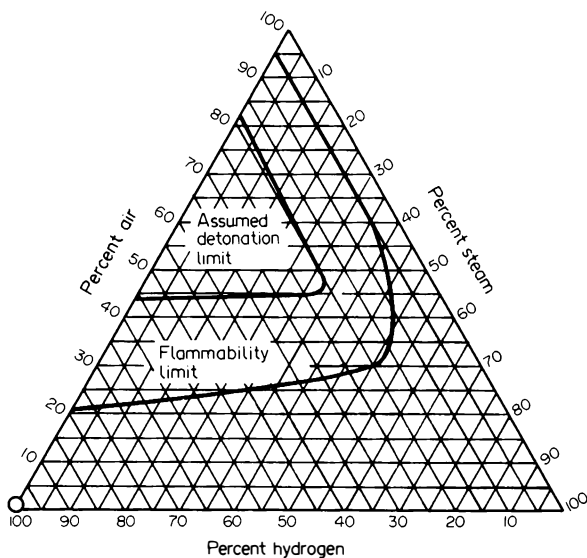


FIG. 11.2. Flammability and detonation limits for mixtures of air, hydrogen and steam. From Z M Shapiro and T R Moffette, *Hydrogen Flammability Data and Application to PWR Loss of Coolant Accident*, USAEC Report WAPD-SC-545, September 1957

indicated detonation limit, a detonation can occur. The diagram presents an oversimplified picture, however. In reality, the flammability and detonability limits are not unique functions of the concentrations but depend also on initial and boundary conditions, e.g. geometry and size.

An appropriate ignition source is required to set off a detonation. The source must be stronger than that required for deflagration. Therefore, deflagration is more likely to occur than detonation in the “detonation range” In addition, the presence of steam makes ignition more difficult and suppresses the pressure and the reaction rate.

The containments of Swedish boiling water reactors are *inerted*, i.e. filled with nitrogen, which prevents hydrogen explosion. Pressurized water reactor containments are air-filled, and a global detonation could be destructive. However, because of the large volume, it is doubtful whether a critical mixture can be obtained over the entire containment. There may be a risk of limited detonations due to local critical conditions, but these detonations are not strong enough to damage the containment.

In order to prevent critical mixtures from occurring, controlled hydrogen combustion has been introduced. Since the range of the flame is limited, ignition must be effected in several regions to ensure complete combustion. This type of combustion does not damage the containment.

11.2 Thermohydraulic Analysis

This section describes the quantitative analysis of the core meltdown process. Some results of case studies are presented.

11.2.1 Calculation models

Knowledge of basic phenomena and mechanisms connected with core meltdown processes has successively increased. Computational models have been developed and verified by experiments. The models are incorporated into computer codes which describe thermohydraulic and other processes in the reactor vessel starting from an assumed initiating event. The codes are based on fundamental equations for the conservation of mass and energy.

MAAP is a typical severe accident analysis code developed in the USA (1104). This code models the progression of core meltdown in the reactor vessel, the melt-through of the vessel, and the interaction between the melt, concrete and water in the containment. The code calculates pressure and temperature in the containment until containment failure due to overpressure or melt-through, or until the core debris is steadily cooled with the containment intact. MAAP mainly consists of relatively simple models of a general nature which can be replaced as better models are developed.

The processes during core meltdown depend to a high degree upon the characteristics of the particular plant, e.g. the safety system design and the

containment configuration. Two boiling water reactors and two pressurized water reactors, representative of U.S. conditions, were originally modelled in MAAP. Special code versions have since been developed for all Swedish plants (1105). Some examples of the results obtained are presented below.

11.2.2 BWR case studies

One severe accident case studied is the transient caused by prolonged *station blackout*. This transient is initiated by the loss of offsite power. Since the transition to house load operation and the start-up of the diesel generators for emergency power are also assumed to fail, all core and containment cooling is lost. However, battery power is supposed to be available for the closure of isolation valves and the opening of pressure relief valves for automatic depressurization.

Calculations have been performed for all Swedish power plants. The results are summarized in Table 11.1. It is interesting to compare the results for Ringhals 1 (R1) and Oskarshamn II (OII), which have containments of the type shown in Fig. 11.1, with those for Forsmark 1 and 2 (F1/F2) as well as Forsmark 3 and Oskarshamn III (F3/OIII), which have annular condensation pools as shown in Fig. 4.6.

In R1 and OII, reactor scram is initiated by the loss of auxiliary power. The primary coolant boils off due to the decay heat. The steam is discharged through the relief valves to the condensation pool to maintain a constant reactor pressure at about 7 MPa. Since there is no coolant make-up, the water inventory in the primary system decreases. After about half an hour the core begins to uncover. A few minutes later the water level has dropped to half the core height. Automatic depressurization is then actuated. The reactor pressure rapidly falls with violent boiling and complete core un-

TABLE 11.1. *Summary of MAAP results for station blackout transients in Swedish BWRs*

		OI	R1	OII	F1/F2	F3/OIII
Reactor scram	sec	1.1	1.4	1.5	2.5	12.7
Core uncovered	min	49	32	34	21	28
Automatic depressurization	min	70	67	42	—	—
Start of core meltdown	hr	3.2	2.6	2.5	1.7	2.5
Vessel melt-through	hr	3.7	3.8	2.9	2.0	3.0
Containment failure	hr	52	38	54	28	42
Peak dry well temperature	°C	167	237	247	567	767
Total mass of hydrogen produced	kg	58	91	53	1600	1460
Corium penetration depth in pedestal concrete floor	m	0.09	0.24	0.21	1.8	1.12

Source: K Becker (Editor). *RAMA Containment Group Final Report*, Studsvik, January 1985

covery. The fuel heats up, the metal–water reaction begins and melting starts after about 2.5 hours. Core debris will collect on the core support plate, and when 25% of the core is molten, the plate is assumed to fail. The core melt then falls into the plenum, where melt-through occurs after an estimated 2.9 hours (OII). The melt flows down into the pedestal region below the pressure vessel. Hydrogen is generated as a result of the melt–concrete interaction. Most of the melt continues further down into the condensation pool, where coolable fragments are formed which heat up the pool water. The entire core is predicted to have left the reactor vessel after about 13 hours. The failure pressure of the containment is reached after 38 hours in R1 and 54 hours in OII.

Scram in F1/F2 is assumed to be initiated by the loss of auxiliary power and in F3/OIII by low water level in the reactor vessel. The core is uncovered after about 20 minutes and melting begins after about 2 hours. Reactor vessel melt-through then occurs at high pressure, since depressurization is not automatic in these reactors. The molten fuel is ejected onto the floor of the lower drywell. The entire core is molten after about 7 hours. The melt attacks the concrete, forming hydrogen which increases the containment pressure. The volume of the lower drywell is smaller in F1/F2 than in F3/OIII, so that the attack on the concrete is more violent. The containment is assumed to fail at a pressure of 1.0 MPa, which is estimated to occur after 28 hours in F1/F2 and 42 hours in F3/OIII.

In the calculations it is assumed that the inlets and airlocks in the drywell (see Fig. 4.6) are protected against attack by the melt. If they had been unprotected and directly exposed to the melt, they would have been penetrated almost simultaneously with reactor vessel melt-through. It should be noted that modifications have been introduced in the Forsmark-type plants so that condensation pool water is supplied to the lower drywell in case of a core melt accident (see 14.3.2).

The pressure in the (upper) drywell is shown in Fig. 11.3. The containments are assumed to fail at about 1.0 MPa except in the case of R1, where the failure pressure is calculated at 0.7 MPa (cf Table 11.5). The pressure rise in OI, OII and R1 is caused by steam generation in the condensation pool while in F1/F2 and F3/OIII, hydrogen gas formation and the heat-up of the containment atmosphere are mainly responsible for the pressure increase. The important conclusion is that, regardless of whether the core meltdown occurs at high or low reactor pressure, the time to containment failure is clearly longer than previously assumed, for example in the Reactor Safety Study.

11.2.3 PWR case studies

In the event of prolonged *station blackout* in a pressurized water reactor such as Ringhals 2, reactor scram is initiated by the loss of offsite power.

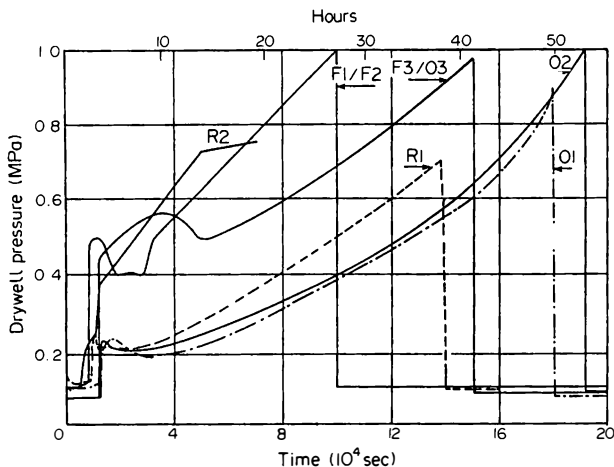


FIG. 11.3. Calculated containment pressure during prolonged station blackout in some Swedish reactors. From *RAMA Containment Group Final Report*, Studsvik, January 1985

Initially, the water on the secondary side acts as a heat sink and the decay heat is removed by steam discharge through the steam line safety valves. After about 1 hour, the water level in the steam generators has boiled down so that heat removal is no longer possible. The primary coolant then starts to heat up and the pressurizer level to rise by thermal expansion of the water. Steam is discharged through the safety relief valves on the pressurizer so that the pressure is kept approximately constant and the decay heat is removed.

After about 2 hours, the shaft seals of the main pumps begin to leak due to loss of seal injection coolant (see 5.2.1). This “*pump-seal LOCA*” results in a drop of the reactor pressure and a decrease in the main coolant inventory. The core begins to uncover after 1.7 hours and to melt after 2.5 hours. The melt and debris accumulate on the core support plate which is assumed to collapse after 3.5 hours when 50% of the core has melted.

The melt falls onto the vessel bottom, which melts through after about 1 minute. The melt is ejected at *high pressure*, causing the water at the bottom of the reactor cavity to boil violently. The water is replaced by water from the accumulator tanks which are actuated during the depressurization. The water flow is predicted to cease after about 9 hours. The water boils off at about 13 hours without the containment failure pressure being reached (see Fig. 11.3). After this time, the debris is reheated and remelted and starts attacking the concrete. The melt could conceivably penetrate the concrete floor in about 3 days. It is also possible that more water may reach the reactor cavity in which case containment overpressure would be caused by the generated steam.

A second example is the case of a *large LOCA with failure of the emergency core cooling*, as studied in a pressurized water reactor of German design, Biblis B (1106). The scenario, which is typical of core meltdown at *low pressure*, was described qualitatively in section 11.1.1. The only essential difference between the Biblis reactor and Swedish pressurized water reactors is the design of the containment building (cf Figs. 5.8 and 5.9).

The loss of both core and containment cooling is assumed to occur as a result of unsuccessful change-over from the injection to the recirculation mode, about 20 minutes after the initial blowdown once the reactor is completely refilled. About half an hour after blowdown, the water level in the reactor vessel reaches the upper edge of the core and the core begins to uncover. One and a half hours later, the reactor vessel fails. About 200 tons of melt with a temperature of 2400°C is then discharged into the reactor cavity. The melt-concrete interaction causes the temperature of the melt to drop and the erosion of the concrete to proceed at a decreasing rate. After about 7 hours the wall of the surrounding annular containment sump is penetrated and the water comes into contact with the melt. The sump water is gradually evaporated causing pressurization of the containment. The design pressure of 0.6 MPa is reached after about 3 days and the failure pressure, 0.9 MPa, after about 5 days (Fig. 11.4).

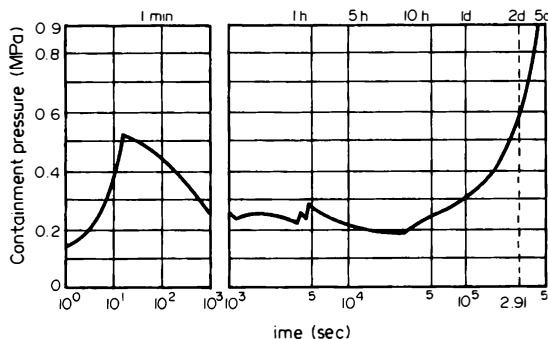


FIG. 11.4. Predicted reactor containment pressure after core meltdown in a German PWR (Biblis B, 1300 MWel). From J P Hosemann, Wechselwirkungen mit der Containmentstruktur und Spaltproduktfreisetzung beim Kernschmelzunfall, *Atomwirtschaft*, Vol 27, No 10, 1982

11.3 Internal Source Terms

During the core melt processes, gases, vapours and airborne particles (aerosols) are formed. A small part of these substances are radioactive fission products, representing the internal source terms. This section examines the mechanisms for the release, transport and removal of the radionuclides, and provides examples of the calculation of internal source terms.

11.3.1 The release of radionuclides

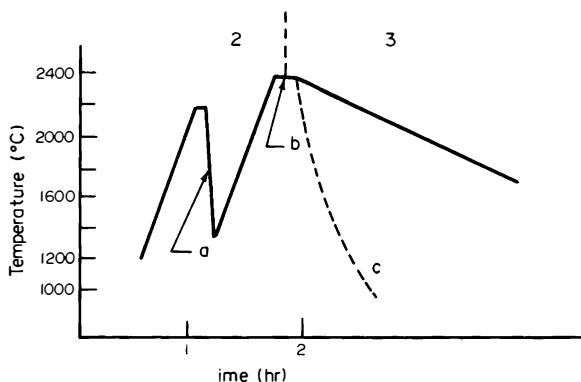
The inventory of radionuclides in the fuel and the mechanisms for their release from a geometrically intact core were discussed in sections 6.2–6.4. It was noted that the release is essentially a function of the fuel temperature. During a core melt accident, the following release processes are characteristic for different phases of the event:

- Gap release* at clad failure during fuel heat-up to 800–900°C. Gaseous and volatile fission products normally contained in the fuel–clad gap are then released. The activity of these nuclides is normally about 0.1% or less of the total activity of the fuel.
- Melt release* at temperatures above about 2000°C, when the fuel begins to melt. All gaseous and volatile products are completely released as well as a part of the less volatile species.
- Vaporization release* during melt–concrete interaction at temperatures of about 2400°C. A further proportion of less volatile fission products is then vaporized and condensed to airborne particles.
- Oxidative release*, associated with the oxidation of fine fuel fragments in the containment, following steam explosions or high-pressure melt ejection.
- Mechanical release* by the flow of steam through the melt, the steam originating from the concrete and partly transformed to hydrogen in the melt. Non-volatile substances can also be carried along and form aerosols.

The meltdown phases and release mechanisms are illustrated in Fig. 11.5. The time scale is representative for the low pressure case of a large LOCA with failure of emergency core cooling, as described in 11.2.3. The meltdown in the reactor vessel is assumed to occur in two stages. Once a part of the core has melted, it collapses and falls to the bottom of the vessel where it is cooled by the remaining water. When the water has evaporated, the temperature increases until the melting point is reached and melt-through of the reactor vessel occurs.

During the first stage, gap release and melt release occur more or less simultaneously. When the core uncover progresses, a strong temperature gradient arises along the fuel rods. Directly above the water surface, where the temperature is still relatively low, gap release occurs to the steam. Higher up in the uncovered region, melt release takes place. The released substances condense and form aerosols. Vapours released in the lower region can condense on particles formed higher up.

During the second stage, at temperatures around 2200–2600°C, liquid phases of molten core material, known as *corium*, are formed. A smoke consisting of metallic oxides, steel, etc., is emitted from the corium. In pressurized water reactors, particles of silver, indium and cadmium from molten control rod material are essential constituents of the smoke. Boiling



- 1 Core heat-up (gap release)
- 2 Meltdown in reactor vessel (melt release)
- 3 Melt-concrete interaction (vaporization release)
- a Core collapses and is cooled by remaining water in the reactor vessel
- b Melt-through of reactor vessel
- c Melt is cooled by water in the containment

FIG. 11.5. Schematic temperature history during core meltdown (low-pressure case). Adapted from K Hassmann, J P Hosemann, Consequences of Degraded Core Accidents, *Nucl. Eng. Des.*, Vol 80, No 2, 1984

water reactor control rods contain steel and boron carbide which are less volatile.

The substances released and the degree of release depends on the inventory of materials in the reactor vessel and on the physical and chemical properties of the individual substances. All constituents which can form aerosols must be considered since the aerosol behaviour is determined by the *total* amount of aerosols. The fission products are only a fraction of the total amount of aerosols, and the radioactive fission products are only a small part of the total amount of fission products.

It is useful to group the fission products according to decreasing volatility (vapour pressure) which determines the degree of release (the characteristic elements in the last two groups are in *italics*):

- Noble gases (Xe, Kr)
- Halogens (I, Br)
- Alkali metals (Cs, Rb)
- Tellurium group (Te, Sb)
- Alkaline earths (Sr, Ba)
- Noble metals (*Ru, Rh, Pd, Mo, Tc*)
- Rare earths (*La, Ce, Pr, Y, Zr, Nb*)

The noble gases are released to 100%. They do not participate in any chemical reactions during their release from the fuel and transport in the primary system and the reactor containment. Iodine was earlier assumed to occur mainly in elementary form and to a small extent as a methyl iodide. It is now considered certain that most of the iodine is released in the form of alkaline iodides, especially cesium iodide, which is less volatile than elemental iodine and forms aerosol. Cesium is mostly found as particles of cesium hydroxide. Other elements also form aerosols.

11.3.2 Removal processes

The substances released during core melting are to a large extent deposited on cooler surfaces within the reactor vessel. Substances not deposited escape into the reactor containment and are transported by steam and gaseous flow and diffusion. The concentration of aerosols in the containment atmosphere decreases by several passive and active removal processes. The removal of noble gases is negligible. Neither is there any removal of the small quantities of methyl iodide formed through the reaction between iodine and organic material in the containment.

When suspended in the containment atmosphere, the aerosol particles increase in size by colliding and sticking together. This process is called *agglomeration*. In humid steam, particles also grow as steam condenses on them. These processes result in a spectrum of particles varying in diameter from less than 0.1 to more than ten-thousandth of a millimetre (micron).

Particles larger than 0.5 micron slowly fall by gravity to the containment floor and settle there. The rate of this *sedimentation* depends on the weight and shape of the particles and on the viscosity of the gas. Sedimentation is the most important deposition mechanism during long residence times in the containment.

The smallest particles—0.1 micron and less—are removed by *diffusion*. These particles are so small and light that they remain suspended for a very long time. When approaching a surface they are caught up in the boundary layer of stagnant gas within about a tenth of a millimetre of the surface.

In addition to the natural deposition processes, the containment spray system contributes significantly to the removal of particles in most accident sequences. In boiling water reactors, iodine and other particles are effectively scrubbed in the condensation pool.

Under certain conditions, deposited particles are returned to the gaseous phase. *Revaporization* means that particles are released when deposited substances are heated by fission product decay heat. Mechanical *resuspension* arises from strong gas streams which dislodge and re-levitate deposited particles. Particles of non-volatile elements might be released during the interaction of core melt with the concrete.

11.3.3 Internal source terms

Understanding of release mechanisms during the meltdown process and the melt-concrete interaction has increased considerably since the Reactor Safety Study. Extensive research programs have been carried out in West Germany and the USA where core melt sequences have been simulated and fission product release and aerosol formation investigated. Melt-concrete interaction has been studied in large-scale experiments. Based on the experiments, calculational models have been developed for determining the gas and vapour release and aerosol formation. These models together with models for aerosol transport and removal are incorporated into computer codes describing the nuclide-specific particle concentration in the containment atmosphere as a function of time.

In order to illustrate the internal source terms quantitatively, some calculational results are presented for the scenario described in 11.2.3, a large LOCA with failure of the emergency core cooling in a German PWR (1107). Initially, the reactor vessel contains (apart from water) uranium dioxide fuel, Zircaloy cladding, Inconel spacers, control rods of a silver-indium-cadmium alloy, and a steel core structure. The fuel is assumed to have an inventory of fission products corresponding to the conditions immediately before refuelling, i.e. when a third of the core has a burn-up of 37 MWd/kg. The total inventory of materials in the reactor vessel is 167 tons, distributed as shown in Table 11.2

The core inventory of fission products is 2.75 tons, about one-tenth of which are radioactive. The release fractions and initial activities during meltdown are given in Table 11.3. As illustrated in Fig. 11.5, the melt release is assumed to occur during two 15-minute periods at 2200° and 2400°C respectively. The released activity is dominated by six elements, namely xenon (Xe), krypton (Kr), iodine (I), cesium (Cs), tellurium (Te) and antimony (Sb).

Iodine and cesium mainly appear as cesium iodide and cesium hydroxide. Altogether 18 kg of iodine is released, of which about 800 g consists of

TABLE 11.2. *Inventory of materials in the reactor vessel of a West German PWR (Biblis B, 1240 MWel)*

Material	Inventory (ton)	Fraction (%)
Uranium dioxide	99.1	59.3
Zircaloy	31.9	19.0
Steel	31.1	18.6
Ag-In-Cd	2.3	1.4
Fission products	2.8	1.7

Source: J P Hosemann, Wechselwirkungen mit der Containmentstruktur und Spaltprodukt-freisetzung beim Kernschmelzunfall, *Atomwirtschaft* Vol 27, No 10, 1982

TABLE 11.3. Core inventory and release fractions of fission products (Biblis B, 1240 MWel)

Element	Core inventory (kg)	Release fraction (%)	Released activity (EBq ^a)
Xe, Kr	439	100	14.7
I	18	100	32.2
Cs	178	100	0.64
Sb	1.2	53	0.86
Te	37	81	5.4
Sr, Ba	179	1.0	0.23
Mo, Te	315	0.2	0.022
Ru, Rh, Pd	307	0.02	0.0023
Y, Zr, Nb	323	0.02	0.0070
La, Ce, Pr	412	0.02	0.0098
Other	600	<0.02	<0.002
Total	2750		46.5

^a1 EBq = 10¹⁸ disintegrations per second.

Source: J P Hosemann, *loc. cit.*

iodine-131, the nuclide which is responsible for a large part of the activity and the radiological hazard in the event of a release to the environment.

The release of gases and aerosols continues after the corium has melted through the reactor vessel and discharged into the containment. In this case, the core melt lies at the bottom of the reactor cavity. The melt consists of an upper oxide layer of UO₂ and ZrO₂ and a lower metallic layer consisting mainly of Fe, Cr, Ni, Zr. Steam which is released from the concrete flows through the melt and is partly transformed into hydrogen. Smoke of vaporized corium is emitted by the melt.

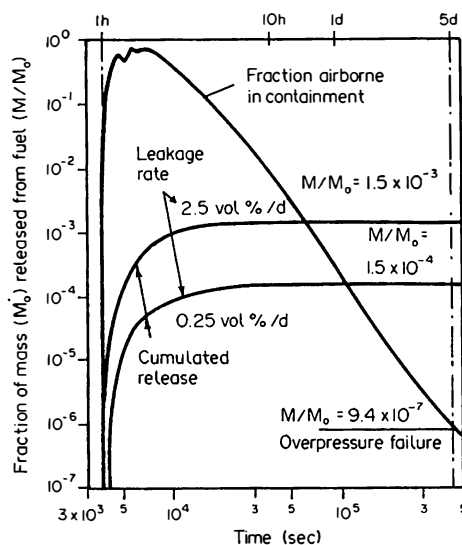
If the aerosols formed after reactor vessel melt-through during 10 minutes of melt-concrete interaction are included, it is estimated that a total of 3.5 tons of particles are released to the containment over a period of an hour after the onset of core melting. As shown in Table 11.4, the majority of the particles consist of control rod material (mostly silver), uranium dioxide and steel. The fission product mass release is about 260 kg, of which about 100 kg is radioactive. The radioactive particles thus constitute about 2.8% of the total aerosol mass.

The foregoing scenario refers to the low pressure case of a core melt accident. For reasons of aerosol physics, the conditions in the high pressure case are quite different. Although the meltdown progresses in a similar way in both cases, the release of aerosol particles to the containment is only about 23 kg in the high-pressure case (1108) as compared to 3.5 tons in the reference low-pressure case.

The time variation of the aerosol mass in the containment atmosphere is illustrated in Fig. 11.6, expressed as the airborne fraction of the released

TABLE 11.4. Distribution of aerosol mass released to the containment (Biblis B, 1240 MWel)

Element	Mass in reactor vessel (kg)	Release fraction (%)	Aerosol mass (kg)
UO ₂	99,100	0.5	490
Fe	20,800	2.3	470
Cr	5700	1.8	107
Ni	4000	2.1	84
Co	60	2.1	1
Mn	450	18	81
Zr	31,500	0.2	80 (ZrO ₂)
Sn	350	20	70
Ag	1850	75	1390
In	350	20	70
Cd	115	100	115
Silicates			300
I	18	100	18
Cs	178	100	178
Te	37	81	30
Other	2300	<1	10
Total	166,658		3494

Source: J P Hosemann, *loc. cit.*FIG. 11.6. Predicted aerosol mass versus residence time in a PWR containment (Biblis B, 1300 MWel). Adapted from J P Hosemann, *loc. cit.*

mass M_o . The particle growth by agglomeration and steam condensation as well as the deposition by sedimentation and diffusion are considered in the calculational model.

As indicated in the diagram, the aerosol mass decreases by a factor of 10^6 in 5 days, until overpressure failure is predicted to occur in the reference case (Fig. 11.4). If the core inventory of fission products and the release fractions are known, the mass of fission product particles in the containment at any time after the initiating event can be estimated from the curve. For example, since cesium is released to 100%, the curve immediately gives the mass of cesium in the containment atmosphere, with $M_o = 178$ kg according to Table 11.3.

Iodine is a special case due to the many different chemical forms it can take. In the reference case it is estimated that the iodine released from the primary system to the containment consists of particulate cesium iodide (CsI) to 99% and of gaseous molecular iodine (I_2) to 1% (1106). The molecular iodine is transported to the containment sump water within a few hours. An equilibrium is reached between the I_2 dissolved in the water and the I_2 in the containment atmosphere. A small part of the iodine is transformed into methyl iodide by surface reaction between I_2 and organic matter. The cesium iodide is partly deposited on surfaces in the containment and partly dissolved in the sump water, where it dissociates into ions (Cs^+ and I^-). When the containment fails after 5 days, some of the ions are released in water drops as the sump water gradually evaporates. The release of cesium and iodine can therefore occur over a period of several days, until all of the sump water has evaporated.

Even if the containment integrity is maintained, it cannot be assumed that the containment is absolutely leaktight. A *diffuse leakage* occurs which can amount to several tenths of a percent by volume per day. The cumulative leakage of particles assuming a leakage rate of 0.25% per day is illustrated in Fig. 11.6. In this case, the leakage reaches its maximum value, $M/M_o = 1.5 \times 10^{-4}$, after about 6 hours.

German pressurized water reactors have a double containment where the inner steel sphere is surrounded by an outer concrete building (Fig. 5.9). The annulus is ventilated via filters to the stack. Swedish boiling water reactors also have a filtered ventilation to the stack of any leakage from the containment into the reactor building. This is estimated to reduce the concentration of iodine and particles reaching the environment by two orders of magnitude.

11.4 Containment Analysis

The prime purpose of the reactor containment is to remove and retain radioactive substances. Severe accidents can lead to pressure build-up in the containment and threaten its integrity. This section begins by presenting

some data on containment strength. Some typical pressure transients are then described. Finally, the principles for probabilistic containment analysis are outlined.

11.4.1 Containment strength

A typical dry containment for a Swedish pressurized water reactor (Ringhals 2, 800 MWe) has a volume of about 50,000 m³ and is filled with air. The design principle is such that energy transferred to the containment during a rupture of the primary system is stored in the large volume of the containment, mainly as steam. A pressure suppression containment for a boiling water reactor is based on the principle that the released energy is stored in the water of the condensation pool. Its volume can therefore be made significantly smaller, e.g. 10,200 m³ for Ringhals 1 (750 MWe), of which the free gas volume, 7600 m³, is filled with nitrogen during operation.

Reactor containments are designed to withstand the pressure resulting from a loss of coolant accident initiated by a double-ended break in a main coolant pipeline. According to generally accepted design standards for pressurized components, the containment is able to withstand a higher pressure than the design pressure before it begins to leak or fail. The design pressure for Swedish reactor containments is 0.5–0.6 MPa (see Table 11.5). The failure pressure is estimated at 1.5–2 times the design pressure.

TABLE 11.5. *Data for Swedish reactor containments*

Unit	Containment	Pressure (MPa)		Failure mode
		Design	Failure	
Ringhals 1	Pressure suppression	0.5	0.75	Opening of roof-cylinder joint
Forsmark 1/2	Pressure suppression	0.55	0.92	Longitudinal cracking
Forsmark 3	Pressure suppression	0.6	1.25	Cracking in roof and pool region
Ringhals 2	Dry	0.5	1.2–1.3	Longitudinal cracking
Ringhals 3/4	Dry	0.5	>0.69	Cracking in base plate

Source: *MITRA Final Report*, Swedish State Power Board, 1985

The type and location of a containment breach and the time to failure are very important for the environmental consequences. Certain containments, e.g. those of Forsmark 1/2 and Ringhals 2, are predicted to fail along a generatrice, while in other cases cracking will occur either in the top region high above the ground (Ringhals 1), or in the base plate (Ringhals 3/4). The probability of leakage in flanges, electrical penetrations, etc., prior to failure can be significant, especially during sequences with high temperatures in the containment atmosphere.

11.4.2 Overpressure failure

Core meltdown can, as shown in 11.2.2, cause pressure build-up in the reactor containment which may result in containment failure. High pressure can also occur without core melting, if the containment cooling is inadequate. The containment pressure is the sum of the partial pressure of steam and non-condensable gases, including the original air (PWR) or nitrogen (BWR) and the hydrogen generated during a core melt accident. Hydrogen burn and steam spikes, when the core melt falls into water, will contribute to the pressure rise and gas heat-up. There may also be direct heating when, in the case of melt ejection at high pressure, fine droplets of molten material are sprayed throughout the cavity (PWR) or lower drywell (BWR).

The rate of pressure build-up depends on the particular accident sequence. With inadequate containment cooling, the pressure will increase slowly, by evaporation of the water in the condensation pool (BWR) or the containment sump (PWR). In order for the pressure to reach the failure pressure, the loss of containment cooling must subsist for at least 24 hours. During this time, there are good possibilities of restoring the cooling and avoiding overpressure.

For transients with loss of emergency cooling, such as during station blackout, containment overpressure occurs *after* core meltdown (see 11.2.2). When the reactor vessel is penetrated by the melt, a pressure peak is obtained which can cause containment failure within a few hours after the initiating event.

During certain core melt sequences, overpressure occurs *before* core meltdown. An example is the case of a transient with loss of the main heat sink (the turbine condenser) without scram in a BWR. The reactor power is then automatically adjusted to a lower level to match the reduced feedwater flow. Steam is discharged to the condensation pool where the water is rapidly heated to boiling, even if the pool cooling systems are operating. The containment failure pressure is reached within an hour. When the feedwater supply ceases, the water level in the reactor vessel falls and core melting begins. The sequence has a low probability of occurrence since there are alternative ways of shutting down the reactor if the hydraulic scram fails.

Another sequence leading to rapid pressurization of the containment in a BWR is the case of a large LOCA with loss of core cooling and ineffective containment pressure suppression. Inadequate pressure suppression results if there is leakage between the containment drywell and wetwell. The steam escaping via the leak does not then condense but contributes directly to the pressure build-up. In the case of a large leak, the containment failure pressure is reached within a few minutes.

Table 11.6 is a summary of the consequences with regard to core meltdown and containment pressure for various combinations of initiating events and safety function failures. In the sequences marked “high pressure

TABLE 11.6. Core melt and containment pressure for various accident sequences in a boiling water reactor

Event	Reactor shutdown	Core cooling	Pressure suppression	Containment cooling	Consequence
Transient LOCA		+			HP without CM(1) HP without CM
Transient LOCA		+			First HP, then CM(1) First HP, then CM
Transient LOCA	+		+		First CM, then HP First CM, then HP
Transient LOCA	+	-	-	+	Rapid HP, then CM Rapid HP, then CM

+ successful function.

failed function.

HP high containment pressure.

CM core melt.

(1) Core meltdown occurs after 15–40 hours once the core spray pumps have cavitated, if there is no alternative make-up water.

without core melting”, it should be pointed out that overpressure in itself can lead to core melting since safety systems may be damaged during the pressure peak which occurs at the moment of containment rupture.

Early overpressure failure could also occur in a PWR containment, for example if the containment fans and sprays failed to operate. In this case the steam produced by the decay heat of the core would not condense and the steam pressure, perhaps augmented by a hydrogen burn or a steam spike, could cause the containment to fail after a few hours. Another example is the case where the reactor pressure vessel is penetrated at high pressure and molten core material is ejected into the cavity. If the molten material is aerosolized and dispersed throughout the containment, a rapid pressure rise could result due to *direct heating* of the containment atmosphere.

11.4.3 Plant damage states

The initial conditions for containment analysis are determined by the core damage sequences discussed in Chapter 10. It is not practical to analyse the containment response for all possible sequences in detail. Attempts are therefore made to assemble sequences with similar initial conditions for containment loadings in representative groups, called plant damage states. The plant damage states form the interface between core damage analysis and containment analysis.

The classification into LOCA and transients used in the Reactor Safety

Study is too broad for characterizing the plant damage states. A more detailed grouping is based on both the type of initiating event and the kind of failed safety function. The principle is illustrated for boiling water reactors in Table 11.6. In this case, the damage states are characterized by whether there is a rapid or slow pressure build-up in the containment and whether core melt occurs before or after overpressure.

A similar classification used for pressurized water reactors is shown in Table 11.7. In this case, the definition of plant damage states is based on four considerations: the type of accident (LOCA or transient), the time at which the core degrades during the accident (early or late), whether the containment cooling (sprays and fans) is available or not, and whether reactor isolation is successful or fails.

TABLE 11.7. *Plant damage states for a pressurized water reactor*

Class	State	Representative core damage sequence
	Core meltdown before containment failure. High pressure in primary system at vessel penetration. Containment cooling available.	Small LOCA without core cooling
2	Like Class 1 but without containment cooling	Small LOCA without core cooling
3	Like Class 2	Station blackout transient
4	Core meltdown before containment failure. Low pressure in primary system at vessel penetration. Containment cooling available.	Large and medium LOCA without core cooling
5	Like Class 4 but without containment cooling	Large and medium LOCA without core cooling
6	Like Class 4	Transient without core cooling in shutdown reactor with open primary system
	Containment failure before core meltdown	LOCA without core cooling with functional low-head recirculation
8	Containment bypass	Steam generator tube rupture

Source: *MITRA Final Report*, Swedish State Power Board, 1985

11.4.4 Containment event trees

Apart from overpressure, the containment can be incapacitated by:

- inadequate isolation* of inlets and penetrations such as unclosed isolation valves, leaky airlocks and cable penetrations. A certain amount of diffuse leakage is unavoidable;
- bypass*, e.g. during an unisolated external break in a pipe connected to the primary system. This includes V-LOCA and steam generator tube rupture;
- melt-through* of the concrete basemat. If this can occur, which is debatable, it will only happen several days after the initiating event.

The containment response can be illustrated by an event tree where the initiating event consists of a plant damage state and the branches represent containment success or failure modes (Fig. 11.7). The paths through the event tree lead to successful containment of the accident or to containment failure of various types. The probability of each of the end states is evaluated for each of the plant damage states.

Each combination of plant damage state and containment failure mode defines a *release sequence*. The failure mode probability is conditional on the particular plant damage state. The product of the core damage frequency and the particular failure mode probability determines the *release frequency*.

The conditional failure mode probability is generally less than 1, which means that the release frequency is lower than the core damage frequency. In most cases core melt will *not* lead to a large release since the radionuclides will be contained. In core damage sequences involving containment overpressure or bypass, the failure mode probability is equal to 1 by definition. However, in these cases the core damage frequency is usually low, which results in a low release frequency.

It is evident that a high core damage frequency is not synonymous with a high release frequency. Also, no core damage sequence can be disregarded on account of low frequency alone. These facts are illustrated in Table 11.8 where calculated core damage and release frequencies are compared for a U.S. pressurized water reactor. The table shows that the relative ranks of the core damage and release sequences is quite different. In this particular case, a small LOCA with failed change-over to recirculation dominates the core damage frequency, while an earthquake resulting in station blackout is the dominant contributor to the overall release frequency.

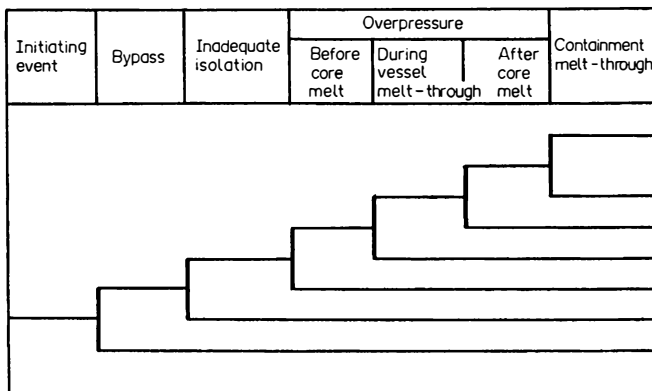


FIG. 11.7. Simplified containment event tree. From *MITRA Final Report*, Swedish State Power Board, 1985

TABLE 11.8. Comparison of core damage frequencies and release frequencies for Zion-1

Rank of core damage frequency	Sequence	Core damage frequency (per year)	Containment failure probability	Release frequency (per year)	Rank of release frequency
1	Small LOCA	1.62E-5 ^a	1E-4	1.62E-10	4
2	ATWS	6.65E-6	1E-4	6.65E-10	5
3	Earthquake	5.60E-6	1.0	5.60E-6	1
4	Large LOCA	6.21E-6	1E-4	6.21E-10	6
5	Medium LOCA	4.33E-6	1E-6	4.33E-10	7
6	Inadvertent safety injection	2.07E-6	1E-4	2.07E-10	8
7	Loss-of-auxiliary power	7.28E-7	2E-4	1.46E-10	9
8	Station blackout	2.00E-7	1.0	2.00E-7	2
9	V-LOCA	1.05E-7	1.0	1.05E-7	3

^aE-5 = 10⁻⁵Source: *Zion Probabilistic Safety Study*, Commonwealth Edison Company of Chicago, September 1981

11.5 External Source Terms

Each release sequence is characterized by the estimated frequency as well as by the time delay (after the initiating event), duration, magnitude and composition of the release, collectively known as the *external source terms*. The external source terms are determined by the behaviour of the radionuclides in the reactor vessel and containment, more specifically by the concentration of radioactive gases and particles in the containment atmosphere at the time of containment failure.

11.5.1 Release categories

As in the thermohydraulic analysis of core damage sequences, it is impractical to determine the source terms for all possible release sequences. Therefore, the release sequences are classified in groups with similar characteristics, known as *release categories*. The classification is based on the assumption that representative source terms can be defined for each group. The representative source terms must neither underestimate nor significantly overestimate the magnitude of the release for any sequence in the group.

Each release sequence is assigned to a specific plant damage state and a release category. This is illustrated in Fig. 11.8. The frequency for each combination of plant damage state and release category can be determined by summing the contribution from individual release sequences. Finally, the frequency for each release category is obtained by totalling the contribution from all plant damage states.

Release category Plant damage state	Category 1	Category 2	...	...	...
Class 1	Individual release frequencies	Individual release frequencies			
Class 2	Individual release frequencies				
...					
...					
...					
	Total release frequency for category 1	Total release frequency for category 2			

FIG. 11.8. Scheme for determining the release frequency per release category

11.5.2 The Reactor Safety Study

The concept of release categories was introduced in the Reactor Safety Study (1109). The classification was based on the extent of core damage (complete core melt or clad failure only), the containment failure mode (overpressure, inadequate isolation or basemat melt-through) and on the performance of the radioactivity removal systems in the containment (availability or failure of sprays and fans). Nine release categories were defined for pressurized water reactors (Table 11.9), and five release categories for boiling water reactors (Table 11.10).

PWR 1 and BWR 1 comprise core melt sequences with *steam explosion* resulting in reactor vessel and containment failure. Such events lead to large

TABLE 11.9. Release categories for pressurized water reactors according to the Reactor Safety Study (1109)

Category	Description
PWR 1	Steam explosion in reactor vessel. Large containment breach
PWR 2	Containment overpressure failure by hydrogen burn and steam spike
PWR 3	Containment overpressure failure with partly operating activity removal systems
PWR 4	Insufficient reactor isolation. Failure of removal systems to operate
PWR 5	Insufficient reactor isolation. Operating removal systems
PWR 6	Containment melt-through. Failure of removal systems to operate
PWR 7	Containment melt-through. Operating removal systems
PWR 8	Gap release with insufficient reactor isolation
PWR 9	Gap release with proper reactor isolation

TABLE 11.10. Release categories for boiling water reactors according to the Reactor Safety Study (1109)

Category	Description
BWR 1	Steam explosion in reactor vessel. Containment breach by missile action
BWR 2	Containment overpressure failure. Release directly to the atmosphere
BWR 3	Containment overpressure failure. Release through the reactor building
BWR 4	Insufficient reactor isolation. Activity removal in the containment and the reactor building
BWR 5	Gap release with operating activity removal systems

releases for two reasons. Firstly, the release occurs directly in connection with the meltdown resulting in a minimal removal of radionuclides in the reactor vessel and containment. Secondly, more of the core inventory of radionuclides is released during a steam explosion than during other core melt sequences.

PWR 2 and 3, as well as BWR 2 and 3, involve core melt sequences with containment *overpressure failure* without and with effective activity removal systems. Failure is assumed to occur after 2.5–5 hours in the pressurized water reactor and after 30 minutes in the boiling water reactor. Before failure, the conservative assumption is made that the diffuse leakage is ten times greater than specified for the containment design.

PWR 4 and 5 as well as BWR 4 represent core melt sequences with *inadequate isolation*. Leakage is assumed to occur relatively slowly allowing the natural removal mechanisms in the containment time to act. In addition, in PWR 5 it is assumed that the radioactivity removal systems are operable.

PWR 6 and 7 cover core melt sequences with *melt-through* of the containment concrete basemat. On present evidence, this failure mode is considered to be unrealistic, or in any case to take a much longer time than the 10–12 hours assumed in the Reactor Safety Study. Melt-through is believed to be irrelevant for boiling water reactors since the containment would fail first due to overpressure.

PWR 8 and 9 as well as BWR 5 include sequences where the reactor safety systems are sufficiently effective so that the core does not melt, but where the fuel cladding is damaged. Category 8 thus involves gap release with failure of the containment to isolate properly. In category 9, the containment isolates correctly.

Each release category is characterized by the magnitude and composition of the release. This is expressed as the part of the core inventory of the seven groups of fission products released. These groups are defined in 11.3.1, i.e. the noble gases, I-Br, Cs-Rb, Te-Sb, Ba-Sr and the Ru and La groups.

Finally, each release sequence is assigned to a release category. The total frequency for each release category is estimated by totalling all the release

frequencies in each release category as indicated in Fig. 11.8. The result is summarized in Table 11.11, which also presents the time delay and duration of the release as well as its composition in fractions of the core inventory.

11.5.3 German source term studies

The source terms of the Reactor Safety Study were also used in Phase A of the German safety study (10.3.2) with certain modifications. A new release category for the case of a large containment leak, corresponding to failure of isolating the containment ventilation system, was introduced, and the PWR 6 and PWR 7 categories were eliminated. Eight release categories were thus obtained (see Table 12.10). The release categories FK2, core melt with a large containment leak (diameter 300 mm), and FK6, core melt with overpressure failure are of special interest.

Phase B of the German safety study included a more detailed study of the core melt sequences. Extensive experiments were carried out and improved calculational methods developed. In some cases, the source terms were found to be lower than previously assumed. A comparison between original and updated calculations for FK2 and FK6, shown in Table 11.12, illustrates this point (1110). Two scenarios were examined for FK6: a large breach resulting in rapid depressurization of the containment, destroying the filter system (see Fig. 5.9), and a small breach with slow depressurization and an operable filter system.

For FK2, the Phase B calculations show a factor of about 15 lower concentration of iodine and cesium in the release. This is due to the fact that the natural removal mechanisms are more efficient than previously believed. For FK6, which is the most probable release category (99.6%), the Phase B results show a decrease in the iodine release by two orders of magnitude and in cesium by three orders of magnitude. In extreme cases, the releases are five orders of magnitude less than previously estimated. This is mainly due to the longer time to failure by overpressure, 5 days as compared to 27 hours in the Phase A study, which allows the removal mechanisms more time to act. Early containment failure due to steam explosion is considered impossible in the Phase B study.

The previous results refer to the low-pressure meltdown scenario. In the high-pressure case, the release fractions, except for the noble gases, are an order of magnitude less in the case of isolation failure (FK2), whilst the opposite is true for overpressure failure (FK6). This is due to the fact that the release to the containment is lower and the time to overpressure failure shorter in the high-pressure case (1108).

TABLE 11.11.11 The Reactor Safety Study release categories

Release category	Probability per reactor-yr	Time of Release (hr)	Duration of Release (hr)	Warning Time for Evacuation (hr)	Elevation of Release (metres)	Fraction of Core Inventory Release								
						Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru ^a	La ^b	
PWR 1	9E-7	2.5	0.5	1.0	25	0.9	6E-3	0.7	0.4	0.4	0.05	0.4	3E-3	
PWR 2	8E-6	2.5	0.5	1.0	0	0.9	7E-3	0.7	0.5	0.3	0.06	0.02	4E-3	
PWR 3	4E-6	5.0	1.5	2.0	0	0.8	6E-3	0.2	0.2	0.3	0.02	0.03	3E-3	
PWR 4	5E-7	2.0	3.0	2.0	0	0.6	2E-3	0.09	0.04	0.03	5E-3	3E-3	4E-4	
PWR 5	7E-7	2.0	4.0	1.0	0	0.3	2E-3	0.03	9E-3	5E-3	1E-3	6E-4	7E-5	
PWR 6	6E-6	12.0	10.0	1.0	0	0.3	2E-3	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5	
PWR 7	4E-5	10.0	10.0	1.0	0	6E-3	2E-5	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7	
PWR 8	4E-5	0.5	0.5	N/A	0	2E-3	5E-6	1E-4	5E-4	1E-6	1E-8	0	0	
PWR 9	4E-4	0.5	0.5	N/A	0	3E-6	7E-9	1E-7	6E-7	1E-9	1E-11	0	0	
BWR 1	1E-6	2.0	2.0	1.5	25	1.0	7E-3	0.40	0.40	0.70	0.05	0.05	5E-3	
BWR 2	6E-6	30.0	3.0	2.0	0	1.0	7E-3	0.90	0.50	0.30	0.10	0.03	4E-3	
BWR 3	2E-5	30.0	3.0	2.0	25	1.0	7E-3	0.10	0.10	0.30	0.01	0.02	3E-3	
BWR 4	2E-6	5.0	2.0	2.0	25	0.6	7E-4	8E-4	5E-3	4E-3	6E-4	6E-4	1E-4	
BWR 5	1E-4	3.5	5.0	N/A	150	5E-4	2E-9	6E-11	4E-9	8E-12	8E-14	0	0	

^a Includes Mo, Rh, Tc, Co.^b Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

TABLE 11.12. *Comparison between original (Phase A) and updated (Phase B) estimates of environmental releases during core melt accidents (low pressure case) in a German pressurized water reactor (Biblis B, 1 240 MWel).*

Core		Release (per cent of core inventory)	
		Iodine	Cesium
FK 2. Large containment leak (300 mm equivalent diameter)	Phase A	39	26
	Phase B	0.64	0.69
FK 6. Overpressure failure (a) 300 cm ² break area, without filter (b) 20 cm ² break area, with filter	Phase A	1.0	8E-2
	Phase B	1E-2	1E-4
	Phase B	5E-5	6E-7

Source: K Hassmann, J P Hosemann, Consequences of Degraded Core Accidents, *Nucl. Eng. Des.* Vol 80, No 2, 1984

11.5.4 U.S. studies

Re-evaluation studies in the USA confirm that the external source terms were partly overestimated in the Reactor Safety Study, mainly for three reasons.

Firstly, it was earlier assumed that iodine and cesium existed in elemental form in the containment atmosphere and were released as gases. In fact, these very chemically active elements react to form particulate cesium iodide and cesium hydroxide, which are removed to more than 90% in the reactor and containment systems.

Secondly, it was assumed that the containment fails once the design pressure is reached. In practice, the failure pressure is 1.5–2 times higher (cf Table 11.5). The time to failure is therefore longer, and the aerosol removal mechanisms, which are actually more efficient than previously believed, have a longer time to act. This results in a lower aerosol content and reduced release.

Thirdly, it was assumed in the Reactor Safety Study that all core melt sequences lead to containment failure. As shown by the more detailed containment analyses which have since become possible, many sequences result in the formation of coolable debris in the reactor vessel or containment, without failure of the containment. In these cases the offsite releases will be determined by diffuse leakage, resulting in much smaller release fractions than assumed in the release categories of the Reactor Safety Study.

Early containment failure, i.e. within a few hours after the initiating event, can result in large releases. While the IDCOR study (1104) concluded that early overpressure failure due to steam explosion or hydrogen detonation is not realistic, the Reactor Risk Reference Study (1111) did not rule out these possibilities in certain severe accident sequences. Moreover, direct containment heating (cf 11.4.2) and direct molten core debris attack on the

containment were considered to represent potential mechanisms for early containment failure.

Table 11.13 summarizes some representative release sequences for the Peach Bottom-2 BWR, analysed within the IDCOR project. Comparison with the Reactor Safety Study results in Table 11.11 shows some reduction of the source terms for both early and late containment failure. The table also serves to illustrate the effects of operator action in reducing the release in the case of an anticipated transient without scram (ATWS). In this case the condensation pool heats up quickly, leading to rapid pressure rise in the containment and to loss of emergency core cooling. If the operator manages to vent the containment, overpressure is avoided, and if in addition he succeeds in providing for coolant make-up, the source terms are strongly reduced.

Figure 11.9 displays some typical source term information from the Reactor Risk Reference Study. The diagram compares results for station blackout scenarios at the Surry-1 PWR. The scenarios are characterized by high-pressure melt ejection from the reactor vessel, inoperable containment sprays, and early containment failure due to steam spike and hydrogen deflagration. The uncertainty bands are determined by statistical sampling of variations in the assumptions that affect the course of the accident and in the mechanisms that affect the release of the radionuclides.

The general conclusion to be drawn from Fig. 11.9 and other results of the Reactor Risk Reference Study is that the uncertainty in the external source terms is large. The magnitude of the upper portion of the uncertainty band is not significantly different from earlier estimates in the Reactor Safety Study, but the lower portion of the band is well below earlier estimates.

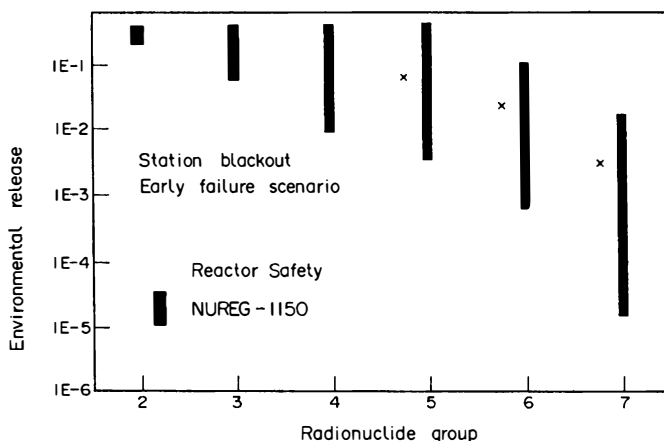


FIG. 11.9. Comparison of results for station blackout scenarios at the Surry plant. From *Reactor Risk Reference Document*, USNRC Report NUREG-1150, Draft, February 1987

TABLE 11.13 Representative release sequences for Peach Bottom-2 (BWR) according to the IDCOR study (1104)

Sequence	Frequency (PMY)	Containment failure (hr)	Release (percent of core inventory)					
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Sr-Ba	Ru-Mo
Transient with loss of containment cooling								
ATWS ^a Case 1 Case 2 Case 3 Case 4	0.2	32	100	20	20	10	4E-2	6E-2
	0.3	1.4	100	10	10	10	4E-2	1E-1
	3	—	100	3	3	6	1E-2	2E-2
	4	1.4	100	3	3	4E-1	8E-3	3E-2
	30		100	6E-2	6E-2	4E-2	4E-2	1E-3
Transient with loss of power	0.5	18	100	5	5	6	8E-3	1E-2

^a ATWS = anticipated transient without scram.

Case 1: No operator actions taken.

Case 2: Operator vents through wetwell when drywell pressure reaches 0.8 MPa.

Case 3: Operator refills condensate storage tank to provide continuous coolant make-up flow.

Case 4: Operator both vents through wetwell and refills condensate storage tank.

11.5.5 Swedish studies

Source term studies were carried out in the MITRA project (1112), largely with the same methods as in the IDCOR study. The results for the Ringhals 1 boiling water reactor in Table 11.14 and for the Ringhals 2 pressurized water reactor in Table 11.15 show the frequencies and releases of the radiologically important nuclides, iodine and cesium, for some typical release sequences.

The MITRA results are presented graphically and compared with

TABLE 11.14 *Representative release sequences for Ringhals 1 (BWR)*

Core melt sequence	Containment failure mode	Release frequency (PMY)	Release of I and Cs (% of core inventory)
Transient due to station blackout	Overpressure at reactor vessel penetration	0.02	30
Transient without containment cooling	Overpressure before core meltdown	0.1	20
Reactor vessel rupture	Overpressure before core meltdown	0.27	15
Transient without reactor scram	Overpressure before core meltdown	0.02	5
External pipebreak in shutdown cooling system	Bypass	0.12	3
Large LOCA with incomplete steam condensation	Overpressure before core meltdown	1.6	

Source: *MITRA Final Report*, Swedish State Power Board, 1985

TABLE 11.15 *Representative release sequences for Ringhals 2 (PWR)*

Core damage sequence	Containment failure mode	Release frequency (PMY)	Release of I and Cs (% of core inventory)
Interfacing systems LOCA (V-LOCA)	Bypass	0.04	31
Transient without containment cooling	Overpressure before core meltdown	0.02	25
Loss of cooling during shutdown	Insufficient isolation	0.09	24
LOCA with loss of core and containment cooling	Insufficient isolation	0.04	
Tube rupture in steam generator with faulty pressure relief valves	Bypass	0.05	6
Ditto with operational pressure relief valves	Bypass	1.0	0.3

Source: *MITRA Final Report*, Swedish State Power Board, 1985

results from the Reactor Safety Study in Figs. 11.10 and 11.11. The diagrams show the exceedance frequency versus the magnitude of the release. The exceedance frequency expresses the probable number of cases per year where the estimated release is greater than or equal to a particular value.

The area below the curves is a measure of the expectation value or mean value of the release. The results indicate that the expectation value for iodine release is about 30 times less for Ringhals 1 according to the MITRA study, as compared to that for Peach Bottom-2 according to the Reactor Safety Study, and more than 100 times less for Ringhals 2 than for Surry-1.

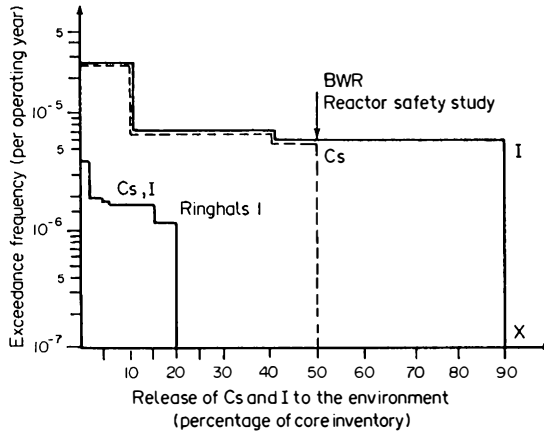


FIG. 11.10. Frequency diagram for the release of iodine and cesium in the event of a core melt accident in Ringhals 1. From *MITRA Final Report*, Swedish State Power Board, 1985

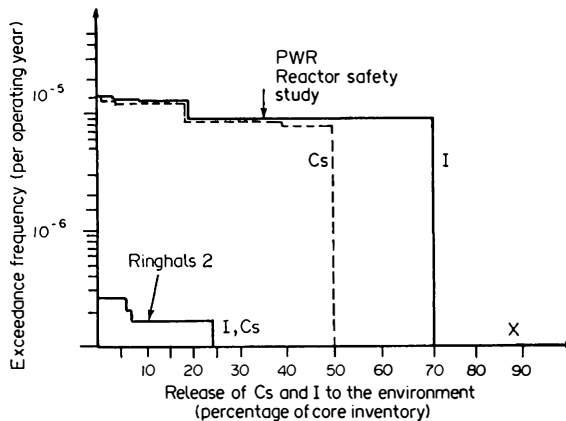


FIG. 11.11. Frequency diagram for the release of iodine and cesium in the event of a core melt accident in Ringhals 2. From *MITRA Final Report*, Swedish State Power Board, 1985

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12

Consequence Analysis

Consequence analysis is the study of the radiological effects of environmental releases from nuclear power plants. The radionuclides are released as gases or airborne particles or in effluent water. Controlled releases and radiation protection during normal operation are discussed in Chapter 6. This chapter describes the effects of uncontrolled releases during accident conditions. The methodology is treated first, followed by some examples of deterministic analysis. The principles of probabilistic risk analysis are then reviewed and some results presented. Finally, the evaluation and comparison of risks are discussed.

12.1 Methodology

Consequence analysis is carried out in stages. The analysis starts with the external source terms described in the previous chapter. Firstly, the dispersion of the radioactive substances in the atmosphere is studied. Their concentration at ground level as a function of time and distance from the release point is calculated. The activity of a radionuclide is proportional to its concentration. The radiation dose is then estimated from the activity, taking into account the various exposure pathways and the effects of emergency action. Finally, the health effects are assessed on the basis of assumed dose–effect and dose–response relationships.

12.1.1 Atmospheric dispersion

An accidental release into the environment is usually composed of steam, gas and airborne particles, some of which are radioactive. A continuous release spreads out like a plume in the wind—in the same way as smoke from a chimney (Fig. 12.1). The elevation above ground and the temperature (energy content) of the release is of great importance. The release is said to be at *ground level* up to about 20 m height, and *elevated* if the release point is about 100 m above ground. The subsequent transport and diffusion of the plume is determined by the meteorological conditions. Figure 12.1 illustrates the effect of atmospheric stability.

In the simplest case, the plume diffusion is characterized by the mean

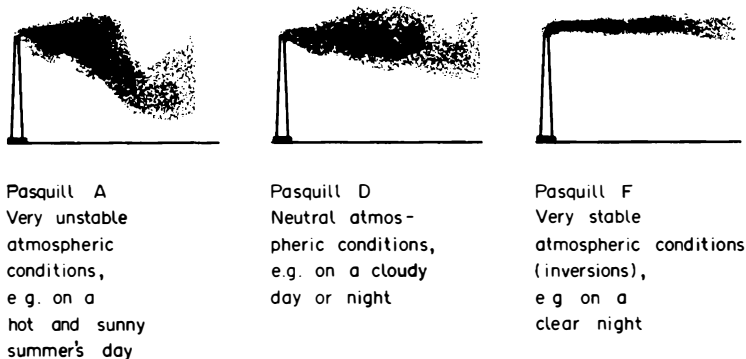


FIG. 12.1. Schematic patterns of plume dispersion for various conditions of atmospheric stability

horizontal wind speed and by vertical and lateral dispersion parameters which express the atmospheric turbulence. This model results in a normal (Gaussian) distribution of the airborne radionuclide concentration (Fig. 12.2). The extension of the plume is determined by the wind speed and the duration of the release while the vertical and lateral spread depend on the dispersion parameters. When the plume passes a certain point, the activity will first rise and then fall during a time equal to the duration of the release.

However, the mean wind speed is not constant vertically; neither does the wind direction remain constant over some period of time. Like turbulence, the mean wind speed is influenced by the "roughness" of the ground surface, and by the vertical temperature gradient which determines the atmospheric stability. An inversion layer can prevent vertical dispersion completely.

A dispersion model which is often used classifies meteorological conditions into six stability categories and is known as the *Pasquill scheme* (1201). Categories A to C refer to unstable, D to neutral and E to F to stable atmospheric conditions (Table 12.1). Each category is characterized

TABLE 12.1. Conditions for which Pasquill stability categories are appropriate

Surface wind speed m s ⁻¹	Daytime insolation			Night-time conditions Cloudiness	
	Strong	Moderate	Slight	≥ 4/8	≤ 3/8
< 2	A	A-B	B	—	—
2	A-B	B	C	E	F
4	B	B-C	C	D	E
6	C	C-D	D	D	D
> 6	C	D	D	D	D

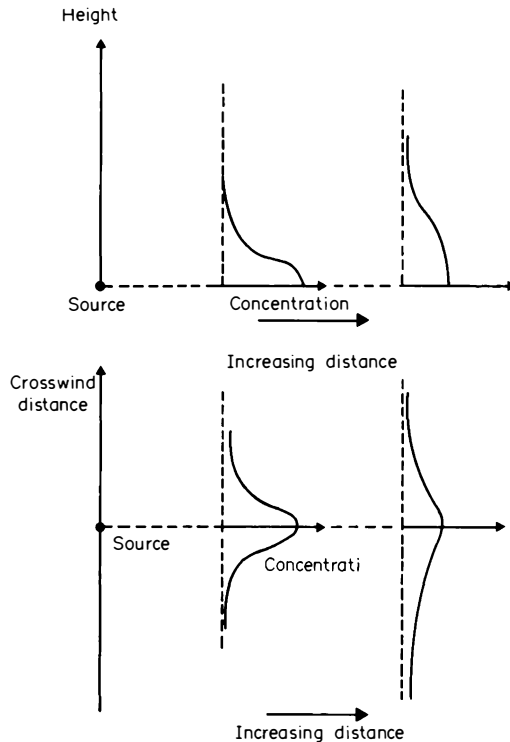


FIG. 12.2. Vertical and lateral concentration profiles at two downwind positions for a ground-level point source. From W Nixon et al, Accident Consequence Analysis, *Nucl. Energy*, Vol 24, No 4, 1985

by the magnitude of the increase of the dispersion parameters with distance from the release point (1202).

Figure 12.3 shows how the activity concentration at ground level varies with the downwind distance from the release point in stability category D.

It can be seen that an elevated release through a stack considerably reduces the activity concentration close to the source. During unstable conditions the maximum concentration is higher and occurs nearer to the release point, whereas during stable conditions the maximum concentration is lower and displaced towards a greater distance. The concentration is proportional to the source strength (Bq s^{-1}) and inversely proportional to the wind speed (m s^{-1}).

In the Pasquill scheme, wind speed, wind direction and meteorological conditions are assumed to remain unchanged during the dispersion process. Although this assumption is unrealistic, it can be partly compensated for by varying the initial conditions in the calculation. However, no practically applicable and general calculational model is as yet available which takes

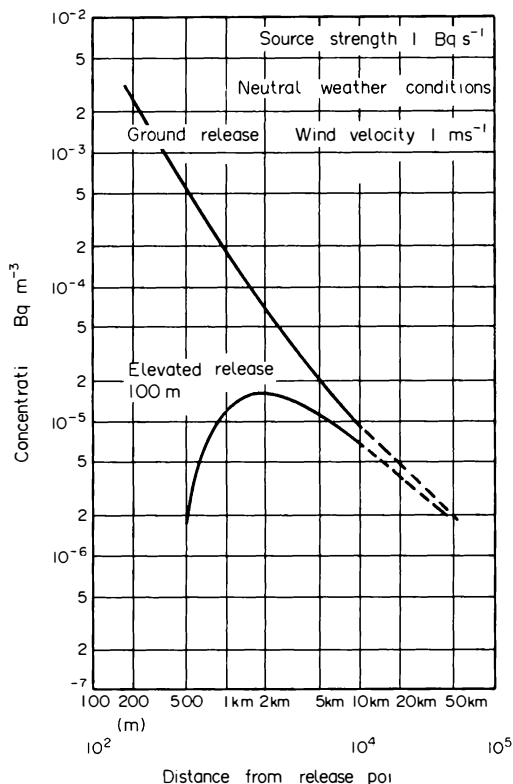


FIG. 12.3. Activity concentration versus downwind distance from the release point at ground level in the centreline plane of the plume

into account changing meteorological conditions. Therefore, the results at large distances from the release point are rather uncertain.

As an alternative to the Pasquill scheme, methods have been developed in which the dispersion parameters are determined as continuous functions of meteorological data obtained by mast measurements (1203). Several annual cycles of mast data are available for the Swedish nuclear power stations. Plant-specific dispersion calculations can therefore be carried out on the basis of statistical information.

As the concentration in the plume decreases by diffusion, depletion also occurs because of radioactive decay and the fallout of particles on the ground. This fallout takes the form of dry deposition, when the plume impacts on the ground, or wet deposition, which involves precipitation. Dry deposition is usually characterized by a *deposition velocity* (m s^{-1}) which expresses the ratio of the deposition rate per unit of ground surface area ($\text{Bq m}^{-2} \text{s}^{-1}$) and the activity concentration (Bq m^{-3}) above the surface.

While dry deposition is mainly a surface effect, wet deposition is a volume

effect since the removal of radioactive material occurs in the whole plume. The deposition rate is defined by a *washout coefficient* (s^{-1}), the magnitude of which depends upon the precipitation intensity. The washout coefficient is a measure of the relative change of the radioactive particulate matter in the plume per unit of time.

If the release is hot, for example from a fire or an overheated reactor core, the plume may rise. In the Chernobyl accident, where a fire occurred, it is estimated that a large part of the smaller radioactive particles rose more than a thousand metres in the atmosphere (cf 13.7.5). The various phases of *plume rise* are illustrated in Fig. 12.4. It is evident that plume rise can have a large effect on the ground-level concentration close to the release point.

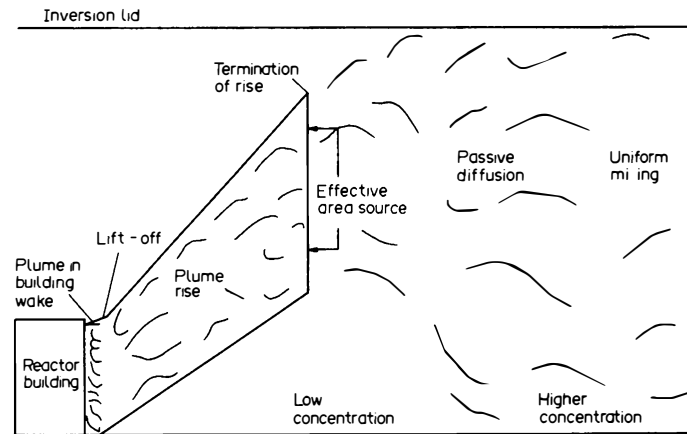


FIG. 12.4. Typical history of plume rise. From W Nixon et al, Accident Consequence Analysis, *Nucl. Energy*, Vol 24, No 4, 1985

Considerable uncertainty exists as to the details of plume rise. This is also true for turbulent building wakes, which are important for ground-level releases. The effects of plume rise and building wakes can, however, be approximately accounted for in the Gaussian formulation of atmospheric dispersion.

12.1.2 Radiation doses

When the spatial and time-dependent radionuclide concentration in the air and on the ground is known, the doses that would be received by individuals and populations can be estimated. The dose is the radiation energy absorbed per mass unit of a body (cf 6.1.2). Radiation doses are calculated for sensitive organs such as the bone marrow, thyroid and lungs, as well as

for the whole body. The most important exposure pathways are characterized by the way in which the radiation dose is received (Fig. 12.5):

Cloud dose	The dose to all organs as a result of exposure to gamma radiation from the passing cloud ("cloud-shine").
Inhalation dose	The dose to certain organs as a result of radiation from substances entering the body through inhalation.
Ground dose	The dose to all organs as a result of exposure to gamma radiation from materials deposited on the ground ("ground-shine").
Ingestion dose	The dose to certain organs as a result of radiation from substances entering the body in contaminated foodstuffs.

The dose received from a passing cloud and that received from ground deposition are examples of *external* doses, whereas inhalation and ingestion result in *internal* doses (cf 6.1.3). For example, iodine is taken up selectively by the thyroid.

The starting point for calculating the *cloud dose* is the time-integrated airborne concentration (Bq s m^{-3}) of each radionuclide as a function of the distance from the release point. The integration is carried out over the duration of the plume passage, which is equal to the duration of the release, or over the residence time of the individual, whichever is shorter. The dose is calculated by adding the contributions from the whole cloud. If the size of the cloud is large compared to the range of the radiation, the cloud can be considered semi-infinite, which considerably simplifies the spatial integration over the cloud. This approximation is useful at large distances from the release point if the plume is broad (Pasquill A to D).

The *ground dose* is calculated, usually at an exposure point 1 m above

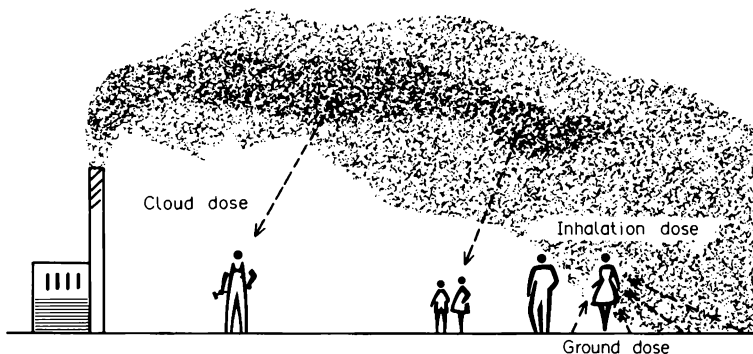


FIG. 12.5. Illustration of the concepts of cloud dose, ground dose and inhalation dose. From *More Effective Emergency Preparedness, Vol 5 Consequence Descriptions*, National Swedish Institute for Radiation Protection, Stockholm, December 1979

the ground, from the deposited concentration (Bq m^{-2}) integrated over the contaminated surface and the exposure time. Since the dose largely originates from activity in the vicinity, the ground deposition can be assumed to be equally distributed over an infinite surface with a concentration equal to that immediately below the exposure point. Unlike the case of a passing cloud, contribution to the ground dose is obtained also after the plume passage. Radioactive decay must be taken into account when calculating the dose over an exposure time which is long compared to the half-life of the particular nuclide.

The *inhalation dose* mainly originates from the plume passage. It is usually calculated as the product of the time-integrated airborne concentration (Bq s m^{-3}) and the rate of inhalation ($\text{m}^3 \text{s}^{-1}$). Although inhalation mainly causes exposure to the respiratory tract, other organs, such as the thyroid gland and red bone marrow, will also be exposed by the transport of specific nuclides, mainly iodine, cesium and strontium, from the lungs. The organ doses are calculated from the inhaled activity using internationally accepted *dose conversion factors* (1204).

The *ingestion dose* is calculated in a similar way via the deposited activity, the particular food chain and the consumed quantity. An example is the case of iodine in the grass-cow-milk chain. However, except for milk, the ingestion dose usually involves long delay times which allow ample time for measurements and protective action.

The calculated doses reflect the extension of the plume and are therefore strongly influenced by the release height, plume rise and meteorological conditions. In the first approximation, the dose is proportional to the local activity concentration. Figure 12.6 shows schematic isodose curves for an assumed “cold”, elevated release of noble gases and iodine for different meteorological conditions. A cold plume and an unstable atmosphere result in relatively fast down-transport of activity and therefore high doses close to the plant, while under stable weather conditions the plume is more concentrated and alights at a greater distance from the plant showing a relatively high concentration there.

12.1.3 Dose reduction

A distinction must be made between *potential* doses and *expected* doses. The potential dose is the dose an individual would obtain if he were to remain outdoors continuously. In practice, the dose is reduced by various shielding effects. Staying indoors gives significant protection, mainly because the building prevents the entry of airborne particles. Even small wooden houses reduce the cloud dose and the ground dose from the plume passage and ground deposition to less than half the outdoor value. In large multi-family houses, as well as in commercial and office buildings, the dose may be reduced to 1/50.

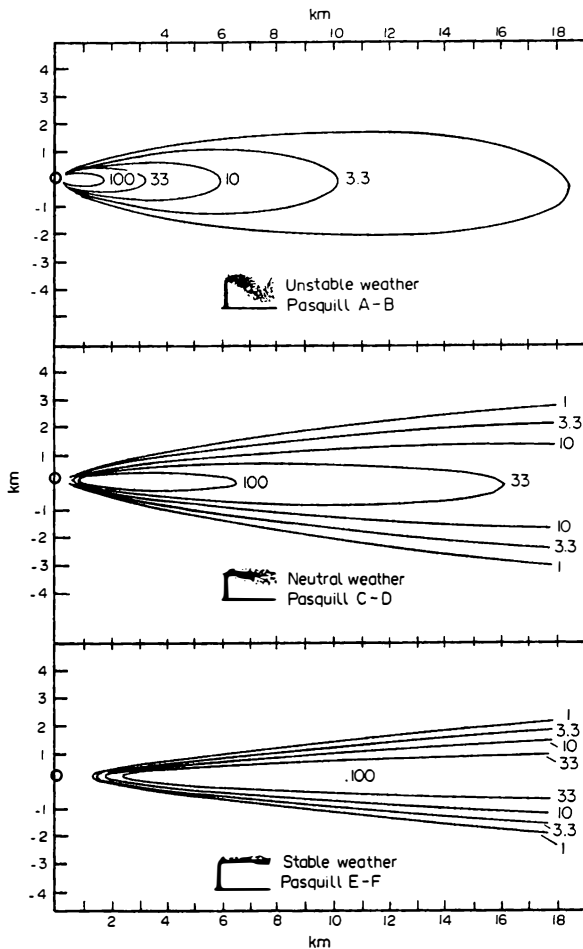


FIG. 12.6. Relative isodose curves for various meteorological conditions: release height, 100 m; wind speed, 1 m s^{-1} ; release duration, 30 min. The curves apply to the cloud dose, ground dose and inhalation dose. From *Reactor Accidents with Extensive Fuel Damage*, Studsvik Report KS-81/12, 1981

Staying indoors also offers some protection against the inhalation of radioactive particles. Stable (inactive) iodine tablets can reduce the uptake of radioactive iodine by blocking the thyroid. If the tablets are taken before inhalation, the thyroid dose will be reduced to less than $1/20$. The uptake is also considerably reduced if the tablets are taken within a few hours after inhalation. Simple breathing protection is also effective in reducing the inhalation dose.

Evacuation of the area over which the plume is expected to pass can completely eliminate exposure if it is carried out before the release. Successful evacuation requires adequate warning time and is considerably affected

by local conditions and by whether the evacuation is planned or improvised. Evacuation after ground deposition has occurred can be justified in some circumstances. Long-term countermeasures include land interdiction, banning of foodstuffs, and decontamination of contaminated areas.

The effects of shielding and protective action are taken into account by multiplying the calculated potential doses by appropriately chosen factors. Examples of such factors are given in Table 12.2. The National Swedish Institute of Radiation Protection used a standard value of 0.33 for the *shielding factor* for the cloud dose and ground dose (1205). The doses so obtained are assumed to represent mean values for a population with normal living habits in a temperate climate.

TABLE 12.2. *Dose reduction after countermeasures*

Countermeasure	Factor by which the calculated dose should be multiplied		
	Cloud dose	Inhalation dose	Ground dose
<i>Residence indoors</i> with closed ventilation before and airing after plume passage	0.1–1	0.2–0.5	0.03–0.33
<i>Evacuation</i> after release:			
within 12 hr instead of 24 hr			0.6–0.9
within 6 hr instead of 24 hr		—	0.4–0.8
<i>Iodine tablets</i> before inhalation		< 0.05	
2 hr after inhalation		0.3	
5 hr after inhalation		0.5	
<i>Natural drainage:</i>			
Residential area		0.5 ^a	
Farmland and forest			0.9 ^a
<i>Decontamination or trench-ploughing plus drainage:</i>			
Town and farmland			0.1 ^a
Forest			0.9 ^a

^a Excluding the “normal” shielding factor of 0.33.

Source: National Institute for Radiation Protection, *More Effective Emergency Preparedness*, Stockholm, December 1979

12.1.4 Health effect models

The absorption of radiation energy by a cell or tissue causes a chain of physical, chemical and biological reactions resulting in damage. The harmful effects of radiation may appear shortly after exposure or much later in the form of cancer or genetic effects (cf 6.1.3). Acute or *early* effects occur only when the radiation dose is high enough. The greater the dose, the more severe these effects (Fig. 12.7). The latent or *late* effects are stochastic in nature, i.e. they occur at random but with a frequency that increases with the radiation dose.

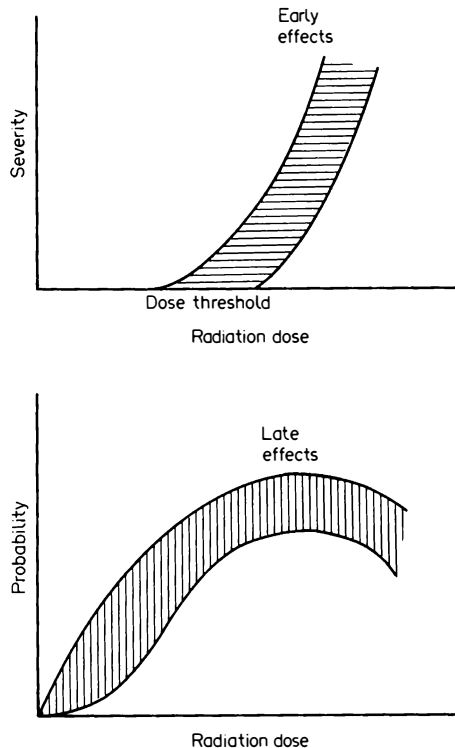


FIG. 12.7 The dose-effect relationship for early effects, such as acute radiation sickness, and the dose-response relationship for late effects, such as cancer and genetic effects. The probability of late effects decreases at high doses since the early effects then dominate the fatality risk. From *German Risk Study. Nuclear Power Plants*, Verlag TÜV Rheinland, 1980

Because of the different kinds of health effects, there is no simple relationship between dose and effect. The effects must be estimated for each type of effect. Due to the lack of empirical data, the results are uncertain. For example, Fig. 12.8 shows the probability of death from *acute radiation sickness* as a result of whole body exposure. Since the critical organ is the bone marrow, the *bone marrow dose* is used synonymously with the whole body dose. The diagram shows that death only occurs at doses higher than 1–2 gray (Gy). At 3–5 Gy there is a 50% possibility of survival and at 6 Gy the exposure is almost certainly fatal. The critical period is 3 weeks after exposure.

A characteristic of early effects is the *dose threshold* below which no effect appears. At low doses the consequence is entirely determined by latent effects which only manifest themselves after 10–20 years and over a prolonged period of time. There is some disagreement as to the extent of the cancer risk from low radiation doses. Because of the random variation of

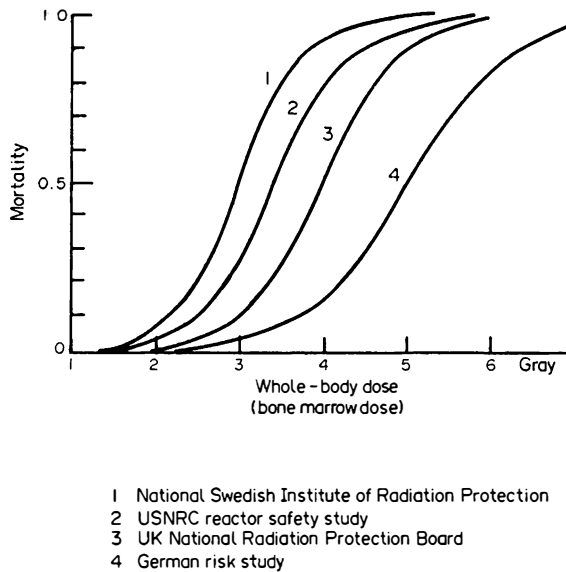


FIG. 12.8. Dose-mortality criteria for acute radiation sickness. The difference between the curves partly depends on the degree of medical treatment assumed

the cancer incidence from causes other than radiation, it is not possible to track any extra cases caused by doses that are slightly higher than those from the natural background radiation.

A linear relationship between cancer risk and radiation dose, without a threshold effect, is usually assumed (Fig. 12.9). The slope of the line is determined by extrapolation of the observed increased cancer incidence from high radiation doses. This method is believed by most experts to result in an overestimation of the cancer risk. With the linear hypothesis, the cancer fatality risk (the cancer mortality) is estimated at 0.01–0.03 per gray for whole body exposure. The risk of acquiring cancer (the cancer incidence) is about twice as great. The risk of serious genetic effects is estimated at about 0.004–0.008 per gray.

The *collective dose* is used to calculate the health effects in an exposed population. The collective dose is the product of the number of exposed individuals and their mean effective dose equivalent (cf 6.6.1). For example, the collective dose 1 mansievert (manSv) is obtained if 1000 people receive 1 millisievert (mSv) or if 100 persons receive 10 mSv. Because of the linear relationship, the risk of death from cancer can be given the significance of 1–3 cases per 100,000 people receiving an average dose of 1 mSv. For comparison, the annual dose to the world's population due to the natural background radiation is about 2 mSv per person.

The linear relationship means that very low dose increments also result in an increase of the cancer risk. A large number of fatalities has been

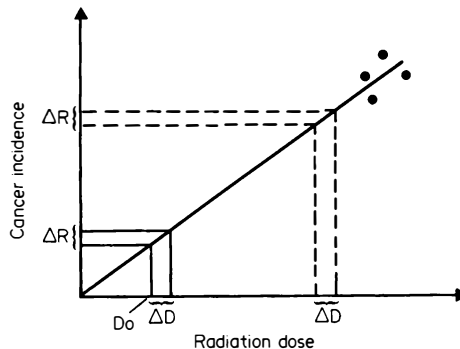


FIG. 12.9. The linear dose-response hypothesis for cancer. D_0 = the natural background dose. ΔD = incremental dose. ΔR = incremental risk. The incremental risk for a given incremental dose is always the same, irrespective of the dose level

estimated for certain very unlikely reactor accidents due to the dispersion of radioactive substances over a large area during unfavourable weather conditions. In spite of the low individual dose, a high collective dose is obtained because of the large number of people involved. Most of the calculated effects in these cases are caused by dose increments which are lower than the total dose due to the natural background radiation received by an individual during his lifetime.

12.2 Deterministic Analysis

In deterministic consequence analysis, the atmospheric dispersion and the environmental doses are calculated based on a postulated release. Such calculations are performed in the licensing process and are presented in safety analysis reports. They have also been used for emergency preparedness planning. Since the publication of the Reactor Safety Study, consequence analysis and risk assessment is mostly based on more realistic source terms. The probabilistic risk analysis is treated in section 12.3.

12.2.1 Licensing calculations

In the early 1960s the U.S. Atomic Energy Commission established siting criteria based on bounding values for radiation doses to the population in the vicinity of nuclear power plants (cf 7.1.1). The criteria involved the definition of protection zones, the extent of which was determined by reference levels for the whole-body dose (bone marrow dose) and the inhalation dose (thyroid dose). These criteria were also applied in the licensing calculations for the Swedish power plants.

For the assessment of consequences, the concept of *Maximum Credible Accident* (MCA) was introduced. The MCA was defined as a double-ended break of a main coolant pipeline, i.e. the same event as the design basis accident (DBA-LOCA) for the emergency cooling systems and the reactor containment (cf 9.1.3). During MCA, 15% of the total core inventory of fission products is postulated to be released to the reactor containment (1206). The released material is assumed to consist of the total core inventory (100%) of radioactive noble gases, half of the core inventory (50%) of radioactive iodine and 1% of the core inventory of "solid" fission products.

Half of the released iodine and the entire amount of solid fission products is assumed to deposit on the walls and surfaces of the reactor system and containment. Thus 100% of the noble gas and 25% of the iodine inventory are available for leakage to the environment. Five percent of this 25% is assumed to exist in particulate form, 4% as organic iodine (methyl iodide) and the remaining 91% as elemental iodine. The activity of the radionuclides decreases through decay during their residence time in the reactor containment and reactor building (if any). The amount of airborne particulates further decreases through scrubbing during containment spraying. The remaining mixture of noble gases and iodine is assumed to leak out of the containment at a rate determined by the technical specifications for the containment. The release occurs at ground level in pressurized water reactors (without stack), while in boiling water reactors most of the leakage occurs via the reactor building ventilation system to the stack.

The source terms thus postulated were established in the late 1960s in the U.S. Atomic Energy Commission's regulatory guidelines for the analysis of MCA (1207). The guidelines also contain instructions for the calculation of atmospheric dispersion and dose conversion. The dispersion factor is based on unfavourable combinations of Gaussian distributions in accordance with the Pasquill scheme, depending on the height and duration of the release. Alternatively, the dispersion factor is calculated on the assumption that the accident has occurred under weather conditions which are worse, as far as the doses are concerned, than those statistically expected to occur at the site for 95% of the time (1208).

The environmental consequences of DBA-LOCA and other postulated accidents are analysed as a basis for the licence application. Common to these *licensing calculations* is the fact that they are carried out with conservative assumptions for the performance of safety systems as well as for the magnitude and dispersion of release. Table 12.3 presents some typical results for U.S. conditions. The calculated dose levels refer to an individual remaining outdoors for 2 hours at any point on the boundary of the exclusion zone, in this case 975 m from the nuclear power plant.

The calculated doses are well below the prescribed limit values. The whole-body doses are comparable to the dose which may be obtained in a medical X-ray examination.

TABLE 12.3. *Examples of calculated doses during postulated accidents*

Accident	Whole-body dose mSv	Thyroid dose mSv
Loss of coolant accident (DBA-LOCA)	30	1550
Control rod ejection	< 10	< 10
Refuelling accident	20	20
Main steam line break	10	160
Dose limit 10 CFR 100	250	3000

Source: U.S. Atomic Energy Commission, *The Safety of Nuclear Power Reactors and Related Facilities*, USAEC Report WASH-1250, July 1973

12.2.2 Ringhals 3/4

The environmental consequences of DBA-LOCA for the identical pressurized water reactors Ringhals 3 and 4 have been analysed in the common Final Safety Analysis Report (1209). It is assumed that 100% of the inventory of noble gases and 50% of the iodine is released from the fuel. Half of the released iodine is deposited on the walls and surfaces of the reactor system and containment and some is removed by the containment spray system. The gas leakage is assumed to correspond to 0.1% of the containment volume during the first 24 hours and thereafter to 0.05% per day for the following 29 days.

The water leakage is assumed to be to 24 m³ during the first day and thereafter 12 m³ per day for the remaining 29 days. One percent of the iodine in the water which leaks out is assumed to vaporize immediately. Half of the vaporized iodine is assumed to deposit on cold surfaces and walls, i.e. 0.5% of the iodine contained in the water will reach the environment.

In all, ten noble gases and five iodine isotopes are allowed for. The amount of activity released is shown in Table 12.4. The short-lived nuclides decay quickly during their residence in the containment. Xenon-133 and iodine-131 dominate the activity release during the assumed 30 days duration of the release.

For each nuclide and time interval, the dose at various distances from the plant is calculated as if it were proportional to the activity concentration.

$$D_i = Q_i M F_i S \quad (12.1)$$

where D_i = dose for nuclide "i" (Sv),

Q_i = released activity (Bq),

M = dispersion factor (s m⁻³),

F_i = dose conversion factor (Sv Bq⁻¹ s⁻¹ m³),

S = shielding factor (-).

TABLE 12.4. Ringhals 3/4 DBA-LOCA. Calculated activity release (Bq) to the environment during various time intervals after the accident

Nuclide	Half-life	0 - 8 hr		8 - 24 hr		24 - 720 hr	
		From gas	From water	From gas	From water	From gas	From water
Kr-85	10.7 y	1.0E13		2.0E13		4.3E14	
Kr-85 m	4.4 h	2.3E14		8.2E13		3.6E12	
Kr-87	76.4 m	1.7E14		2.2E12		—	
Kr-88	2.8 h	4.7E14		7.4E13		7.4E11	
Xe-131 m	11.8 d	6.7E12	1.9E11	1.3E13	2.2E12	1.8E14	5.0E13
Xe-133	5.29 d	2.0E15	6.7E13	3.9E15	4.1E14	2.0E16	5.7E14
Xe-133 m	2.26 d	5.0E13	4.7E12	9.3E13	2.8E13	2.4E14	3.3E13
Xe-135	9.14 h	6.0E14	3.5E14	8.7E14	5.1E14	2.9E14	4.5E14
Xe-135	15.6 m	2.0E14	6.9E14	1.2E14	4.7E14	1.4E13	1.4E13
Xe-138	17.5 m	9.2E13	—	—	—	—	—
I-131	8.06 d	9.5E12	9.0E12	1.7E13	2.3E13	1.3E14	2.0E14
I-132	2.28 h	6.4E12	3.7E12	5.6E11	7.8E11	—	—
I-133	20.3 h	1.9E13	1.7E13	2.4E13	3.2E13	1.7E13	2.7E13
I-134	53 m	5.7E12	1.0E12	—	—	—	—
I-135	6.68 h	1.4E13	1.1E13	7.7E12	1.1E13	9.3E11	1.5E12

Source: Swedish State Power Board, Ringhals 3/4 Final Safety Analysis Report, Chapter 15, 1983

The dispersion factor gives the specific concentration, i.e. activity concentration (Bq m^{-3}) per unit of release (1 Bq s^{-1}). It is calculated for various release heights as a function of the distance from the release point in the plane of the plume's centreline at ground level. The calculated dispersion factor for Ringhals (Fig. 12.10) is based on local meteorological observations during several years. The curves cannot be referred to specific weather conditions but represent frequency distributions chosen so that the specific activity concentration for the particular duration of the release is exceeded only 5% of the time. The curve for long periods is lower than the curve for short periods, since unfavourable weather conditions seldom persist for long periods of time.

By multiplying the activity release according to Table 12.4, and the dispersion factor according to Fig. 12.9, the activity concentration integrated over the respective time interval is obtained. The dose conversion factors in equation (12.1) are nuclide-specific and depend on the type of dose

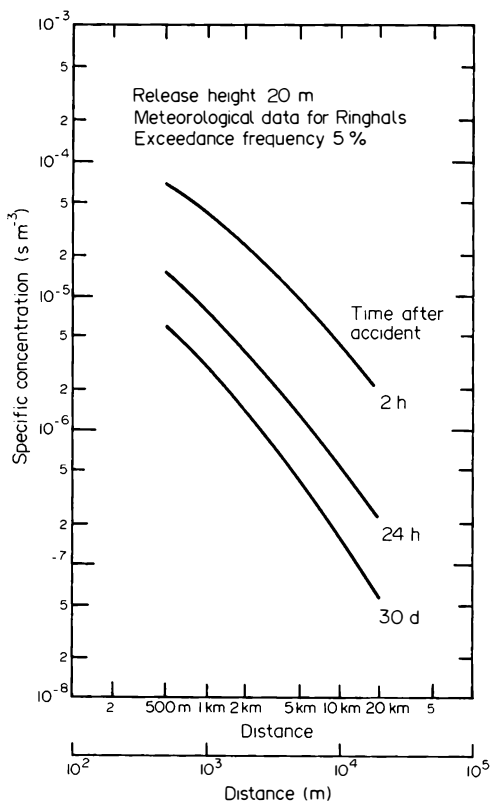


FIG. 12.10. Dispersion factor for ground-level release at Ringhals. Adapted from *Ringhals 3/4 Final Safety Analysis Report*, Swedish State Power Board, 1984

involved. In this particular case, the external whole-body dose was calculated from noble gases and iodine in the radioactive cloud, and the inhalation dose, both the whole-body dose and the thyroid dose, was calculated from iodine. A shielding factor of 0.8 was used for time intervals ≤ 2 hours and 0.35 for longer intervals.

The sum of the cloud dose and the whole-body part of the inhalation dose gives the total whole-body dose. The calculated doses in the direction where they are highest are presented in Table 12.5.

A sensitivity study showed that the leakage rate during the first day is important for the calculated doses. At 0.3% leakage per day (instead of 0.1% per day), the 30-day doses at 2 km distance increased to 7.8 mSv for the whole-body dose and to 1000 mSv for the thyroid dose. However, these values still fall below the reference values of 250 mSv for the whole-body dose and 3000 mSv for the thyroid dose (cf 7.1.1).

TABLE 12.5. *Ringhals 3/4 DBA-LOCA. Calculated doses (mSv) during various time intervals after the accident*

Interval (hours)	Whole-body dose		Thyroid dose	
	0.5 km	2 km	0.5 km	2 km
0-8	7.4	2.6	500	180
8-24	0.4	0.08	200	50
24-720	0.2	0.04	530	120
0-720	7.9	2.7	1230	350

Source: Swedish State Power, *Ringhals 3/4 Final Safety Analysis Report*, Chapter 15, 1983

12.2.3 Forsmark 3

The licensing consequence analysis for Forsmark 3 was carried out along largely the same lines as that of Ringhals 3/4 (1210). In addition to the established guidelines for fission product release, it is assumed that 25% of the core inventory of radioactive cesium and 0.5% of that of strontium is available for leakage. Removal of iodine as well as of cesium and strontium by the containment spray system is assumed to be effective. The gas leakage is assumed to be 1.33% of the containment volume during the first day and 0.67% per day during the following 29 days.

A total of twenty-four nuclides, four isotopes of krypton, seven of xenon, five of iodine, four of cesium and four of strontium, were considered. The offsite release was calculated taking into account the release from the fuel, the transport and removal processes in the reactor and the containment, and the specified gas leakage from the containment. Ninety per cent of the leaking gas is assumed to pass coal filters on its way out through the stack.

The remaining 10% is assumed to constitute a ground-level release to the environment, at a height of 20 m. The filter effect is assumed to be 90% for iodine, cesium and strontium.

Most of the iodine, cesium and strontium is transferred to the water phase by scrubbing. The relative water leakage is assumed to be the same as the gas leakage, i.e. 1.33% of the volume during the first day and thereafter 0.67% per day. One per cent of the leaking iodine and 0.1% cesium and strontium are assumed to vaporize immediately. Half of the vaporized material is assumed to deposit on surfaces and walls. Ten per cent of the remaining vaporized material is assumed to escape directly into the environment while 90% reaches the environment via coal filters and the stacks where 90% of the material is removed. The total activity release to the environment is shown in Table 12.6.

TABLE 12.6. *Forsmark 3 DBA-LOCA. Calculated activity release (Bq) to the environment during various time intervals after the accident*

Interval	Leakage route	Iodine	Cesium	Strontium
0-2 h	gas	3.9 E14	1.6 E14	4.5 E12
	water	—	—	—
2-6 h	gas	9.8 E13	6.0 E12	4.7 E11
	water	6.9 E12	4.4 E10	8.3 E9
6-24 h	gas	1.5 E14	3.3 E10	4.8 E9
	water	1.8 E13	1.8 E11	2.4 E10
24-720 h	gas	3.9 E14	—	—
	water	4.6 E13	3.1 E12	2.4 E11

Source: *Forsmark Nuclear Power Plant Unit 3. Final Safety Analysis Report*, Chapter 9, Forsmark Power Group AB, 1984

The dispersion factor was calculated from meteorological statistics for Forsmark (Fig. 12.11). Dose conversion factors are given for each nuclide and dose type. The external dose from deposited activity is calculated as well as the cloud dose and the inhalation dose. The shielding factor for the cloud dose and the ground dose is set at 1 and 0.7 respectively for the interval 0-7 hours, at 0.6 and 0.2 from 7-24 hours, and at 0.7 and 0.33 for time intervals exceeding 24 hours.

The calculated doses in the direction where they are highest are shown in Table 12.7. For example, the total whole-body dose is 46 mSv at a distance of 0.5 km and 18 mSv at a distance of 2 km. The corresponding thyroid dose is 1100 and 440 mSv. As for Ringhals 3/4, the thyroid doses have been calculated for children. Children are supposed to receive a thyroid dose which is three times as high as that of adults for the same intake of radioiodine.

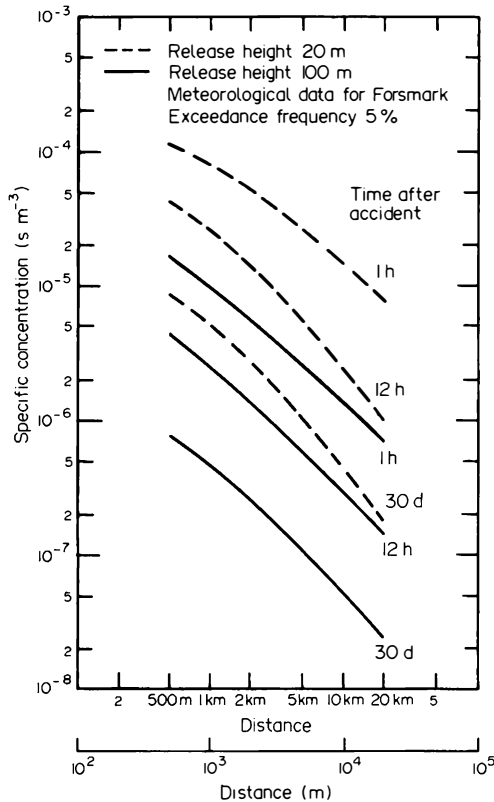


FIG. 12.11. Dispersion factors for elevated and ground-level releases at Forsmark. Adapted from *Final Safety Analysis Report Forsmark Unit 3*, AB Asea-Atom and Swedish State Power Board, 1983

12.3 Probabilistic Analysis

Deterministic analysis deals with the consequences of “model accidents” with postulated releases and without considering the probability of the accidents. In probabilistic risk analysis (PRA), both the probability and the consequences are estimated. The consequence analysis starts with the release sequences and external source terms discussed in Chapter 11. This section reviews the overall calculational model and examines the results of some risk studies. Finally, the importance of the source terms is discussed.

12.3.1 Calculational model

A complete risk analysis comprises four stages:

- *Plant analysis*, in which core damage sequences are identified and core damage frequencies are estimated.

TABLE 12.7. Forsmark 3 DBA-LOCA. Calculated doses (mSv) during various time intervals after the accident

Dose type	Interval (hr)	Distance (km)					
		0.5	1	2	5	10	20
External whole-body from cloud	0- 2	10	5.7	3.9	2.3	1.2	0.56
	2- 6	3.3	2.0	1.1	0.46	0.23	0.11
	6- 12	1.0	0.61	0.34	0.13	0.057	0.027
	12- 24	0.57	0.34	0.18	0.065	0.030	0.014
	24-720	0.46	0.27	0.14	0.054	0.023	0.010
External whole-body from ground	0-720	16	8.9	5.7	3.0	1.5	0.72
	0- 2	3.1	1.9	1.4	0.82	0.42	0.20
	2- 6	3.5	2.2	1.5	0.89	0.46	0.21
	6- 12	1.2	0.73	0.50	0.28	0.15	0.067
	12- 24	0.93	0.57	0.38	0.21	0.11	0.049
Inhaled whole-body dose	24-720	20	12	7.7	4.0	2.0	0.93
	0-720	28	17	11	6.2	3.2	1.4
	0- 2	2.1	1.3	0.94	0.56	0.29	0.13
	2- 6	0.16	0.11	0.062	0.026	0.013	0.003
	6- 12	0.025	0.015	8.4E-3	3.1E-3	1.5E-3	6.1E-4
Thyroid dose	12- 24	0.020	0.012	6.6E-3	2.5E-3	9.2E-4	4.5E-4
	24-720	0.059	0.034	0.018	5.6E-3	2.6E-3	1.1E-3
	0-720	2.4	1.5	1.0	0.60	0.31	0.14
	0- 2	760	460	340	200	100	48
	2- 6	110	73	42	18	8.6	3.9
Thyroid dose	6- 12	48	29	16	5.9	2.5	1.2
	12- 24	43	27	14	4.7	2.0	0.97
	24-720	120	69	36	12	5.4	2.3
	0-720	1100	620	440	240	120	56

Source: Forsmark Nuclear Power Plant Unit 3. Final Safety Analysis Report, Chapter 9, Forsmark Power Group AB, 1984

- Containment analysis*, in which the behaviour of the core melt in the reactor vessel and containment is studied and the probability of containment failure is estimated.
- Source term analysis*, which assesses the amount of radionuclides released and the characteristics of the release.
- Consequence analysis*, predicting the environmental dispersion of the radionuclides, and estimating the radiation doses and health effects.

Plant analysis is described in Chapter 10. On the basis of reported studies, the mean core damage frequency is estimated at $10^{-4} - 10^{-6}$ per year of reactor operation in both boiling water and pressurized water reactors. The studies also indicate that different sequences tend to dominate the core damage frequency in PWRs as compared to BWRs, although the results are highly plant-specific.

Containment and source term analysis is treated in Chapter 11. Certain core damage sequences are shown to result in containment failure and uncontrolled releases to the environment. Release categories are defined with regard to the containment failure mode and the release characteristics. Each core damage sequence can be assigned to one or several release categories. A total frequency for each release category is obtained by summing all release frequencies within each release category.

As shown in this Chapter, the analysis of offsite consequences is carried out in three steps. Firstly, the atmospheric dispersion of the radioactive cloud is calculated, including the fallout of radionuclides on the ground. The expected doses for different exposure pathways to the population are then estimated taking into account protective action and countermeasures. Finally, on the basis of assumed dose–effect and dose–response relationships, the number of early and late effects is estimated.

The offsite consequences depend on the magnitude and composition of the release as well as on the prevailing meteorological conditions and the population distribution downwind of the radioactive cloud. The probability of a particular consequence is determined by a combination of partial probabilities for the release, the weather conditions and the wind direction as follows:

$$p_{\text{conse}} = p_{\text{release}} \times p_{\text{weather}} \times p_{\text{direction}}$$

The release probability is determined by the total frequency of the particular release category.

By combining the release categories with the weather conditions and wind directions, a large number of cases are obtained, each characterized by a frequency (probability per year) and a consequence. If several combinations of partial frequencies give approximately the same consequence, the combined frequencies are added. Hence, each consequence interval can be assigned a particular frequency (Fig. 12.12).

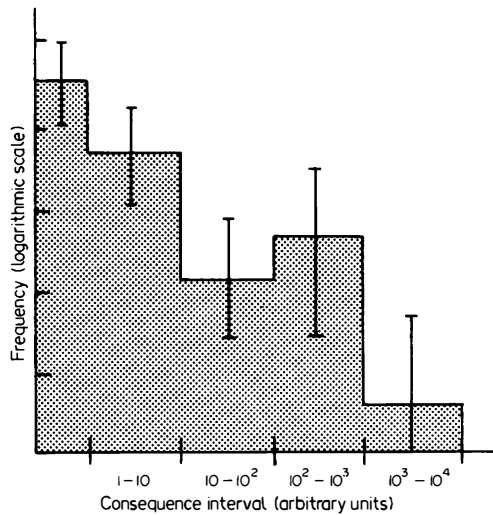


FIG. 12.12. Frequency distribution of consequences

Usually, it is of interest to determine the probability that the consequence, for example the number of fatalities, is greater than a certain value X . All frequencies for consequences $> X$ are then totalled to obtain the *complementary cumulative frequency distribution (CCFD)*, Fig. 12.13. The distribution is complementary and cumulative since it gives the frequency for the consequence being $> X$. The cumulative distribution itself gives the frequency for the consequence being $< X$.

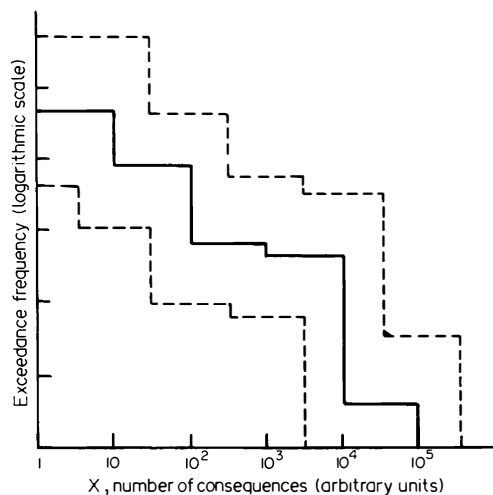


FIG. 12.13. Complementary cumulative frequency distribution of consequences

The CCFD is also known as the *exceedance frequency distribution*. The exceedance frequency is of particular interest when dealing with rare events with large consequences. The scales on the axes are then made logarithmic. The area under the curve (with due account to the logarithmic scales) is a measure of the *expectation value*, or the mean value of the consequence.

The dashed lines shown in Fig. 12.13 represent an uncertainty band, known as the *confidence interval*. The significance of the confidence interval is that the true curve falls within the interval with 90% probability. The confidence interval is obtained by considering all uncertainties in the estimation of both frequency and consequence.

12.3.2 The Reactor Safety Study

The Reactor Safety Study was the first complete probabilistic risk analysis for a nuclear power plant. It included both pressurized water and boiling water reactors (1211). The dominant core damage sequences are shown in Tables 10.2 and 10.3. The release categories are defined in Tables 11.9 and 11.10. Corresponding releases and frequencies are summarized in Table 11.11.

The Pasquill scheme, featuring six weather categories, was used to characterize the weather conditions. The data were obtained from meteorological statistics from six sites typical of the first hundred reactor units in the USA. A total of ninety weather sequences were characterized in this way with regard to thermal stability, windspeed and precipitation. Each weather situation was assigned a probability of 1/90.

The first hundred reactor units are distributed among sixty-eight nuclear power stations. The population distribution around each station was mapped in sixteen sectors in terms of the distance from the station. Each unit was assigned one of the six typical sites. For example, fourteen units were allotted to the first site type which resulted in $16 \times 14 = 224$ sectors with different population distributions. The population distribution in these 224 sectors was then used to generate sixteen representative sectors. Each representative sector was assigned a probability equal to the ratio between the number of original sectors in each representative sector and the total number of original sectors.

The frequency and consequences were calculated for each combination of release, weather and population distribution. The number of combinations is given in Table 12.8.

As an example of the results, exceedance frequencies for early and late fatalities are presented in Figs. 12.14 and 12.15. The curves represent average values for pressurized water reactors and boiling water reactors and refer to *100 reactors*. Corresponding uncertainty factors for early fatalities were estimated at 5 and 1/5 on the probability, and at 4 and 1/4 on the consequence, and for late fatalities, at 5 and 1/5, and 3 and 1/6, respectively.

TABLE 12.8. *Combination of data used in the Reactor Safety Study (1211)*

Reactor type	BWR	PWR
Number of units	34	66
Release categories	5	10
Weather sequences	90	90
Sites	6	6
Population sectors	16	16
Number of cases	43,200	86,400

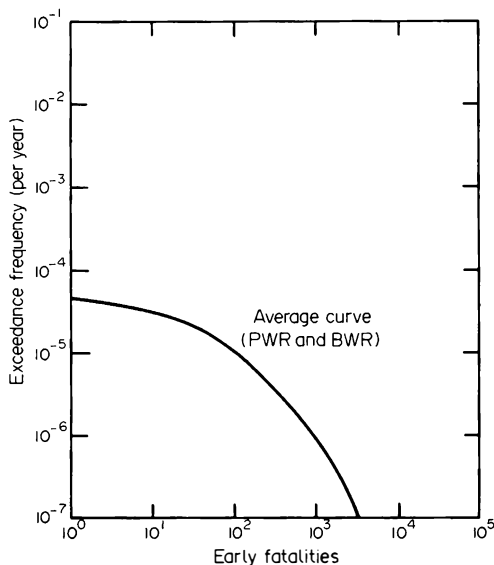


FIG. 12.14. Exceedance frequency distribution of early fatalities for 100 reactors according to the Reactor Safety Study

These uncertainties were later found to have been underestimated (cf 2.1).

Note that the number of late fatalities *per year* is given in Fig. 12.15. Since the late fatalities are assumed to occur over a 30-year period starting about 10 years after the accident, the total number of late fatalities (for a given exceedance frequency) is 30 times greater than the value on the abscissa in Fig. 12.15.

As previously mentioned (10.3.1), the total probability for a severe accident is estimated at 5×10^{-5} per reactor year. This means an expected core damage frequency of 1/200 per year for 100 reactors. However, only a few core damage sequences result in large releases. Moreover, only a few core damage sequences with large releases will have large consequences. This requires both unfavourable weather conditions and an unfavourable population distribution. These facts are illustrated in Table 12.9.

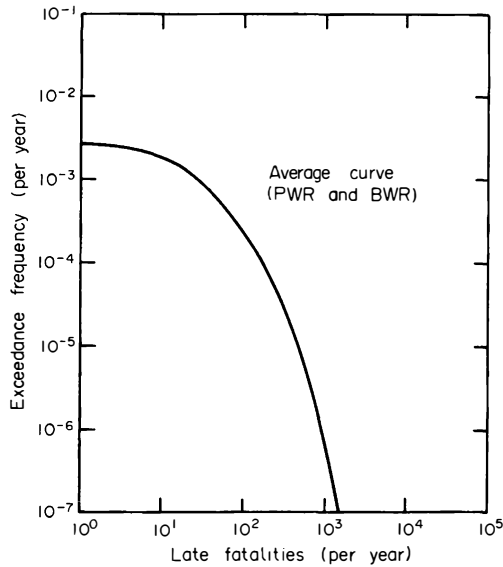


FIG. 12.15. Exceedance frequency distribution of late fatalities (cancer) for 100 reactors according to the Reactor Safety Study

TABLE 12.9. *The probability (per year) that the number of fatalities will equal or exceed the given values for 100 reactors*

Probability per year	Early fatalities	Late fatalities ^b
	per year	
1 in 200 ^a	< 1.0	< 1.0
1 in 10,000	< 1.0	< 1.0
1 in 100,000	110	460
1 in 1,000,000	900	860
1 in 10,000,000	3300	1500

^aProbable core damage frequency for 100 reactors.

^bThe normal cancer fatality frequency for the particular population is 17,000 per year.

Source: U.S. Nuclear Regulatory Commission, *Reactor Safety Study*, USAEC Report WASH-1400, Washington D.C., 1975

Consequences with frequencies lower than 10^{-7} per year are not shown, since numbers so low are meaningless considering the uncertainty of the analysis.

12.3.3 The German Risk Study

In principle, the German Risk Study (1212) used the same methodology as the Reactor Safety Study, with some modification of the release categories, weather categories and population distribution to suit West German conditions. Core damage sequences were studied in a West German type pressurized water reactor (cf 10.3.2). The definition of release categories and the corresponding release frequencies are given in Table 12.10 (cf Table 11.9).

By combining eight release categories, 115 weather sequences, thirty-six wind directions and nineteen sites a total of 629,280 cases were obtained for which probability and consequence calculations were performed for twenty-five reactor units. The results were presented as distributions of exceedance frequencies versus consequences. Figures 12.16 and 12.17 provide examples for early and late effects. The dashed bars indicate 90% confidence intervals.

A comparison with the corresponding results of the U.S. Reactor Safety Study shows that, taking into account the different number of reactors involved, the calculated values for early effects are in agreement within the estimated confidence intervals. The number of late effects is greater in the German study, since a more conservative dose-response relationship was used (Fig. 12.18), and since the average population density in Europe is higher.

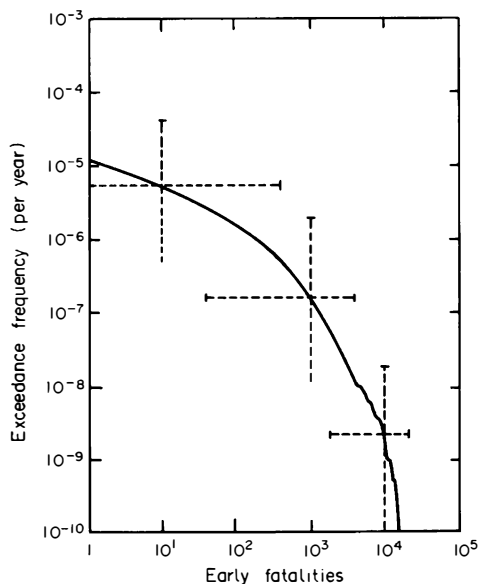


FIG. 12.16. Exceedance frequency distribution of early fatalities from twenty-five reactors according to the German Risk Study

TABLE 12.10. Release categories in the German Safety Study (1212)

Category	Description	Release frequency (PMY)
FK 1	Core meltdown with steam explosion	2
FK 2	Core meltdown with large containment leak (dia 300 mm)	0.6
FK 3	Core meltdown with medium containment leak (dia 80 mm)	0.6
FK 4	Core meltdown with small containment leak (dia 25 mm)	3
FK 5	Core meltdown with containment overpressure failure without filtering	20
FK 6	Core meltdown with containment overpressure failure and filtering	70
FK 7	Mitigated LOCA with large containment leak	100
FK 8	Mitigated LOCA with intact containment	1000

Relative contributions from various release categories are presented in Table 12.11.

Table 12.11 shows that only the first four release categories contribute to the expectation value for early effects. This is because a threshold value of 1 sievert was assumed in the dose–effect relationship, and because the doses

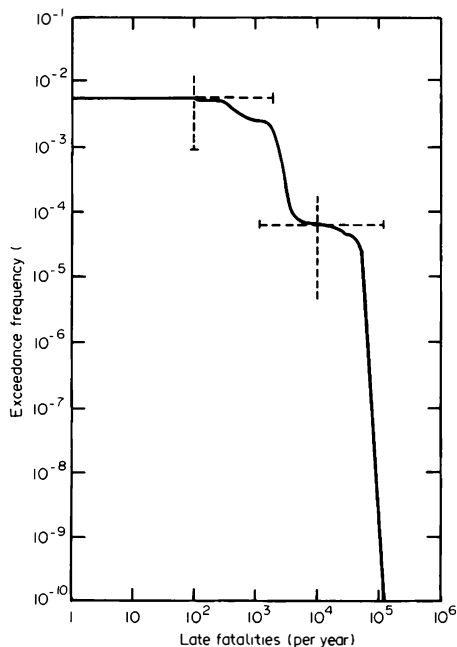


FIG. 12.17. Exceedance frequency distribution of late fatalities from twenty-five reactors according to the German Risk Study. Note that the number of fatalities is given per year

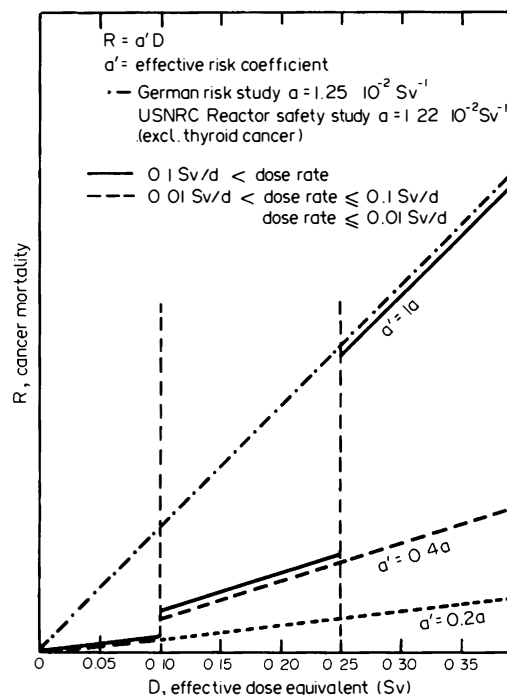


FIG. 12.18. Dose-response criteria for radiation-induced cancer used in the U.S. and German Safety Studies. From A Bayer, F W Heuser, Basic Aspects and Results of the German Risk Study, *Nucl. Safety*, Vol 22, No 6, 1981

TABLE 12.11. *Relative contribution by release category to the expectation value for early and late effects*

Category	Percentage contribution	
	Early effects	Late effects
FK 1	46.5	24.0
2	47.5	3.3
3	3.1	0.7
4	2.9	1.2
5	0	3.3
6	0	8.3
7	0	59.3
8	0	0.005

Source: *German Risk Study. Nuclear Power Plant, Main Report*, Verlag TÜV Rheinland, 1980

in release categories 5–8 did not reach the threshold value. However, all release categories contribute to the risk of late effects due to the linear dose–response relationship without any threshold dose.

A large part of the contribution to both early and late effects comes from release category 1, core meltdown and steam explosion. As previously mentioned (11.1.2), a steam explosion of sufficient strength to rupture the reactor vessel and containment is considered impossible on present evidence. Calculations were made both with and without this assumption. If the steam explosion case is disregarded, the *maximum* number of predicted early fatalities decreases from 14,500 to 5100 and the maximum number of late fatalities from 104,000 to 44,000. It should be noted that these maximum numbers are estimated to occur with the extremely low probability of 4.8×10^{-10} per year for twenty-five reactors.

The greatest contribution to the risk (expectation value) for late fatalities comes from category FK7, a mitigated LOCA with a large containment leak. In this case, the source terms are limited to those corresponding to gap release from the fuel (see 11.3.1). The relatively high probability for this release category makes a large contribution to the expectation value in spite of the relatively low number of fatalities (mean value 2400). It should also be noted that about 90% of the fatalities stem from radiation doses lower than 50 millisievert, i.e. the maximum permissible annual dose to radiation workers, recommended by the International Commission on Radiological Protection (cf 6.6.1).

12.3.4 Swedish consequence studies

The U.S. and German Safety Studies were *generic*, i.e. they were considered representative of types of reactors and sites. No similar study has been conducted in Sweden, although severe accidents and offsite consequences have been studied separately for specific plants. The first comprehensive consequence studies were carried out during 1977–8 for the Barsebäck nuclear power station. Barsebäck is located on the shore of Öresund in southern Sweden, 17 km from the centre of Malmö and about 25 km from Copenhagen. The station has two 600 MWe1 BWR units, commissioned in 1975 and 1977.

The studies for Barsebäck were aimed at illustrating the consequences of severe accidents, namely those corresponding to the release categories BWR 1, BWR 2 and BWR 3 of the Reactor Safety Study (Table 12.12). As previously mentioned, present evidence indicates that these source terms are too high. Nevertheless, some results are presented here because they are of fundamental and historical interest.

The Swedish dispersion model describing the weather in a certain wind direction based on meteorological observations was used in one of the studies carried out by Studsvik, Sweden (1213). About 17,500 hours (2

TABLE 12.12. Assumed source terms for the Barsebäck consequence study

Release category	Unit	BWR 1	BWR 2	BWR 3
Time after initiating event when release occurs	h	2	30	30
Release duration	h	0.5	3	3
Release height	m	25	10	25
Thermal power of release	MW	20.2	4.8	3.2
Fraction of core inventory released into environment:				
Xe-Kr		100	100	100
I		40	90	10
Cs-Rb		40	50	10
Te-Sb		70	30	30
Ba-Sr		5	10	1
Ru etc		50	3	2
La etc		0.5	0.4	0.4

Source: O Edlund, C Gyllander, *HS-77 Safety Study Barsebäck. Consequence Calculation*, Studsvik Report SM-78-5, 1978

years) of data from mast measurements taken at Risø, Denmark, were used as representative for Barsebäck. From this material, all cases with a particular wind direction were selected. The doses at various distances from the nuclear power station were calculated for these cases and then processed statistically. The model thus relates to real weather situations and takes into account the fact that the weather conditions in the selected wind direction may vary during the duration of the release.

Similarly, meteorological data from Risø were used in another Barsebäck study carried out for the Energy Commission (1214). In this study, the wind direction, wind speed and stability category were considered as statistical variables with distribution functions adjusted to observed, coherent values. Plume rise and deposition rate were also treated as statistical variables. Dose calculations were carried out for 1000 cases. This method involves the risk of obtaining unrealistic dose values, due to the fact that certain combinations of variables may be physically impossible.

The Pasquill scheme, with dispersion parameters for the various stability categories adjusted to Risø data, was used in a third study, carried out by Risø (1215), aiming at dose calculations in the direction of Copenhagen. The weather was assumed to remain unchanged for the duration of the release. The highest probability of obtaining large doses in Denmark is obtained for meteorological conditions with neutral stability (Pasquill D) and precipitation (Fig. 12.19). A smaller contribution comes from stable meteorological conditions (Pasquill F) with low windspeeds (Fig. 12.20).

As in the Danish study, Studsvik calculated the dose on the assumption that an individual will remain indoors for the duration of the plume passage and 24 hours afterwards and then leave the contaminated area. The number

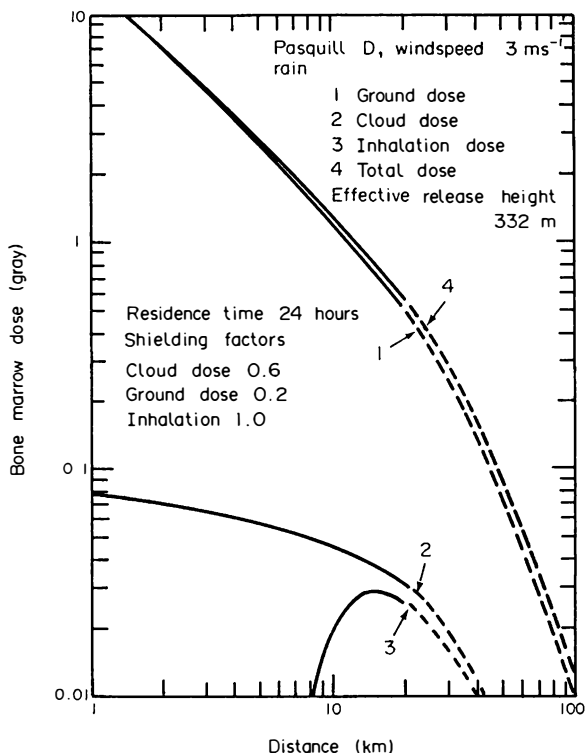


FIG. 12.19. Calculated bone marrow dose from a BWR I accident in Barsebäck, distributed into dose components. The doses are at ground level, vertically under the centreline of the plume. From Risø Report M-1905 (1215)

of health effects was estimated on the basis of the dose–effect and dose–response criteria according to Table 12.13. It should be noted that the linear relationship was not extrapolated to dose zero, but that threshold values were assumed, below which no late effects were supposed to appear.

The estimated number of health effects for a BWR 1 type accident is presented in Table 12.14. The calculations were made for two representative wind directions with high population densities: direction 70° Kävlinge and direction 240–260° Copenhagen. Collective doses were calculated up to 150 km from the nuclear power station. Assuming that an accident has occurred, median values (exceedance frequency 50%) and “worst” case exceedance frequency 0.1%) are shown for each direction. The numbers in each column are *not* additive, since they generally apply to different weather conditions.

The conclusion that no early fatalities will occur in the direction of Copenhagen is confirmed by the Danish study. If the number of cancer fatalities is assumed to be equally distributed over 30 years, an average mortality of

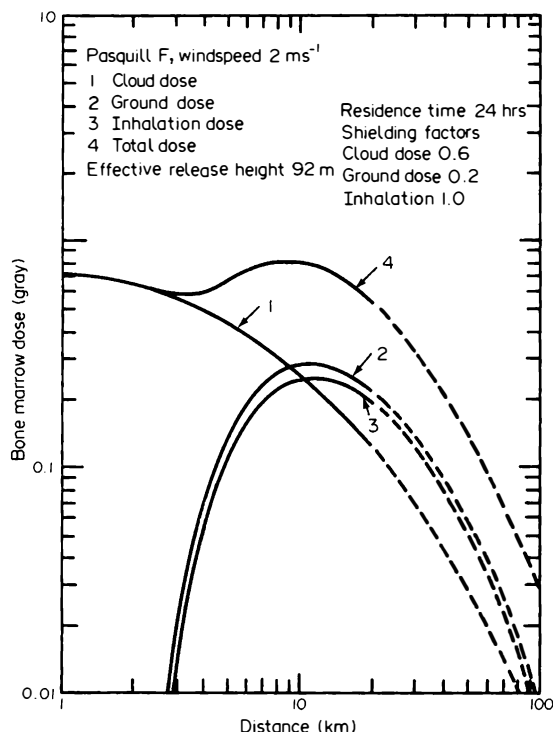


FIG. 12.20. Calculated bone marrow dose from a BWR 2 accident in Barsebäck, distributed into dose components. The doses are at ground level vertically under the centreline of the plume. From Risø Report M-1905 (1215)

TABLE 12.13. Relationship between dose and effects, used in the Studsvik consequence study for Barsebäck (1213)

Early effects (within about 3 weeks)	Dose (Sv) for 50% damage frequency	
Fatalities	3	
Radiation sickness	1.5	
Thyroid damage	250	
Late effects (after 5–50 years)	Risk coefficient (Sv ⁻¹)	Dose interval (Sv)
Leukemia	0.2×10^{-2}	0.01–3
Thyroid cancer	0.05×10^{-2}	0.03–10
Other cancers	1.3×10^{-2}	0.01–3

TABLE 12.14. *Estimated number of health effects from BWR 1 accident at Barsebäck*

	Bearing 70° (Kävlinge)		Bearing 240–260° (Copenhagen)	
	Mean value	“Worst” case	Mean value	“Worst” case
<i>Early effects</i>				
Fatalities	0	A few	0	0
Radiation sickness	0	30	0	40
Thyroid damage	13	450	0	0
<i>Late effects</i>				
Leukemia	5	63	46	470
Thyroid cancer	160	330	910	2600
Other cancers	30	410	300	3100

Source: O Edlund, C Gyllander, *HS-77 Safety Study Barsebäck. Consequence Calculations*, Studsvik Report SM-78/5, 1978

200 cases per year in the “worst” case is obtained. This number should be viewed in relation to the mortality from other causes of cancer, which for the particular population is about 3200 per year.

How large is the probability of the “worst” case? If BWR 1 type events are physically impossible, as many believe, the probability is zero. However, if we assume like the Reactor Safety Study a release frequency for this case of 1 per million reactor years, a frequency of $10^{-6} \times 1/17,500 = 6 \times 10^{-11}$ per reactor year is obtained for the “worst” case, since “worst” weather conditions only existed during one of the 17,500 hours covered by the statistics. The probability becomes 3×10^{-9} for the entire remaining lifetime of both Barsebäck reactors. Even allowing for the uncertainty of the estimate, this value is practically negligible.

It should be added that larger numbers of health effects were reported in the study carried out for the Energy Commission. However, due to the assumptions and conditions on which the worst cases for this study were based, the results must be assigned an even lower probability than that given above.

In the wake of the Three Mile Island accident, the National Institute of Radiation Protection investigated the consequences of severe reactor accidents in Swedish reactors (1205). The main aim was to provide a qualitative description of possible consequences for Swedish conditions and to show how the consequences depend on various factors, particularly how they can be influenced by emergency action.

The Reactor Safety Study data on release categories PWR 1 and BWR 1 (see Tables 11.11 and 12.12), i.e. core meltdown with steam explosion, were adapted to Swedish reactors and used as source terms. Atmospheric dispersion and ground deposition were calculated for various weather con-

ditions and wind directions at the Swedish nuclear power plants. External and internal doses as well as early and late effects were estimated.

The study showed the importance of the weather conditions for the dose level. At low wind speeds, the plume has time to rise near the release point which reduces the potential cloud and inhalation doses in the vicinity of the plant. In addition, the time to reach remote areas is longer, which allows some of the radionuclides to decay *en route*. The direct effects of the cloud are greatest with strong winds, partly because there is not enough time for countermeasures. The ground deposition is greater near to the plant with light winds, particularly when it rains.

The risks for early effects such as acute radiation sickness, pneumonia, thyroiditis and foetal damage as well as for latent cancer and genetic effects were estimated on the basis of dose calculations, where the ground dose is particularly subject to great uncertainty. More unfavourable dose–effect and dose–response criteria than those in the U.S. and German Safety Studies were used. For this reason, a higher maximum number of fatalities were obtained than in these studies. The probability of the “worst” case is of the order of 10^{-9} for the whole Swedish reactor programme (twelve reactors in 25 years).

12.3.5 U.S. re-evaluation studies

The Reactor Safety Study was a pioneer effort which established the method of probabilistic risk analysis and applied it to a pressurized water reactor, Surry-1, and a boiling water reactor, Peach Bottom-2, both of which were typical of the reactor technology in the late 1960s. The consequence analysis was based on six fictitious sites representing the real sites of the first hundred reactor units in the USA. Plant-specific risk studies using Reactor Safety Study methodology have since been carried out for many U.S. nuclear power stations.

A major re-evaluation study, the Reactor Risk Reference Study (1216), provides updated risk analyses for five representative U.S. nuclear power plants (Table 12.15), including the reference plants of the Reactor Safety Study. The distinctly different containment design for each of the plants was an important factor in their selection. The Industry Degraded Core Rulemaking Program (IDCOR) evaluated accident risks for four of these plants (1217).

Both studies used state-of-the-art methods as described in section 12.3.1. The analysis was limited to internal accident initiators. The Reactor Risk Reference Study explicitly considered aspects of uncertainty in the estimation of core damage frequency, the evaluation of containment behaviour, and the determination of source terms. Hence, the analysis produced a range of values in which the true value would lie. The IDCOR study was guided by the “best estimate” principle, aiming at realism in the choice of

TABLE 12.15. *Reference plant characteristics*

Plant	Type	Capacity MWel, gross	Grid connection	Manufacturer	Containment type
Surry	3-loop PWR	811	7/72	W	Subatmospheric
Zion	4-loop PRW	1085	6/73	W	Large, dry
Sequoyah	4-loop PWR	1183	7/80	W	Ice condenser
Peach Bottom	BWR/4	1098	2/74	GE	Mark I
Grand Gulf	BWR/6	1372	10/84	GE	Mark III

Source: *Reactor Risk Reference Document*, USNRC Report NUREG-1150, Draft, U.S. Nuclear Regulatory Commission, February 1987

data and models, and resulting in point estimates of core damage frequencies, containment failure probabilities and offsite consequences.

In both studies, the accident risk is related to some consequence measure: early fatalities, early injuries, latent cancer fatalities, population doses, and offsite costs. The risk is obtained by multiplying the frequency of each accident sequence per reactor year by the associated consequence, *averaged over the weather conditions* around the specific plant and summing over all accident sequences. Usually, the risk is determined by a few dominant sequences.

The ranges of risk for early and late fatalities from the Reactor Risk Reference Study are displayed in Figs. 12.21 and 12.22. For comparison, the corresponding results from the Reactor Safety Study and IDCOR are also shown. The risk ranges were obtained by a statistical sampling technique combining point estimates of the core damage frequency, containment failure probability and source terms within their uncertainty ranges.

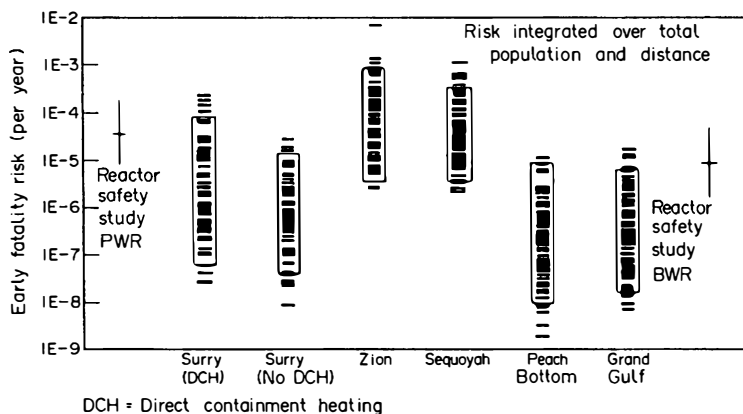


FIG. 12.21. Comparison of early fatality risks. From *Reactor Risk Reference Document*, USNRC Report NUREG-1150 Draft, February 1987

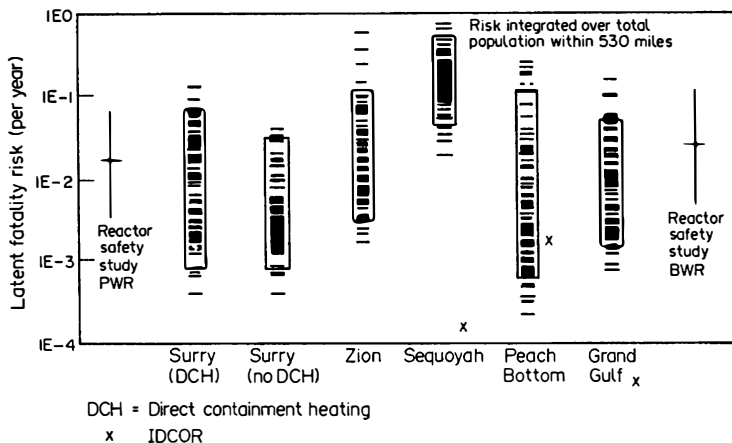


FIG. 12.22. Comparison of late fatality risks. From *Reactor Risk Reference Document*, USNRC Report NUREG-1150 Draft, February 1987

However, due to the lack of precise data, no significant information could be obtained about the mean risk and its variance.

It can be seen that the level of early fatality risk varies considerably from plant to plant. The relatively high fatality risk for the Sequoyah plant appears to mainly result from a relatively high core damage frequency. The high early fatality risk for Zion is due to a substantially higher population density around this plant. The lower early fatality risks for Peach Bottom and Grand Gulf are primarily the result of a significantly lower core damage frequency in the former case, and a low population density around the plant in the latter case.

The late fatality risks show less variability among the studied plants, as can be expected since late effects are predicted to occur over larger regions and are therefore less sensitive to site population characteristics. The late consequences are generally proportional to the total magnitude of the radioactive release and are rather insensitive to other source term characteristics. The long-term health effects are predicted to be received principally from the consumption of slightly contaminated foodstuffs.

The risk-dominant accident initiators and containment failure modes are summarized in Table 12.16. It can be seen that station blackout and early containment failure by overpressure are important for several of the studied plants. Failure of the component cooling system leading to reactor coolant pump seal LOCA is found to be a dominant contributor for two of the pressurized water reactor plants.

As seen from Figs. 12.21 and 12.22, the Reactor Safety Study results for Surry and Peach Bottom lie near the upper end of the Reactor Risk Reference Study risk ranges, particularly if direct containment heating is not a significant threat to early containment failure. The lower estimated risk in

TABLE 12.16. *Risk-important accident initiators and containment failure modes*

	Accident initiator	Containment failure mode
Surry	Station blackout	Early overpressure (direct containment heating)
Zion	Loss of component cooling (pipe rupture)	Early overpressure (direct containment heating)
Sequoyah	Loss of component cooling (pump failure)	Early overpressure (hydrogen combustion)
Peach Bottom	Station blackout (battery failure)	Early failure (drywell melt-through)
Grand Gulf	Station blackout (diesel-generator failure)	Failure by hydrogen combustion

Source: *Reactor Risk Reference Document*, USNRC Report NUREG-1150, Draft, U.S. Nuclear Regulatory Commission, February 1987

the updated study is primarily due to lower predicted core damage frequencies and source terms. This appears to be partly offset by the revised consequence model predicting larger effects (for similar releases).

The IDCOR results generally fall below the risk ranges of the Reactor Risk Reference Study. This is a result of considerable differences in the assessment of containment loads and the resulting source terms. In addition, IDCOR assumed that the whole of the nearby population participated in evacuation, while the Reactor Risk Reference Study assumed a 5% non-participation. This directly affects the early fatality risk estimation and partly explains why IDCOR predicted that no early fatalities would occur in the cases studied.

The risks and consequences in Figs. 12.21 and 12.22. represent mean values with respect to the weather conditions. The Reactor Safety Study used the exceedance frequency distribution method (see 12.3.1) to display the results, including the variability of consequences over a range of possible weather conditions. For comparison, this method was also illustrated in the Reactor Risk Reference Study. A sample display is shown in 12.23.

The Reactor Safety Study results shown in Fig. 12.23 have been modified to use actual Surry site data instead of the "generic" site data in the original study. The "high" and "low" curves correspond to the upper and lower ends of the risk ranges in Figs. 12.21 and 12.22 (including the effect of direct containment heating).

The comparison shows that the Reactor Safety Study estimates for early fatalities fall within Reactor Risk Reference Study range for a small number of fatalities, but that the Reactor Safety Study data show a higher likelihood of a large number of early fatalities. For estimates of late fatalities the Reactor Safety Study estimates lie consistently somewhat below the upper curve of the re-evaluation study. This confirms the conclusion that the Reactor Safety Study results are near the upper end of the Reactor Risk Reference Study risk range.

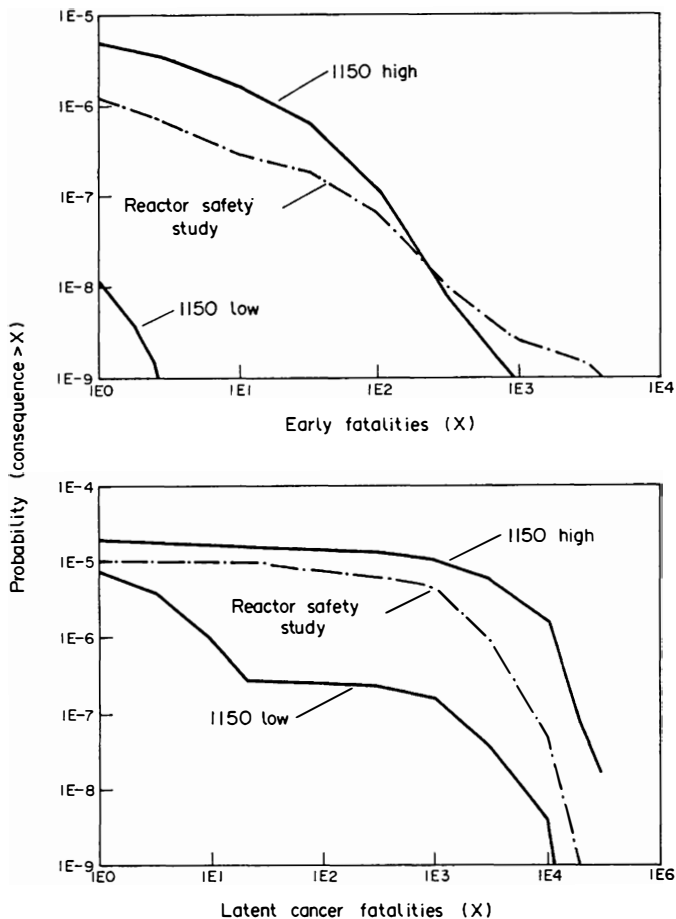


FIG. 12.23. Comparison of Reactor Safety Study and Reactor Risk Reference Study exceedance frequency distributions for the Surry plant. From *Reactor Risk Reference Document*, USNRC Report NUREG-1150 Draft, February 1987

12.4 Risk Assessment

This section discusses the concept of risk and its application for the comparison of societal risks.

12.4.1 The concept of risk

The Reactor Safety Study established the concept of risk as the product of an accidental release and its associated consequence. This has caused some confusion since the word "risk" is used in everyday speech to denote

both a hazardous event and the likelihood of such an event. In this book, "risk" has occasionally been used in the latter sense.

The concept of risk originates from classical decision theory dealing with rational choice between different courses of action. The theory attempts to structure the options and their possible consequences as well as to quantify their probability and value. The values of the consequences are multiplied by the associated probabilities of occurrence. The sum of these products is the *expectation value* of the particular option. A rational approach would be to choose the option with the highest expectation value.

The method is illustrated in Fig. 12.24. H_1 , H_2 and H_3 designate different options. The branches represent the corresponding consequences which can have positive or negative "values" in the example given. The numbers above the branches indicate the estimated probabilities. H_3 has the highest expectation value and should therefore be chosen according to the principle of maximizing the expectation value.

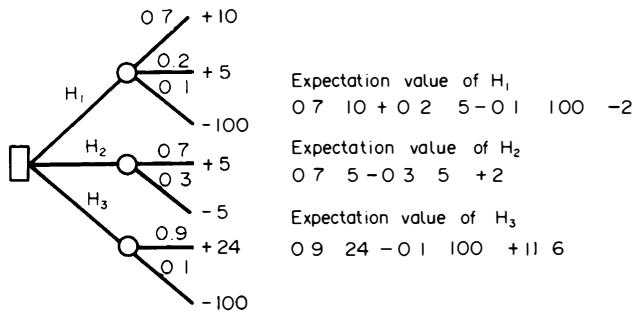


FIG. 12.24. Decision alternatives and expectation values. From Swedish Department of Industry, *Risk Evaluation*, Report DsI 1978:15

If this model is transferred to accident risk analysis, H_1 , H_2 and H_3 may designate initiating events and the branches different release sequences. The quantitative measure of the damage to life, health or property corresponds to the "value" of the consequence. The expectation value is the "risk" as defined in the Reactor Safety Study. Probabilistic risk analysis is the overall term for the method.

Probabilistic risk analysis of severe accidents involves several problems. The analysis is concerned with *extreme events*, extreme both in terms of the phenomena involved and in terms of the level of probability of the events themselves. The significance is uncertain for the very low probabilities of events which have never occurred in practice. However, it is generally possible to break down a sequence of events into basic events for which the probabilities can be estimated on the basis of experience. In some cases, when empirical data are lacking, educated guesses are required. The result-

ing total probability becomes a mixture of objectively verifiable and subjectively estimated partial probabilities.

When assessing the results of risk analysis, it must be kept in mind that the numerical values are estimates which are subject to *uncertainty*. Some of the uncertainty stems from the very nature of the theory, which deals with probabilities. Other uncertainties arise from the data base for quantifying the fault trees and from the calculational models for describing the accident progression. Problems arise when combining the uncertainties since some of the partial probabilities may not be strictly verifiable. The resulting uncertainties must be interpreted as “subjective confidence intervals” (1212).

A fundamental uncertainty lies in the *incompleteness* of the analysis. However, because of the systematic approach and the increasing operating experience, it is unlikely that any major failure modes or sequences would be overlooked. Neither is it probable that the totality of omitted cases would substantially increase the risk.

A different problem arises from the attitude of the general public to accidents with large consequences. Compare an event which statistically occurs once a year and involves an average of 1 fatality per event with an event expected to occur once in 10,000 years leading to 10,000 fatalities. both events have the same expectation value, namely 1 fatality per year, but the latter will obviously be considered the more frightening of the two. This phenomenon is called *risk aversion*. Risk aversion means that the mere possibility of a large accident, regardless of how low the probability may be, is a large enough deterrent against accepting the risk. In decision theory, this attitude is represented by the “minimax” principle. This principle leads to choosing the option for which the worst consequence offers the best possible outcome. In Fig. 12.24, the minimax principle leads to the choice of H₂.

12.4.2 Risk comparison

Great caution must be exercised when comparing reactor accident risks with other societal risks because of the one-dimensional character of the risk concept. Probabilities and consequences should preferably be presented separately. This has also been done in most risk analyses carried out so far, where the normal form of presentation is the exceedance frequency distribution of consequences (see Figs. 12.14–12.17). Diagrams of this type illustrate both the “worst case” and the risk, i.e. the expectation value of the consequence, which is equal to the area under the curve.

The *individual* risk for a certain event is obtained by dividing the total risk by the population around the nuclear power plant. Figure 12.25, which is reproduced from the German Risk Study (1212), shows the expectation value for early and late effects *per caput* as a function of the distance from

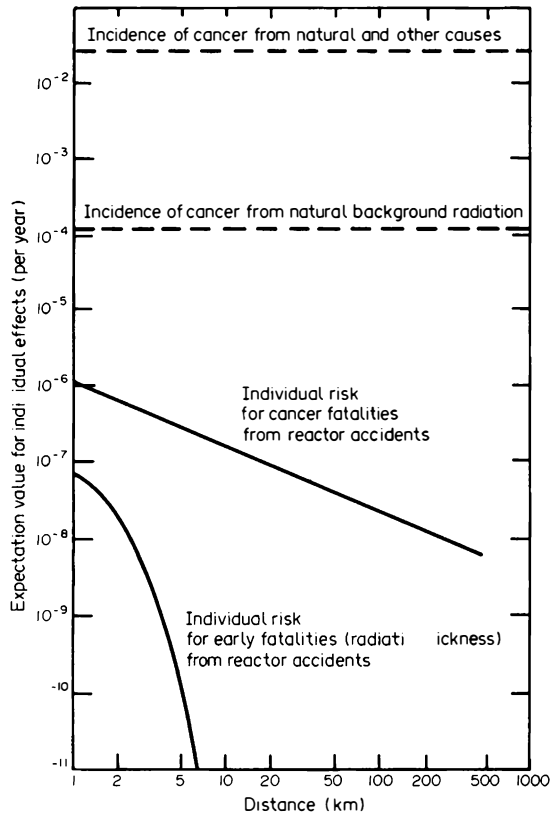


FIG. 12.25. Expectation value for individual health effects from a reactor accident versus the distance from the nuclear power plant for conditions in West Germany. From the *German Risk Study. Nuclear Power Plants*, Verlag TÜV Rheinland, 1980

the nuclear power plant. The curves refer to the total individual risk from all release categories for the population distribution in the vicinity of a typical German reactor site. It can be seen that the risk for early effects decreases rapidly with distance, while the risk for late effects is spread over a considerable distance and affects regions beyond the frontiers of the country. For purposes of comparison, the expectation values for cancer fatalities from the natural background radiation and from all natural and societal causes are also shown.

To set perspectives, the expectation value for the *collective* dose, given that an accident has occurred, is approximately of the same order of magnitude as the annual collective dose from various natural and other radiation sources in Sweden (Table 12.17). The *total* number of cancer fatalities within a 30-year period starting some 10 years after the accident, will therefore be

TABLE 12.17. *Collective doses and health effects from radiation exposures in Sweden*

Radiation source	Population affected	Annual collective dose in the early 1980s (manSv)	Total number of fatalities or serious hereditary effects from one year's dosage
Cosmic radiation	8 million	2400	48
Naturally occurring radioactive substances in the body	8 million	3500	70
Natural gamma radiation from the ground	8 million	800	16
Dwellings, radon daughters	8 million	57,000	1140
Dwellings, gamma radiation	8 million	4000	80
Mine and underground workers	5000	75	1.5
Dental X-ray, patients	8 million	600	12
Health service X-ray, patients	8 million	5000	100
Isotope examinations, patients	100,000	580	12
Nuclear weapons	8 million	100	2
Nuclear power, normal operation, personnel	3000	15	0.3
Nuclear power, normal operation, environmental	8 million	0.3	0.006
Other		20	0.4
Total		about 74,000	about 1500

Source: State Public Investigation, *Cancer. Causes, Prevention etc.*, SOU 1984:67, Stockholm, 1984

about equal to the *annual* number of fatalities (in Sweden) from natural and other radiation sources. It will not be possible to observe the increase of the cancer frequency resulting from a reactor accident, because of the high cancer frequency from other causes than radiation—a total of about 20,000 fatalities per year in the beginning of the 1980s—and the random variation of this frequency.

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13

Operating Experience

During the 1970s there was a rapid increase in the number of light water reactors put into operation. The operating experience shows that it has been possible to attain and maintain a high level of safety. The release of radionuclides during normal operation has remained far below permissible values. Although incidents and accidents have occurred, the offsite releases have been negligible in all cases.

This chapter reviews statistical data on normal operation and safety-related events for both pressurized and boiling water reactors with emphasis on the experience in the United States and Sweden. Some selected events, including the Three Mile Island accident, as well as methods for the analysis and feedback of information are described. The chapter concludes with a review of the Chernobyl accident and its implications for light water reactor safety.

13.1 Plant Availability

For economic reasons, it is important that a nuclear power plant be utilized for as large a part of the time as possible, i.e. the availability should be high. The *plant load factor* is the ratio of the delivered average power during a certain time interval and the maximum power of the plant. Since a light water reactor needs to be shut down for refuelling about once a year, it is not possible to reach a 100% load factor on a long-term basis.

Inspection and servicing of plant components are carried out in conjunction with refuelling. These *planned outages* normally last for 4–8 weeks. In Sweden, they are scheduled for the summer when the electricity demand is at its lowest. The planned outages reduce the maximum possible load factor to 85–90%. If a plant in spite of this shows a load factor of more than 90% in a single operating year, it is due to the fact that a reactor may be operated for more than a year, for example 18 months, without refuelling, if the fuel is given a suitable enrichment.

The load factor alone is not sufficient for assessing the availability. A plant can be operated at reduced capacity for some period of time if the load demand is low. Another way in which the load factor is reduced is by stretch-out operation at the end of an operating period when the fuel is

depleted. Plant load factor data should therefore be supplemented with additional information on plant operation. The *availability factor* is often used, i.e. the time (as a percentage of the total time) the generator has been connected to the grid, regardless of the output. While the load factor is mainly of importance for assessing plant economics, the availability factor is a measure of plant reliability.

The availability factor is affected by planned outages for refuelling, maintenance and repair as well as by forced outages caused by component failure. The statistics for a typical operating year are shown in Table 13.1.

TABLE 13.1 *Operating statistics for the Oskarshamn Nuclear Power Plant, Unit I, calendar year 1982*

Planned outage	1402 hr = 16%
Unplanned outage	386 hr = 4.4%
Operating time	6972 hr = 79.6%
Plant load factor	76.2%

The forced outages were largely caused by turbine and generator system failures.

The availability of the Swedish nuclear power plants during 1981–83 is shown in Table 13.2. The boiling water reactors had a consistently high availability. The average values for three years are a load factor of 75.1% and an availability factor of 83.9%.

During 1982, unit 2 of the Barsebäck power station attained a load factor of 92.2% and an availability factor of 97.8%. The unit was in operation for

TABLE 13.2. *Availability of Swedish nuclear power plants during 1981 to 1983*

Reactor unit	Plant load factor			Availability factor		
	1981	1982	1983	1981	1982	1983
Barsebäck 1	82.8	79.2	80.2	87.9	84.5	88.1
2	76.2	92.2	74.9	86.6	97.8	84.3
Forsmark 1	76.9	70.4	75.5	83.3	81.4	92.4
2	72.2	67.4	72.8	90.1	69.4	89.9
Oskarshamn I	74.9	76.2	81.7	80.9	79.5	87.9
II	76.8	85.1	79.7	84.8	90.2	87.9
Ringhals 1	61.8	71.3	50.0 ^a	71.7	81.8	61.3 ^a
2	58.4	64.9	56.5	70.9	67.6	69.7
3	26.8 ^b	15.6 ^b	36.4 ^c	29.5 ^b	42.0 ^b	67.2
Mean value BWR	74.5	77.4	73.5	83.6	83.5	84.5

^aInspection and exchange of tubes in secondary process systems after cracking indications.

^bOperation at reduced power (40%) and during limited time, due to vibration problems and modification of steam generators.

^cOperation at reduced power during the first half-year, and extended revision period.

532 of 544 days during an 18-month operating period from September 1981 to March 1983.

Information on the operation of nuclear power plants in the West is published on a regular basis. Figure 13.1 shows load factors during 1983 for all light water reactors with a capacity greater than 100 MWel (1301). The average value is 64% for the pressurized water reactors (101 units) and 61% for the boiling water reactors (56 units). The Swedish boiling water reactors had a significantly higher plant load factor than average, while that of the pressurized water reactors was somewhat lower than average.

At the end of 1983 the total operating time for all light water reactors in the West with a capacity greater than 100 MWel amounted to 1210 reactor years. A closer analysis of the data reveals a slight upward trend for the load factor with operating time. Attempts to correlate the load factor and the reactor size indicate no dependency for pressurized water reactors and slight downward trend with increasing size for boiling water reactors (1301). However, the statistical uncertainty is considerable since there are only a few boiling water reactors in the high capacity range (1100–1300 MWel).

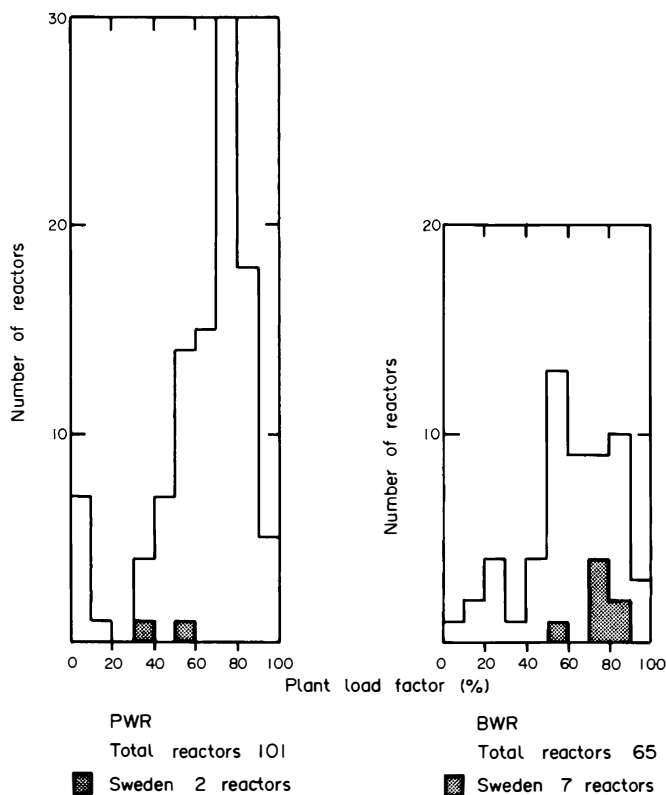


FIG. 13.1. Plant load factors during 1983. All LWRs > 100 MWel in the West

The distribution of the cumulated load factor (weighted with the operating time) is shown in Fig. 13.2 and the availability factor in Fig. 13.3. (1302). On the whole, the pressurized water reactors show somewhat better results than the boiling water reactors. The high availability of the Swedish boiling water reactors is also confirmed in the cumulated data.

13.2 Activity Release and Occupational Exposure

The release of radioactive substances is continually monitored in the ventilation stack and before discharging waste water through the cooling water channels into the sea. In Sweden, data on releases to air and water are submitted on a regular basis to the National Institute for Radiation Protection where they are compiled and published (1303). International reports

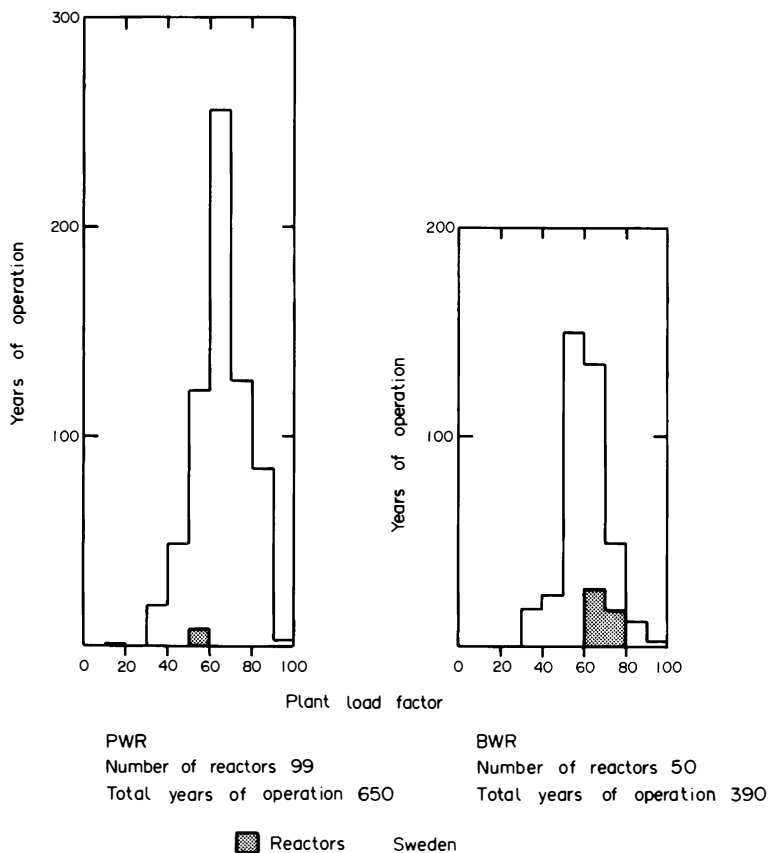


FIG. 13.2. Cumulated load factors up to and including 1982. All LWRs > 100 MWe in the West

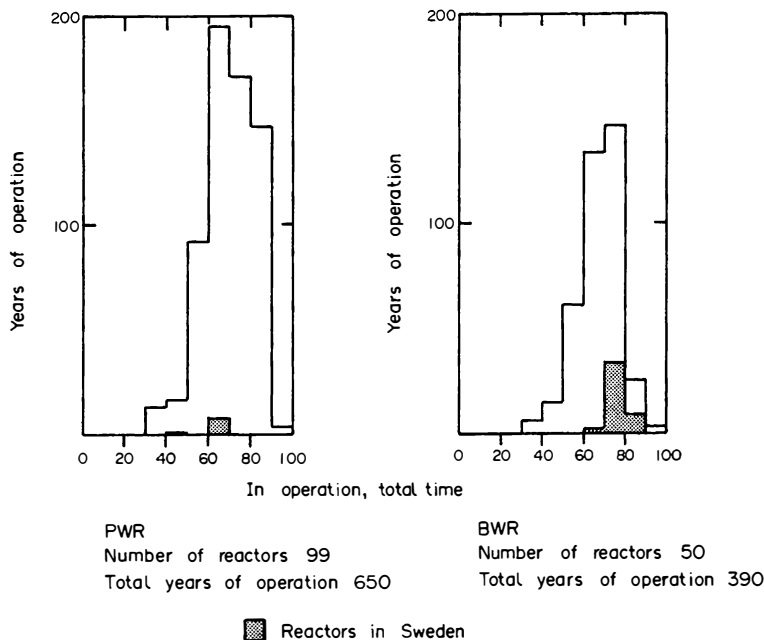


FIG. 13.3. Cumulated availability factors up to and including 1982. All LWRs > 100 MWe in the West

are compiled by the United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR (1304).

Swedish regulations prescribe that nuclear power plants shall be designed so that any releases to the environment during normal operation will result in a dose equivalent less than 0.1 millisievert (mSv) per year to nearby residents (cf 6.6.1). This value is very low in comparison with other dose levels (Table 13.3).

It is useful to express the measured release in relation to the *reference dose* level. The sum of the releases to air and water from 1981 to 1983 by Swedish nuclear plants is shown in Table 13.4 (cf Tables 6.11 and 6.13).

It can be seen that the releases fall well below the design specifications,

TABLE 13.3. Comparison of dose levels

	mSv/year
Highest permissible dose for radiological workers	50
Average dose from radon daughters in Swedish dwellings	approx 5
ICRP's limit for individual dose	5
Natural background radiation (excl radon)	approx 1
Design criterion for nuclear power plants	0.1

TABLE 13.4. Releases from Swedish nuclear power plants

	Fraction of reference release		
	1981	1982	1983
Barsebäck	0.006	0.012	0.003
Forsmark	< 0.0001	0.001	0.002
Oskarshamn	0.21	0.080	0.041
Ringhals	0.39	0.42	0.047

and that they have decreased. The downward trend is also confirmed in a larger series of measurements at the Oskarshamn nuclear power station (Fig. 13.4), which has the oldest Swedish reactor unit OI in addition to OII and OIII.

The releases are correlated to the fuel quality and the reactor operating mode. Power changes, for example during start-up, cause stresses in the fuel which can result in damage and subsequent leakage of small quantities of fission products into the reactor coolant. For example, shortly before the 1975 refuelling outage in OI, a test was conducted to determine whether the power ascension at start-up could be faster. The test resulted in several cases of fuel damage and an increased susceptibility to fuel leakage in the

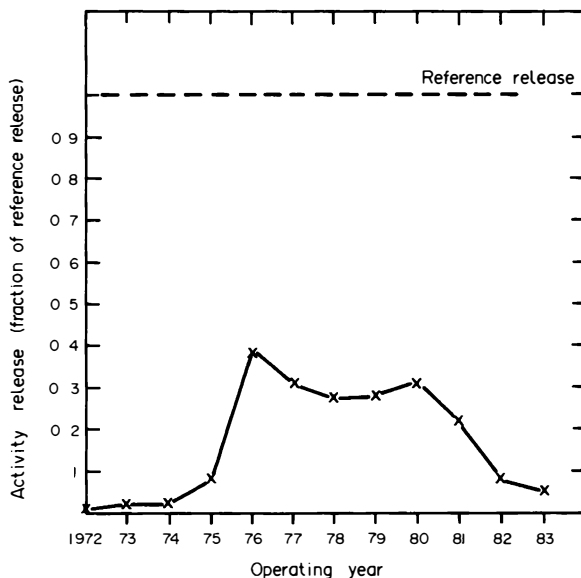


FIG. 13.4. Activity releases to air and water at Oskarshamn nuclear power station, units 1 and 2

initial core during subsequent years. These conditions are reflected in Fig. 13.4. Since the last fuel assemblies of the initial core were replaced in 1980, only a few leaky assemblies have appeared.

In addition to the offsite releases, the radiation doses received by the plant workers are monitored. The entire staff uses dosimeters which measure the individual dose. The registered dose is regularly reported to the radiation protection agency, who also sets the dose limits. The upper limit for individuals engaged in radiological work is 50 mSv/year (cf Table 13.3). The collective dose at a nuclear power station, i.e. the sum of the products of the number of persons with a measurable exposure (> 0.1 mSv) and the corresponding individual dose, is a measure of the total occupational exposure.

In recent years the average individual occupational dose at Oskarshamn has been about 2 mSv/year, i.e. about 4% of the upper limit. The largest doses are normally received during refuelling outages and mainly by contract workers. The occupational doses are noted in a central dose register for the country and totalled for each individual, regardless of where the dose was obtained. As a result, the dose to contract workers, who normally move from plant to plant, can be monitored.

The occupational collective doses at the Swedish nuclear power plants are low in an international comparison (Fig. 13.5). This is the result of well-planned plant layout, ample radiation shielding, suitable choice of water chemistry and materials, effective procedures and instructions and careful planning of maintenance and repair work.

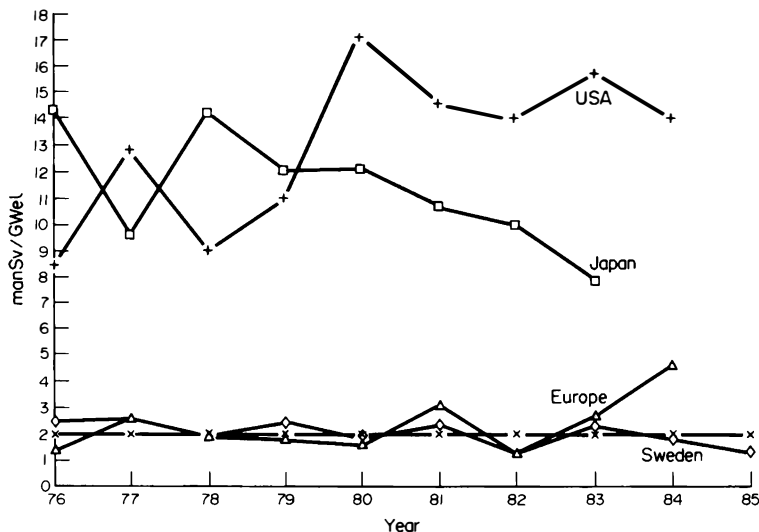


FIG. 13.5. Occupational collective doses – BWR worldwide. From P Drake, How Sweden Achieved 15 Years of Low Occupational Doses, *Nucl. Europe*, December 1986

13.3 Safety-related Events

Failures can occur in nuclear power plants as in any complex technical system and result in more or less extended outage. In cases where the safety of the plant is involved, the term safety-related event is used. All safety-related events must be reported to the supervisory agencies according to the operating rules for the plant. Examples of events to be reported include:

- Exceedance of limit values of plant variables essential for safety.
- Severe damage to fuel and systems pressurized from the reactor.
- Unplanned or uncontrolled large releases of radioactive substances.
- External events threatening the safe operation of the plant.
- Component failure or manoeuvring errors which prevent or could have prevented the intended performance of safety-related systems.

All outages must be reported and the reason for the outage stated, e.g. reactor scram, turbine trip.

In the event of uncontrolled releases, special reference values apply for the offsite activity or dose levels. These values are established by the radiation protection agency on a case-by-case basis.

13.3.1 U.S. operating experience

According to U.S. safety regulations, a safety-related event must be orally reported to the Nuclear Regulatory Commission within 24 hours of its occurrence and a written Licensee Event Report (LER) submitted within 2 weeks. Data from the LERs are stored in a central computer for statistical processing. Reports on the compiled data are published annually. By providing information on the frequency of component failures, systems affected, causes, etc., these reports form a basis for safety improvement.

In a typical year (1980), some 1500 events for boiling water reactors and some 1700 events for pressurized water reactors in commercial operation were reported (Table 13.5). This amounts to an average of sixty-two reports per BWR and forty-two reports per PWR. During 1980, five PWRs were in a state of power ascension (not included in Table 13.5). The number of reports from these five reactors was eighty-two on average.

The affected systems and components are shown in Tables 13.6 and 13.7. The auxiliary cooling systems (high-head and low-head injection systems, shutdown cooling system) are responsible for most of the BWR reports, whereas most of the PWR reports concern secondary systems (steam generators, feedwater system). The most frequent components in the reports are valves and instruments. The most common deficiencies are leakage and set point drift. Faults were often detected and corrected in connection with performance testing and maintenance. Of the reported events, only twenty-

TABLE 13.5. *Reported safety-related events in U.S. light water reactors during 1980*

	BWR	PWR
Number of reactors in operation during year	25	40
Number of reports (LER)	1547	1683
LER per reactor year	62	42

Source: K E McCormack, R B Gallaher, Review of Safety-Related Events at Nuclear Power Plants in 1980, *Nucl. Safety*, Vol 23, No 3, 1982

TABLE 13.6. *Systems involved in safety-related events in U.S. light water reactors in 1980*

System	Number of reports (percentage)	
	BWR	PWR
Reactor containment	13	11
Main cooling system	9	14
Secondary systems	—	22
Steam system	9	5
Auxiliary cooling systems	27	13
Power supply systems	12	17
Monitoring and control systems	11	13
Service systems	12	13
Other equipment	6	7

N.B. The percentage sum exceeds 100, since more than one system are involved in some reports.

Source: K E McCormack, R B Gallaher, *loc. cit.*

TABLE 13.7. *Components involved in safety-related events in U.S. light water reactors in 1980*

Components	Percent of reports	
	BWR	PWR
Valves	26	21
Pumps	10	7
Pipes and connections	9	11
Switches	18	8
Circuit breakers	4	4
Pressure transmitters	9	4
Level transmitters	6	4
Radiation instruments	4	6

Source: K E McCormack, R B Gallaher, *loc. cit.*

nine (1.9%) for the BWRs and fifty-two (2.4%) for the PWRs resulted in reactor shutdown.

For obvious reasons, the number of unanticipated events is greatest at the beginning of a reactor's lifetime, especially during power ascension. Figure 13.6 shows the number of LERs per reactor and year (1980) as a function of the reactor age. The number of reported events in the oldest reactors (with up to 20 years' operating time) is only about one-third of that during the initial years of operation.

Figure 13.7 presents the same data versus plant size, expressed as net electrical output. The number of LERs per reactor seems to increase in proportion to the increase in reactor size (except for large BWRs). However, this trend is partly deceptive, because the large reactors have a lower average age. Factors other than age and size may be important, i.e. the manufacturer (for PWRs), "vintage" and management of operations.

The results in Figs. 13.6 and 13.7 must therefore be evaluated with caution.

13.3.2 Swedish experience

In Sweden, the reporting of safety-related events is regulated in the Technical Specifications for reactor operation (see 7.2.6). A distinction is made between an *abnormal event* which denotes an unanticipated plant condition

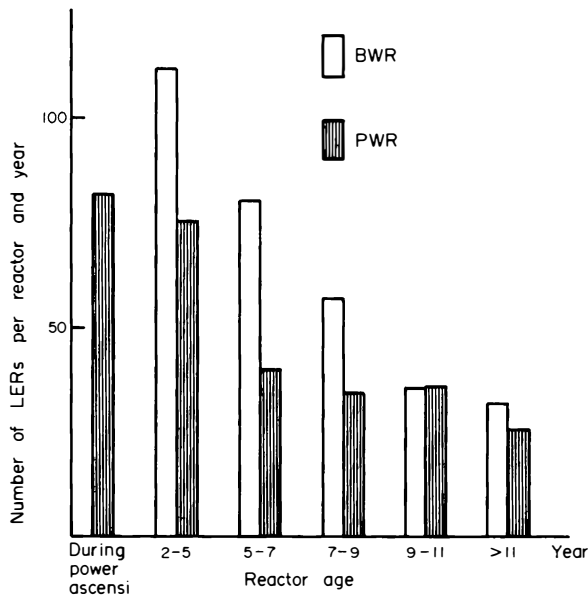


FIG. 13.6. Number of LERs per reactor and year (1980) versus reactor age

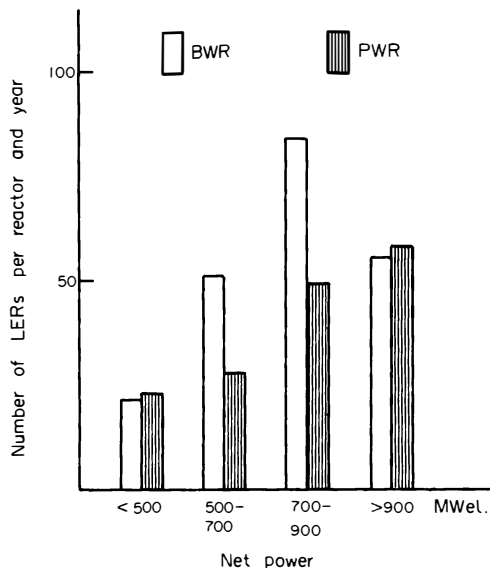


FIG. 13.7. Number of LERs per reactor and year (1980) versus reactor capacity

which is so serious that continued operation is not permitted without a special safety review, and a *reportable occurrence* (RO) of importance to safety. In case of an abnormal event, the Nuclear Power Inspectorate (SKI) must be notified within 24 hours and a final report be submitted within 10 days. A reportable occurrence must be reported to SKI within 30 days if the conditions so require.

SKI publishes a summary of the received reports every six months (1306). The safety-related events are grouped into four categories (category (1) and (2) relate to unanticipated events of no importance to safety):

- (3) A component or system failure which, because of available back-up, does not require immediate shutdown of the reactor according to the Technical Specifications.
- (4) A component or system failure which, according to the Technical Specifications, requires the immediate shutdown of the reactor or is deemed by SKI to be of equivalent severity.
- (5) A crack or rupture of a tube (diameter < 50 mm) in a system which is pressurized from the reactor and inside the reactor containment. (For PWR also within the secondary system inside the containment.)
- (6) Other more extensive events.

For each event, data are reported on the operating conditions at the time of discovery, the manner of discovery, symptoms, effect on operations, effect on components, type of component, action adopted or planned, direct

cause and possible primary cause. Each item of information is given a code number for computer processing and evaluation.

The number of safety-related events reported during the three-year period from 1980 to 1982 is presented in Table 13.8.

It can be seen that 95% of the events belong to category (3), not requiring immediate reactor shutdown. Only one category (5) and no category (6) event occurred during the three years covered. No abnormal event in the sense of the Technical Specifications occurred. The category (5) event concerned a tube leak in one of Ringhals 3's steam generators in October 1981.

Tables 13.9 and 13.10 indicate the systems and components involved in the reported events. The power supply system accounts for most of the BWR events, while the reactor cooling system, which includes the steam generators, is dominant in the PWR events. Valves appear to be the most vulnerable component, although control equipment and pumps and exhaust fans recur in many reports.

TABLE 13.8. *Reported safety-related events in Swedish light water reactors from 1980 to 1982*

	BWR	PWR
Number of operating years	20	6.5
Number of reports (RO)	592	123
Number of RO per reactor	30	19
Category (3)	567	115
(4)	25	7
(5)	0	1
(6)	0	0

TABLE 13.9. *Systems involved in safety-related events in Swedish reactors 1980-2*

System	Percent of reports	
	BWR	PWR
Reactor containment	3	2
Reactor	6	0
Reactor coolant system ^a	23	45
Turbine/generator set	9	7
Monitoring and control system	10	16
Power supply system	27	14
Service system	20	15
Other equipment	1	2

^aIncludes main coolant system, secondary system (PWR) and auxiliary cooling systems.

TABLE 13.10. *Components involved in safety-related events in Swedish reactors 1980-2*

Component	Percent of reports	
	BWR	PWR
Pressure vessel	1	3
Heat exchangers	3	11
Pipes and connections	9	7
Valves	20	23
Pumps, fans	14	20
Motors, generators	8	3
Control equipment	19	18
Switchgear	7	2
Cables	3	3
Other components	15	10

13.3.3 Reactor scram

Reactor scram is automatically initiated on receipt of a signal from sensors indicating abnormal values of essential primary system variables (cf 8.1.1.). During a scram transient, many systems and components are subjected to thermal and hydraulic stress. The transient can be aggravated if essential safety functions fail (cf. Fig. 10.12). Therefore, a low scram frequency is desirable, while at the same time a very high reliability is required of the actuating safety chains. The desire for a low scram frequency must not make the operator hesitate to initiate scram manually if necessary.

Experience shows that the scram frequency, especially for the older plants, is relatively high in the beginning of the operating history, and falls off later on. Figure 13.8 presents the average values for the scram frequencies per reactor from sixty U.S. light water reactors from 1978 to 1983. The falling trend is evident, as is the fact that the frequency is lower than average in plants which have been in operation for more than 3 years. The number of manual scrams is about 15% of the total number.

A closer analysis reveals no significant differences between boiling water and pressurized water reactors. In PWRs, events resulting in scram often spring from problems with the feedwater control system, while turbine trip is a common precursor to scram in BWRs. About two-thirds of the scrams are caused by equipment failure, while manoeuvring errors account for about 12%. This may be due to the fact that the feedwater and turbine control systems are not really safety systems and are designed with less emphasis on redundancy.

The scram data for Swedish reactors largely confirm U.S. experience (Fig. 13.9). The graph shows a decline in the scram frequency with increasing

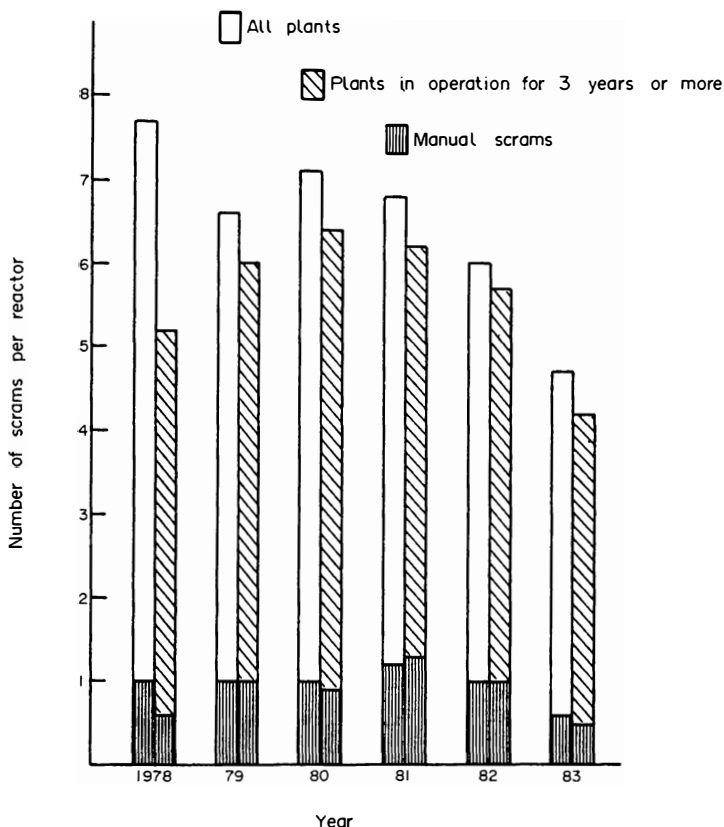


FIG. 13.8. Number of scrams per reactor and operating year in U.S. plants 1978–83. From *Reactor Trips in U.S. Nuclear Power Plants*, Institute of Nuclear Power Operations, 1984

operating time and a substantially lower frequency for second and third generation plants than for first generation plants. The reason for this trend is mainly attributed to improved operating and maintenance procedures as well as improvements in design and training.

The high scram frequency during the first years in first generation boiling water reactors was mainly due to problems with feedwater preheating and control. These problems were eliminated by design improvements with an attendant reduction of the scram frequency. During the first years of operation, many scrams in the pressurized water reactor Ringhals 2 were caused by problems with the manual control of the water level on the steam generators' secondary side at low power. Since automatic feedwater control was implemented in 1979, the scram frequency has decreased considerably.

Operating experience shows that it has largely been possible to eliminate human error as a cause of scram in Swedish nuclear power plants. Loss of

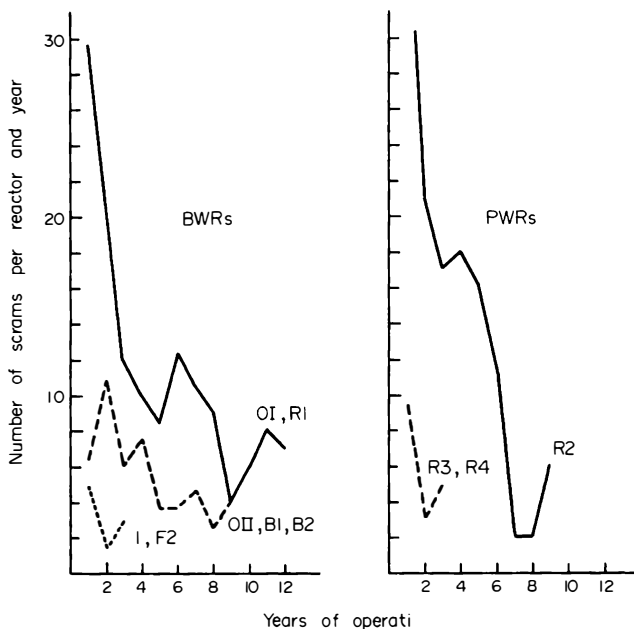


FIG. 13.9. Number of scrams per reactor and operating year in Swedish plants. From *Experience in Plant Transients. The Swedish RKS Program*, Report RKS 83-11, Nuclear Safety Board of the Swedish Utilities, 1983

offsite power has proved to be a considerable contributor if the switch-over to house load operation also fails. During the nationwide blackout on 27 December 1983, all nuclear power units were disconnected from the grid. Only Forsmark 1 succeeded in switching over to house load operation while the others tripped. However, at the three affected sites (Barsebäck, Oskarshamn and Ringhals) all emergency diesel generators started automatically and operated satisfactorily. Also, the gas turbines in Barsebäck and all but one in Oskarshamn were started automatically and operated well. Most of the main grids were recovered in about an hour.

13.4 Significant Events

Thousands of safety-related events at nuclear power plants are reported each year. The reports cover a broad spectrum of events and circumstances. More than 95% of the cases represent failures not directly affecting safety, during which plant operation continued without interruption. In a few cases a safety function failed or a safety system on standby was not available. Only in one case during some 3000 operating years (January 1988) did severe core damage occur.

13.4.1 Occurrences in Swedish plants

In the 107 operating years accumulated in Sweden (January 1988), only one abnormal event, according to the definition of the Technical Specifications (cf 13.3.2), has occurred, namely in *Ringhals 2* on 16 June 1979. In conjunction with start-up, when the reactor was on hot standby, a leak in a temperature detector return line connected to the primary system was observed via TV cameras in the reactor containment. In order to minimize the amount of water escaping, the reactor operator attempted to lower reactor pressure as soon as possible. The low-pressure signal for automatic start-up of the safety injection system was therefore blocked. The pressure, temperature and flow in the primary system were carefully controlled to avoid boiling. However, the operator forgot to control the water level in the pressurizer. As a result, for 20–25 minutes, the pressurizer water level dropped below the set point and probably somewhat below the top of the reactor vessel. However, the risk of core uncover and heat-up was minimal because of the low level of decay heat and because the coolant flow was maintained by a main coolant pump. When the low water level in the pressurizer was discovered, water was supplied by the charging pumps of the volume control system. Normal cooling and shutdown of the reactor then followed. In all, about 57 m³ of water leaked out of the primary system.

The leakage was caused by a faulty stuffing-box. Since then, all flanges which might result in leakage in pipes connected to the primary system have been redesigned and seal-welded. Blocking the safety injection system was in violation of the Technical Specifications. The required rapid pressure decrease could have been achieved in other ways. As a result of the incident, the instructions in Technical Specifications were modified and the maintenance procedures reviewed.

On 24 July 1987 an incident occurred at the Oskarshamn III BWR plant during the approach to start-up after annual refuelling and maintenance. Due to a combination of administrative and human error, a routine criticality test was conducted with the hydraulic scram system disconnected. In the test, two to three of the reactor's 150 control rods were withdrawn to achieve local criticality in order to check the shutdown margin. The test was repeated three times before the operator discovered that the scram system was blocked off, in violation of the Technical Specifications. While no fuel damage occurred and the electrical system for fine-motion insertion of the control rods remained operable during the tests, the event was considered serious by the Nuclear Safety Inspectorate. A review of the safety and test procedures at low power was required for all Swedish plants.

13.4.2 Occurrences in U.S. plants

In the USA, several events have occurred which have also attracted considerable attention in the mass media. The most discussed event—and the only event resulting in severe core damage—occurred in March 1979 at the Three Mile Island power plant. Table 13.11 is a selection of safety-related events up to and including 1986, in chronological order.

Several events have been initiated by disturbances in the feedwater supply. The reactors are designed to cope with such disturbances, but if an auxiliary system fails in addition, temporary DNB (departure from nucleate boiling) may result. However, if the primary system integrity is retained, there will be no abnormal release to the reactor containment and therefore no abnormal release to the environment.

Certain events can be characterized as small LOCA, e.g. the failure of a pressure relief valve to reclose, or seal leakage in a main cooling pump. If the isolation valves close and containment integrity is maintained, there will be no release to the environment. However, for PWR steam generator tube rupture, an increased offsite release can result when radioactive steam is discharged through the steam line safety valves before the reactor pressure has been decreased and the affected steam generator isolated. For severe core damage to occur, as in Three Mile Island, a combination of several failures and errors is required.

13.5 The Three Mile Island Accident

On 28 March 1979 the most severe accident so far in a light water reactor power plant occurred. Loss of feedwater in Three Mile Island Unit 2 (TMI-2) resulted in a transient which, through a series of unfortunate circumstances, led to severe core damage and large fission product release to the reactor containment. Some of the radioactive substances leaked into the environment by various routes.

13.5.1 The reactor

The Three Mile Island nuclear power plant is located on an island in the Susquehanna river near Middletown and Harrisburg, Pennsylvania. Both units have identical Babcock & Wilcox pressurized water reactors with a 900 MW_e capacity. TMI-1 was taken into operation in 1974, while TMI-2 had only been in operation for about 3 months when the accident occurred. The reactor was operating at 97% full power with a thermal output of 2734 MW_{th}. TMI-1 was shut down for refuelling. Each reactor has two main coolant loops with two pumps and one steam generator in each loop. A unique feature of the Babcock & Wilcox design is the once-through

steam generator which contains relatively little cooling water in reserve if feedwater supply should fail.

The reactor pressure is controlled in the usual way by a pressurizer which is connected to one of the two outlet nozzles of the reactor vessel (Fig. 13.10). The pressurizer normally holds about 23 m³ water and 20 m³ steam above the water surface. The steam pressure and thus the coolant pressure in the primary system is controlled by heating and cooling the water in the pressurizer with immersion heaters and cold water spraying (cf Fig. 5.6). The pressurizer is equipped with two safety valves and a pressure relief valve with an electrically operated control valve and a block valve. A pipeline leads from the pressure relief valves to a pressure relief tank in the bottom of the containment.

The emergency core cooling system consists of a high-head injection system which during normal operation functions as the chemical and volume control system and also supplies the main coolant pumps with salt water. There is also an accumulator system driven by high-pressure nitrogen, and a low-head injection system which normally functions as the residual heat removal system. The high-head injection system draws borated water from a storage tank. Gas is pumped from the volume control tank via decay vessels and filters to the stack. The radioactive water is pumped from the containment sump to a waste storage tank in the auxiliary building.

13.5.2 The accident sequence

At the time of the initiating event, maintenance work was being carried out on an ion-exchange system for feedwater polishing. At about 04.00 hours on 28 March 1979 all the feedwater pumps and turbines tripped, thus interrupting heat transport from the primary system. Since disturbances in the feedwater supply are not uncommon, auxiliary feedwater pumps are provided to replace the main feedwater pumps when required. There are three such pumps in TMI-2, two electrically operated pumps and one operated by a steam turbine (so that at least one pump will be operable, even for total loss of electric power). Although all three pumps started automatically as intended, the pumps take about 15 seconds to reach normal operating pressure. Meanwhile, the temperature and pressure in the primary system had increased, initiating scram shortly after the opening of the pressurizer relief valves. Up to this point, the sequence had taken place in agreement with the design specifications.

Unfortunately, two problems had arisen at this time, which were not known to the operators. The first was related to the two block valves in the auxiliary feedwater pump pressure lines, which are normally used during maintenance work. These valves must always be kept open during plant operation, and at most only one valve at a time may be closed for short periods. However, contrary to the specifications, both valves had been inad-

TABLE 13.11. Selected significant events in U.S. nuclear power plants

Date	Reactor	Event	Description
75-03-22	Browns Ferry-1 BWR, 1065 MWel Commissioned 1974	Cable fire	A fire, initiated by a small lighted candle in an electric cable penetration, spread and affected about 2000 cables causing damage to vital safety equipment
77-08-31	Cooper BWR 788 MWel 1974	Loss of essential electrical bus	Two independent failures caused interruption of DC power supply to the feedwater control system leading to partial loss of feedwater and high pressure in the reactor coolant system
78-03-20	Rancho Seco-1 PWR 917 MWel 1975	Loss of essential electrical bus	Shortcircuit caused interruption of power supply to non-nuclear instrumentation and erroneous signals, leading to dryboiling of steam generators and an overcooling transient
79-03-20	Three Mile Island-2 PWR 906 MWel 1978	Loss of feedwater, non-closure of relief valves, failure of safety injection	The combined effects of equipment failure, design deficiencies and operator error caused severe core damage and higher than normal radioactive releases to the environment
79-06-03	Hatch-1 BWR 768 MWel 1975	Loss of feedwater, failure of emergency core cooling system	Due to contaminated oil, the throttle valve of the steam-driven pump of the high-head emergency core cooling system failed to open
80-02-26	Crystal River-3 PWR 855 MWel 1977	Loss of essential electrical bus	Interruption of power supply to non-nuclear instrumentation caused erroneous signals leading to dryboiling of steam generator and loss of coolant due to an inadvertently open relief valve
80-06-28	Browns Ferry-3 BWR 1965 MWel 1977	Partial failure of reactor scram	At manual scram for planned outage, about half of the control rods did not fully insert due to failure of a discharge valve to the hydraulic drive system
80-10-17	Indian Point-2 PWR 873 MWel 1974	Flooding of the reactor containment	Due to a combination of several component failures, about 400 m ³ of service water leaked into the containment, which was not detected until the containment was opened for maintenance
82-01-25	R E Ginna PWR 470 MWel 1970	Loss of coolant due to steam generator tube rupture	Steam generator tube rupture resulted in rapid pressure drop in reactor coolant system and automatic scram. During cooling down, bubble formation occurred in the reactor coolant system. Increased radioactive releases to the environment were observed
83-01-25	Maine Yankee PWR 810 MWel 1972	Pipe break of feedwater line	In connection with reactor scram, water hammer occurred in the feedwater lines to two of three steam generators resulting in rupture of one pipeline

83-02-22	Salem-1 PWR 1079 MWel 1977	Failure of automatic reactor scram	Low water level in a steam generator at power ascension resulted in a scram signal, but both scram breakers remained closed until scram was actuated manually after 30 secs, when the breakers opened
85-06-09	Davis Besse-1 PWR 918 MWel 1978	Loss of feedwater	A combination of equipment failure and operator error caused loss of both main and auxiliary feed water systems resulting in rising temperature and pressure in the reactor coolant system. The relief valve opened three times but did not reclose the third time. The operator then closed the block valve. The feedwater system was restored after 12 minutes
85-12-26	Rancho Seco-1 PWR 917 MWel 1975	Loss of essential electrical bus	A single failure caused interruption of DC power supply to the integrated control system resulting in inadvertent automatic valve manoeuvres in the feedwater and turbine systems, causing an overcooling transient. The pressurizer emptied and a gas bubble was formed under the reactor pressure vessel head
86-12-09	Surry-2 PWR 811 MWel 1973	Pipe break in feedwater system	After inadvertent closure of a main steam line isolation valve causing turbine trip and reactor scram, a sudden double-ended rupture occurred in a bend of a 450 mm diameter feedwater pipeline. Eight workers were burned by the ejected water. Four of them died later. The pipe break was caused by wall thinning due to erosion/corrosion

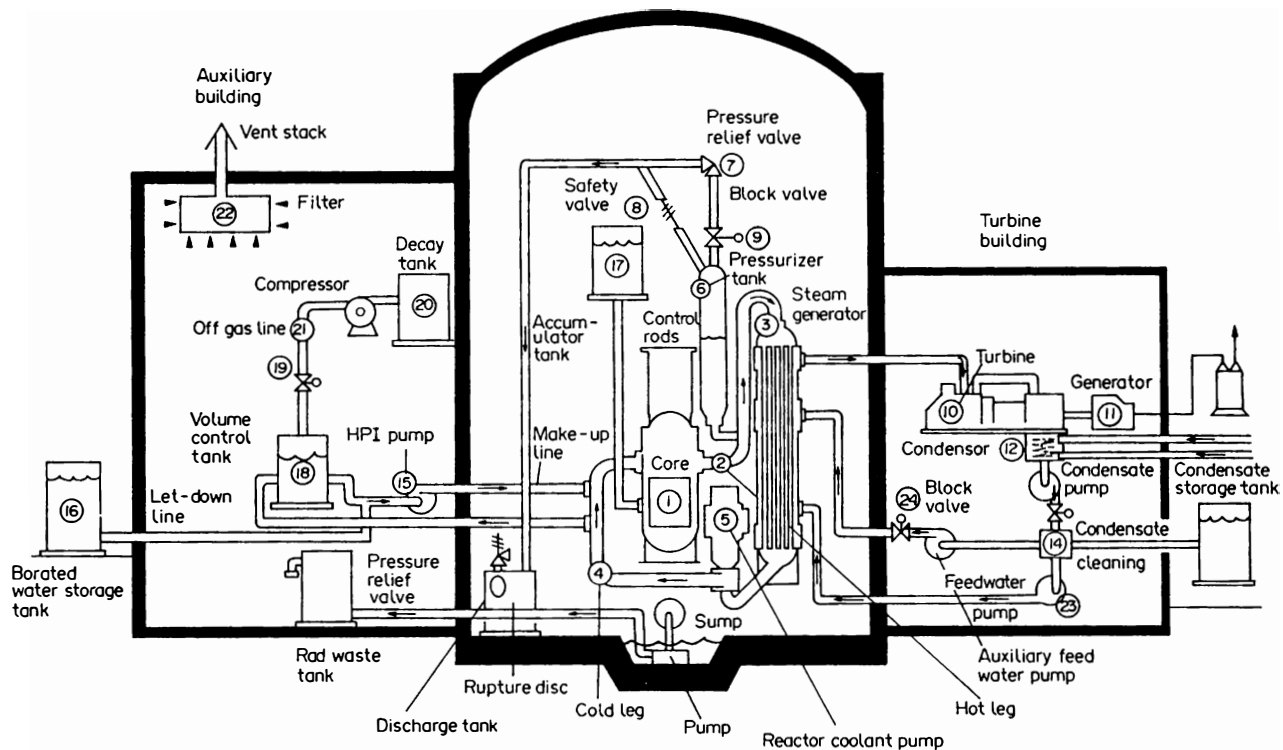


FIG. Schematic layout of TMI-2

vertently left in the closed position, probably in connection with the maintenance work carried out 2 days prior to the accident. Consequently, there was no cooling water flow on the secondary side, which caused the water in the steam generators to boil off within 2 minutes.

The second problem was the failure of a pressure relief valve to reseal when it received the signal to close after about 15 seconds. As a result, a leak appeared in the primary system, roughly corresponding to a small LOCA. A light on the control room instrument panel indicated that the pilot operated relief valve had been de-energized, and this led the operator to believe that the valve had closed. There was no direct indication of the position of the main valve.

Then followed a long and complex series of events and actions which had been detailed in the investigations after the accident (1307). The unintentionally closed block valves were discovered and opened after about 8 minutes. The open relief valve was only detected after nearly 2½ hours. The leakage was then stopped by closing the pressure relief valve block valve.

During the first stage of the accident, the operators had been misled into believing that there was too much water in the primary system when, in fact, the opposite was true. Therefore, when the safety injection system started up automatically after about 2 minutes and began to cool the core as intended, the operators only allowed it to operate for a few minutes before turning it off. As a result, the core was uncovered for several hours before the situation was brought under control. During that time, core damage was extensive.

After 1 hour, the main coolant pumps started to vibrate violently, probably because of cavitation. The operators then turned off two of the four pumps to avoid pump seal leakage. However, the vibrations continued and after a further 40 minutes the remaining two pumps were stopped, so that all forced coolant circulation ceased. After almost 3 hours one main coolant pump was restarted which then stopped again 20 minutes later because of violent vibration.

Hydrogen was formed by metal–water reaction (cf 3.4.6) in the fuel cladding. The gas collected as a “bubble” in the upper part of the reactor vessel. The dramatic race against time to remove the hydrogen bubble before the build-up of oxygen could result in an explosive mixture, occupied the attention of the mass media for several days. It was later realized that there was never any risk of a hydrogen explosion.

13.5.3 Releases and doses

It is estimated that most of the core inventory of noble gases and about 50% of the iodine and cesium as well as small amounts of other fission products were released from the fuel into the main coolant system during the accident. Some of the activity leaked from the coolant system through

the open relief valve to the pressure relief tank in the bottom of the reactor containment. When the tank overfilled and its rupture disc burst after about 15 minutes, the radioactive water landed in the containment sump and the gases were released into the containment atmosphere. At first, some of the water was pumped from the sump into the drain tanks in the auxiliary building.

Another leakage route from the primary system was created when the operators opened the letdown system (see Fig. 13.10), to drain off the supposedly excess water in the primary system. The letdown flow is normally led via a purification system to the volume control tank. The volume control tank is connected to an off-gas system which compresses the released gases and evacuates them via decay tanks and filters to the stack. The large amount of gas accompanying the primary coolant during the accident caused the off-gas system to overload and evacuate through the volume control tank safety valve.

On the basis of radiation dose measurements around the plant it has been estimated (1307) that 0.1–0.5 EBq of the noble gas xenon-133, corresponding to 2–10% of the core inventory, was released to the environment. In addition, an estimated 0.63 TBq of iodine-131 was released, which corresponds to 2.7×10^{-7} of the core inventory. This is about 100,000 times less than previously assumed for this kind of accident. As far as known, no cesium or other metallic fission product particles were released to the environment. When the accident occurred, cesium-137 and other long-lived fission products had not yet reached equilibrium due to the reactor's short operating history.

The largest offsite doses were obtained from radioactive xenon in the gaseous releases during the early part of the accident. The collective dose to the population within an 80 km radius from the plant was estimated at 33 mansievert (1308). This dose could result in one cancer fatality within a 30-year period. The maximum possible dose to a person residing in the vicinity of the plant has been estimated at 0.37 mSv, which is approximately equal to the average dose received in an ordinary X-ray examination.

The difference in the release fractions for noble gases and iodine can be explained by the following circumstances (1309):

- Noble gases do not react chemically with other elements, they are very volatile and not easily retained in water.
- Most of the released iodine was chemically absorbed in the reactor coolant water. Sodium hydroxide was injected into the containment, which increased the iodine adsorption, since the water became more alkaline.
- The generally reducing, hydrogen-rich containment atmosphere with very little free oxygen was favourable for the formation of metallic iodides.
- About 90% of the iodine released to the auxiliary building was collected by filters.

It is important that although the reactor containment was not completely leaktight, it remained mechanically intact. The radioactive substances were released through leaks to the auxiliary building. Most of the iodine released from the fuel is believed to have been converted into cesium iodide which easily dissolves in water and is relatively non-volatile. Therefore, it was either retained in the reactor coolant water or leaked from the primary system into the containment.

13.5.4 The recovery work

Once decay heat removal had been restored and the risk of an immediate, large release was over, the situation was as follows (1310). It was apparent that the core was severely degraded, but its detailed condition was unknown. In order to prevent unintentional criticality, boric acid was supplied to the coolant. The high activity of the reactor coolant and containment atmosphere made access to the containment impossible. The water level continued to rise in the containment due to leakage from the primary system, finally reaching a level of 2.4 m, corresponding to a water volume of 2500 m³. There were about 1500 m³ of medium-level radioactive water in the radwaste storage tanks in the auxiliary building. The general radiation level was so high that access to the building was only possible for short periods.

The first measure was to recover the auxiliary building for use as a work-site. The water was purified by means of a special ion exchange system installed in an adjoining building. The auxiliary building and fixtures were decontaminated, followed by the decontamination of the reactor building. In order to make access to the containment possible, the airborne activity had to be reduced, which was to be achieved by controlled releases to the environment. Although the releases of what was mainly krypton-85 would be small, the idea provoked a strong reaction from the public and permission was not granted until the summer of 1980. The slightly radioactive clean-up water was released by forced evaporation to the environment.

An important step in the clean-up process was to determine the condition of the core in preparation for its subsequent removal. This is carried out in four different ways: by mechanical drilling, video inspection, ultrasound, and sampling. The reactor vessel head was removed in July 1984. Work on removing the core began at the end of 1985 and is expected to continue for about 2 years. It is planned that the clean-up will be completed at the end of 1988.

The present picture (early 1988) is that the upper tie plate is largely intact although there is an approximately 1.5 m deep cavity in the upper part of the core which extends almost to the periphery of the core (Fig. 13.11). The cavity corresponds to approximately a quarter of the core volume. At the bottom of the cavity there is a 0.6 m high bed of debris consisting of UO₂,

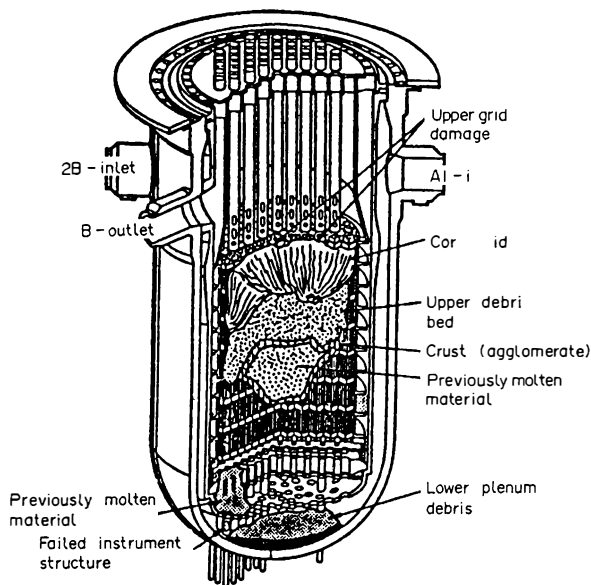


FIG. 13.11. TMI-2 end-state core configuration (1987)

Zircaloy and stainless steel. The debris rests on a hard crust of resolidified material and apparently intact remnants of fuel assemblies. Damage to reactor components below the core region appears to be less than expected. There are about 20 tons of debris at the bottom of the reactor vessel.

The analysis of bore samples indicates that UO_2 melting actually did take place (which had previously been questioned) but that the bottom 0.6–1 m of the core remained covered with water. The current hypothesis is that once the core had partly uncovered during the first 2–3 hours, the upper part of several fuel elements melted and fell into the lower part of the core. Clad oxidation and cracking in the remaining fuel rods was extensive. The temporary start-up of a main coolant pump caused rewetting and rupture of the cracked parts of the fuel rods. Consequently, the upper part of the core collapsed which resulted in the formation of the cavity and the gravel bed.

The parts of the fuel rods which had first fallen into the lower half of the core formed a lump of ceramic material. The rewetting appears to have resulted in the formation of an insulating crust around the lump. Therefore, the temperature in the inner uncooled region of the lump may have reached the melting point due to the decay heat. Finally, the crust at the bottom of the lump melted, causing the melt to break through and fall into the reactor vessel's lower plenum. Fragments were formed which were cooled by the remaining reactor coolant. The reactor vessel was then refilled and the accident sequence terminated. It is noteworthy that the melt did not break

through any of the numerous penetrations at the bottom, but remained in the reactor vessel.

13.6 Feedback of Experience

The analysis of safety-related events and the feedback of operating experience are important means of improving safety and maintaining a high level of safety. The objective is to identify significant events, to determine causes and to prevent recurrence. Probabilistic safety analysis is used for quantifying the significance of events and the effect of preventive measures.

13.6.1 Reliability data

Nuclear power plants contain a large number of mechanical and electrical components which are important to safety. Operating information and component failure reports provide the basis for compiling and processing statistical data. Failure probabilities are of great interest for the development of improved components and for use in probabilistic risk analysis. In Sweden, data are centrally collected and stored. Reliability data for components are processed from raw data and operating experience, and have been published in a handbook (1311). Two kinds of failure probabilities are of interest (cf 10.2.4):

- failure rates for components in operation;
- failure per demand for components on standby.

The handbook data mainly refer to components in safety-related systems in Swedish boiling water reactors. The type of components involved are pumps, valves, drive mechanisms/control rods, instruments and diesel generators. Pumps, valves and instruments are grouped into main categories for which generic information is presented. The plant-specific updating of generic information is carried out by statistical methods.

13.6.2 Incident evaluation

The utilities have been exchanging information from the operation of nuclear power plants for a long time. The importance of a structured exchange was highlighted by the TMI-2 accident, after which the U.S. utilities set up a computerized data base system. Similar systems were implemented in Sweden and other countries. The aim is to rapidly disseminate correct information on safety-related events, as well as to evaluate significant events and recommend action for improving safety.

The reports on safety-related events and scrams which are submitted to the Nuclear Power Inspectorate are the basis of the Swedish system. The

reports are screened for significant events by applying qualitative criteria, for example determining whether multiple failures or common cause failures have occurred or whether the Technical Specifications have been violated. Recurrent failures and conditions which indicate deterioration of the fuel, the primary system or the containment are also of concern. Experience has shown that the significant events represent less than 5% of all reported safety-related events.

The significant events are subjected to closer analysis to determine whether corrective action is necessary and what kind should be adopted. In this respect, event tree methodology is used to determine the risk of severe core damage and the effects of risk-reducing measures. According to U.S. experience, about 25% of the significant events result in corrective action. Corrective action can include anything from ensuring that the operator staff is made aware of the problem, to the modification of equipment, procedures and instructions.

The Swedish system for experience feedback (1312) is managed by the utilities' Nuclear Training and Safety Centre (cf 7.4.6). The system contains data from both Swedish and foreign nuclear facilities which are stored in a central computer. Incoming reports on safety-related events are screened and classified into three categories:

- significant events, which are analysed in detail;
- recurrent events, which are subjected to trend analysis;
- events which do not require closer analysis but which are stored for statistical reference.

According to Swedish experience, significant events represent less than 2% of all reported safety-related events. If an event is deemed as requiring action for safety improvement, recommendations are made to the utilities who are then responsible for their implementation.

13.6.3 Precursor analysis

Precursor analysis is a quantitative method of evaluating significant events. A *precursor* is an observed event which, in combination with one or several postulated events, may lead to severe core damage. Precursor analysis was introduced in a U.S. study (1313), known as the ASP study (Accident Sequence Precursor).

All reports on safety-related events in U.S. reactors from 1969 to 1979 were screened in order to identify and classify the precursors. An event is designated as a precursor if any of the following conditions is fulfilled:

- loss of at least one function needed to counteract an initiating event which could result in core damage;

- partial loss of at least two functions needed to counteract an initiating event;
- an unusual initiating event, such as loss of offsite power, a stuck-open relief valve.

Event trees for the real event as well as for postulated core damage sequences, of which the precursor is an integral part, are constructed for each precursor. The first type of event tree is used to estimate the probability of recovering the unavailable function within a certain time, by operator action or otherwise. The other event tree is used to calculate the *conditional* probability of core damage, provided that the precursor occurs. The conditional probability can be considered as a measure of the potential risk of severe core damage.

In the ASP study, 19,400 events were screened, of which less than 1% were identified as precursors. Of these, fifty-two were estimated to have implied a conditional probability of core damage greater than 1 in 1000. The events with the highest probability are presented in Table 13.12.

The total number of operating years for U.S. light water reactors from 1969 to 1979 was 432. Since each precursor occurred once, the total probability of core damage *during the particular period* can be calculated as 1/432 times the sum of all the conditional probabilities. A value (point estimate) of between 1.7×10^{-3} and 4.5×10^{-3} is thereby obtained. The values are dominated by the events at TMI-2, Browns Ferry and Rancho Seco (cf

TABLE 13.12. *Conditional probability for core damage during occurrences in U.S. reactors 1969–79*

Reactor	Event	Probability
TMI-2	Loss of feedwater, non-closure of safety valve, failure of safety injection	1
Browns Ferry 1	Loss of feedwater due to cable fire	0.39
Rancho Seco	Loss of feedwater due to failure of non-nuclear instrumentation	0.25
Point Beach 1	Shutdown transient with loss of auxiliary feedwater	0.025
Turkey Point 3	Failure of auxiliary feedwater pumps to start during testing	0.025
Kewaunee	Failure of auxiliary feedwater pumps during reactor start-up	0.025
Davis Besse 1	Failure of auxiliary feedwater pumps during testing	0.025

Source: J W Minarick, C A Kukielka, *Precursors to Potential Severe Core Damage Accidents 1969–1979. A Status Report*, USNRC Report NUREG/CR-2497, Vol 1, U.S. Nuclear Regulatory Commission, 1982

Table 13.11), which together account for 85% of the total probability. The following conclusions were also drawn from the study:

- many of the empirical failure probabilities and frequencies for initiating events are in fair agreement (within a factor of 10) with those used in the Reactor Safety Study;
- no correlation between the number of precursors and the age, manufacturer or capacity of the reactors could be made;
- about 38% of all precursors involved human error.

The ASP study has been criticized for underestimating the possibilities of recovering or compensating for a functional failure. According to a critical evaluation of the study (1314), the conditional probabilities for the dominating events are overestimated by a factor of 300 to 3000. Criticism was also focused on the method of combining plant-specific and generic information in the event trees, which is unsatisfactory from a theoretical standpoint, and overestimated the conditional probabilities.

13.6.4 Bayesian analysis

As operating experience from nuclear power plants increases, the assessment of risk based on this experience becomes more reliable. Operating experience can be used for updating the results of probabilistic risk analysis by the application of Bayesian methodology (1315). This method is based on Bayes theorem in the theory of probability, which states that:

$$p_B(A) = \frac{p(A) \cdot p_A(B)}{p(B)} \quad (13.1)$$

where $p_B(A)$ is the conditional (*a posteriori*) probability of event A when event B is known to have occurred, and $p(A)$ is the (*a priori*) probability of A (without knowledge of B). When applying equation (13.1), event A is made to represent severe core damage while B represents the total experience of significant precursors.

As an example, a study was carried out (1316) based on estimated core damage frequencies from a number of safety studies (Table 13.13), which were updated using the ASP study precursor analysis described in the previous section. The results are shown in Fig. 13.12. The theoretical analysis has, as it were, been rendered more reliable by the use of operating experience, regardless of whether core damage occurred or not.

13.7 The Chernobyl Accident

On 26 April 1986 an accident occurred at Unit 4 of the Chernobyl nuclear power station in the Ukraine, which was to be the most serious accident to

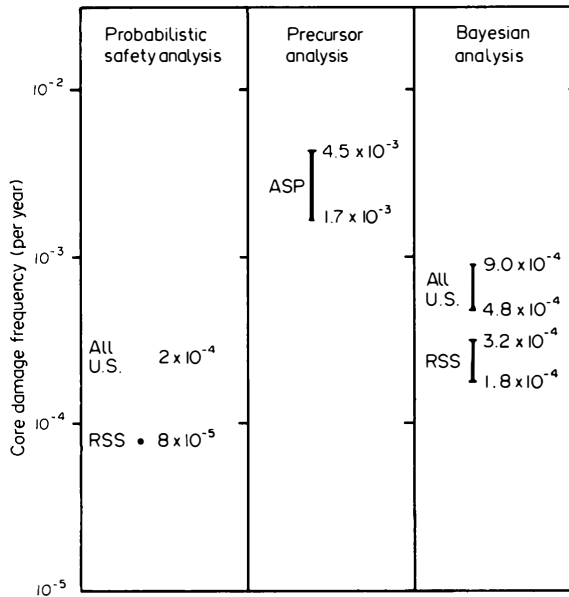


FIG. 13.12. Comparison of estimated core damage frequencies

TABLE 13.13. Comparison of estimated core damage frequencies

Unit	Type of study	Core damage frequency (PMY)	
		Median value	Mean value
<i>Pressurized water reactors</i>			
Surry	RSS ^a	60	120
Biblis B	GRSS ^b	40	96
Indian Point 2	I ^c	70	90
Indian Point 2	I+E ^d	400	470
Indian Point 3	I	60	130
Indian Point 3	I+E	90	190
Zion	I	50	57
Zion	I+E	62	67
Oconee	RSSMAP ^e	200	400
Sequoyah	RSSMAP	60	120
<i>Boiling water reactors</i>			
Peach Bottom	RSS	30	60
Big Rock Point		970	1000
Limerick	I	15	28
Grand Gulf	RSSMAP	30	60

^a Reactor Safety Study.^b German Safety Study.^c Internal events.^d Internal and external events.^e Reactor Safety Study Methodology Application Programme.Source: C D Heising, A Mosleh, Bayesian Estimation of Core Damage Frequency Incorporating Historical Data on Precursor Events, *Nucl. Safety*, Vol 24, No 4, 1983

have happened in a nuclear power reactor in the world. The reactor core and parts of the reactor and turbine buildings were destroyed, and large amounts of radioactive materials were released to the atmosphere. Evacuation of the surrounding area was required, and fallout from the radioactive cloud affected countries outside the USSR.

Although the destroyed reactor was quite different from the reactors treated in this book, it is necessary to understand the causes and effects of the accident and to evaluate the possible implications for the safety of light water reactors. In this section, therefore, a brief account is given of the reactor design and physics characteristics, the accident chronology and the radiation impact as well as of the information derived from the accident analysis. The description of the reactor and the accident is largely based on information made public by Soviet specialists at the IAEA Experts' Meeting in Vienna, August 1986 (1317) as interpreted and extended in a report published by the U.K. Atomic Energy Authority (1318).

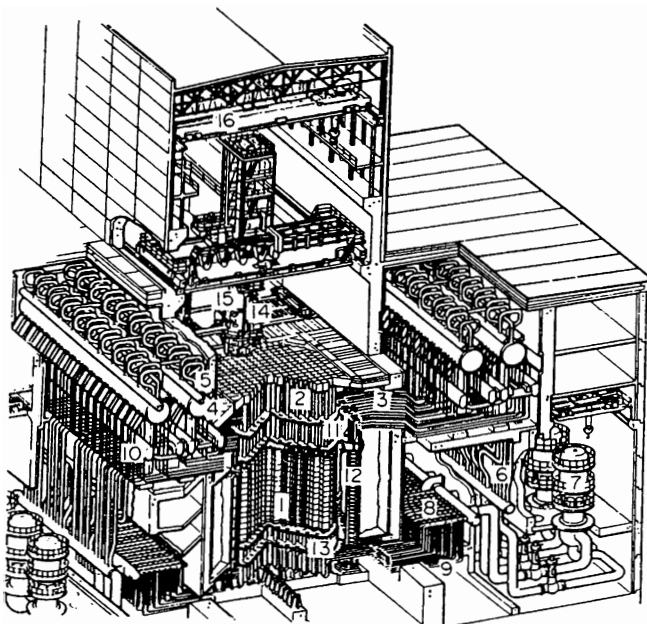
13.7.1 The reactor

The Chernobyl nuclear power station is located on a tributary of the river Dnjepr near the town of Pripyat (population 49,000) about 120 km north of Kiev. The station had four 1000 MWel RBMK reactors in operation and two more under construction at a distance of 1.5 km. The four reactors were built in pairs, sharing common buildings and services. Construction of Units 3 and 4 started in 1975/76 and Unit 4 was commissioned in 1984.

The RBMK is a *graphite-moderated, pressure-tube reactor cooled by boiling water*. The combination of a pressure-tube coolant circuit with a graphite moderator in a commercial nuclear power plant is unique to the USSR. Its origin can be traced to the early reactors built to produce military plutonium. The chief design features are:

- vertical pressure tubes, containing the fuel and coolant, enabling on-load refuelling;
- fuel assemblies in the form of eighteen-rod clusters, each rod consisting of slightly enriched uranium dioxide fuel pellets in a zirconium alloy cladding tube;
- graphite moderator and reflector, enclosed in a leaktight shell filled with slowly circulated helium/nitrogen mixture;
- boiling water coolant in forced circulation, supplying steam directly to the turbine.

The RBMK-1000 reactor has a thermal output of 3200 MW. Figure 13.13 shows a sectional view. At the centre is the reactor core with its supporting structures and biological shielding. The reactor coolant circuit has two identical loops, each with four main recirculation pumps, supplying water to the



- 1 Reactor
- 2 Fuel - channel standpipes
- 3 Steam / water riser pipes
- 4 Steam drums
- 5 Steam headers
- 6 Downcomers
- 7 Main circulating pumps (MCP)
- 8 Group distribution headers
- 9 Reactor inlet water pipes
- 10. Burst - can detection system
- 11 Upper biological shield
- 12 Side biological shield
- 13. Lower biological shield
- 14. Irradiated fuel storage pond
- 15 Fuelling machine
- 16 Bridge crane

FIG. 13.13. Sectional view of an RBMK-1000 reactor

fuel channels. The water in the channels is heated to boiling point and partially evaporated. The steam/water mixture is transported to the steam drums where steam and water are separated. Above the reactor is the reactor hall with the fuelling machine. A containment building partially surrounds the reactor and primary circuit. Further design information is presented in the UKAEA report (1318).

13.7.2 Physics characteristics

The basic design of RMBK reactors has some shortcomings from the standpoint of safety, the most important being the unfavourable reactivity coefficients. The “optimized” RMBK-1000 design has a positive void coefficient, a relatively small fuel temperature coefficient, and a positive moderator temperature coefficient. This is illustrated in Fig. 13.14 which shows the variation of the reactivity coefficients with operating time.

The explanation for the positive void coefficient is that the coolant water acts predominantly as a neutron absorber in the equilibrium core. The negative void coefficient for fresh fuel is due to the presence of solid absorber rods for eliminating excess reactivity in the initial core. This decreases the relative neutron absorption in the coolant and makes the negative effect of reducing moderation predominate when coolant is removed. In normal light water reactors, the moderating effect of the coolant is always much stronger than the absorbing effect so that the void coefficient is negative.

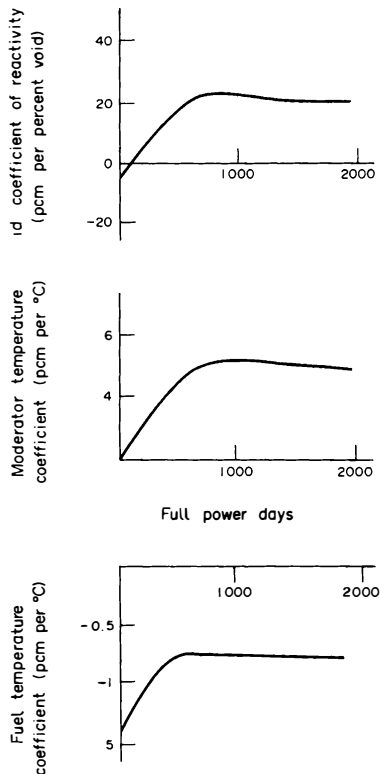


FIG. 13.14. Reactivity coefficients in an RMBK-1000. Adapted from V S Romanenko, A V Krayushkin, Physical Characteristics of an RMBK Reactor in the Transitional Period, *Atomnaya Energia*, Vol 53, No 6, 1982

The increasing positive contribution to the fuel temperature coefficient with operating time is due to the build-up of plutonium in the fuel and is a characteristic of well-moderated reactors with oxide fuel such as the RBMK. For light water reactors, which have a harder (more energetic) neutron spectrum, the fuel temperature coefficient is more negative and less dependent on burn-up.

The relatively small fuel temperature coefficient has important implications for the possibility of RBMK reactors to limit reactivity-induced power excursions (cf 3.3.3). The energy deposited in the fuel during a self-limited power excursion is approximately inversely proportional to the prompt negative reactivity coefficient. Therefore, the smaller the fuel temperature coefficient, the more energy will be deposited.

The positive void coefficient is destabilizing, but the combined effect of the positive void coefficient and the negative fuel temperature coefficient is that the *power coefficient* (cf 3.3.5) is negative during normal operating conditions. However, if the coolant is saturated or near saturation a small power increase (or pressure decrease) will give a relatively large voidage increase. Therefore, the power coefficient may become positive and the reactor unstable under certain conditions of low power and high coolant flow. It is reported (1318) that sustained operation of RBMK-1000 below 20% power is prohibited according to the operating rules.

The void coefficient can be made less positive and even negative under normal operating conditions by increasing the fuel enrichment, or by operating the reactor with absorber rods in the core (cf Fig. 13-16). Both these measures have the effect of decreasing the relative neutron absorption in the coolant, thus decreasing the positive reactivity effect of reducing coolant density.

Like all well-moderated graphite reactors, the RBMK has a positive moderator temperature coefficient, mainly due to the effect of plutonium build-up in the fuel. The positive coefficient has a positive feedback effect on the power, but the instability is easily controlled because of the relatively large time constant for changes of the moderator temperature.

As in all thermal high neutron flux reactors the fission product xenon-135 has a destabilizing effect on the reactor power, which is easily controlled because of the large time constant involved. However, in physically large reactors such as the RBMK, the positive reactivity feedback due to xenon (and moderator temperature) gives rise to instability not only of the power level, but also of the power distribution in the core (cf 3.3.7). The RBMK therefore requires a fairly complex control system to stabilize the power density distribution as well as the power level.

It is evident that control rods for emergency protection should be capable of quick insertion. In the RBMK the control rods are motored into the core at a speed of 0.4 m/s. This relatively slow speed is partly compensated for by the large number of rods. However, fully withdrawn rods would have to

move a considerable distance in order to produce significant effects. An operating rule therefore requires that a specified number of control rods be partially inserted so that they will quickly produce an adequate effect in case of scram actuation.

13.7.3 The accident sequence

The accident was triggered by an experiment, planned to be carried out in connection with the annual maintenance shutdown. The objective was to determine whether a turbo-generator, cut off from both its steam supply and the grid, would be capable by means of its mechanical inertia of supplying power to essential systems during a short period after a power failure. The experimental initial conditions required the reactor to operate at about 25% of full power with one of its turbo-generators shut down. The other turbo-generator, which was to coast down, would supply two main recirculation pumps in each loop. The remaining two pumps in each loop and the auxiliary plant were to be fed from the grid.

Just over 24 hours before the accident, the reactor was operating at full power. Then at 01.00 hours on 25 April, power reduction for the maintenance shutdown was begun. At 13.05 hours the 50% level had been reached, and one of the turbo-generators was switched off as planned. At 14.00 hours the emergency core cooling system was disconnected in accordance with the experimental programme. However, further power reduction was delayed by a request from the regional power controller to keep supplying the grid. Operation continued for nearly 10 hours at half power. This caused the xenon concentration to increase, which necessitated the withdrawal of more control rods than anticipated to compensate for xenon poisoning (cf 3.3.7).

At 23.10 hours power reduction was resumed. However, the automatic power control system was unable to stabilize the power at the desired 700–1000 MWth, and the power fell to 30 MWth. At 01.00 hours on 26 April, the operator succeeded in stabilizing the power at 200 MWth. A further increase was difficult due to the xenon poisoning. To reach the 200 MWth level the operator had been forced to withdraw control rods in excess of the limit established in the operating rules. Even so, it was decided to proceed with the experiment. During the further preparations, the reactor was brought into a clearly non-permissible state. The temperature and pressure in the primary circuits were close to saturation, the coolant flow exceeded permissible values and the feedwater supply was overbalanced.

At 01.23.04 the experiment was started by closing the emergency stop valve to the turbine. This should have caused the reactor to shut down, but the corresponding emergency protection signals had been blocked to permit the experiment to be repeated if it were not successful the first time. Shortly after, the reactor power began to rise slowly. At 01.23.40 the operator pressed the emergency stop button, which would insert all control and emer-

agency rods into the core. The rods began motoring in, but after a few seconds a number of shocks were felt and the operator saw that the rods had halted without inserting fully to the lower stops. The rods were then manually released so that they would fall by their own weight.

According to observers outside the plant two explosions were heard at about 01.24. Burning material and sparks were ejected into the air, some of which fell on the roof of the turbine building and started a fire there. The fire alarm reached the fire brigades in the nearby towns of Pripyat and Chernobyl within 5 minutes and three fire brigades were in place within 15–30 minutes. The fires were brought under control during the period 02.10–02.30 and were completely extinguished at 05.00 hours. Unit 3 was then shut down and Units 1 and 2 were shut down after about 24 hours.

Immediately after the accident, attempts were made to cool the destroyed core with water via the emergency core cooling system. This was not successful, since the pipeline system was damaged. It was therefore impossible to avoid the graphite fire which started and spread in the destroyed core on the day of the accident.

The following day, efforts were started to cover the burning core with boron, dolomite sand, clay and lead from helicopters. In total, about 5000 tons were dumped, mainly from 28 April to 2 May. The dumped material acted as a heat sink and cooled the core. After the dumping had ceased, the temperature in the core and the release of radioactive substances increased again and reached a maximum on 5–6 May. At that time, core cooling with nitrogen started, which rapidly decreased the temperatures and releases.

13.7.4 Analysis of the accident

When the experiment began, the reactor was in an unstable state at low power and high coolant flow. Most of the control rods had been withdrawn from the core to compensate for the reactivity loss due to the high xenon content and low voidage. The coolant water was almost at its boiling point. When the turbine stop valve was closed and the recirculation pumps began to coast down, the flow reduction soon caused boiling in the fuel channels. The generation of steam increased the reactivity because of the positive void coefficient, and the rate of steam generation increased due to the positive feedback. This led to a power excursion in spite of emergency shutdown actuation.

The variation of reactivity and power during the transient, as demonstrated in the simulation of the accident by the Soviets, is shown in Fig. 13.15. Time zero in the diagram corresponds to the time when the operator pressed the emergency shutdown button.

As shown in the diagram, the reactivity was slightly positive and rising already at time zero. Since the delayed neutron fraction is estimated at about 0.4%, the reactivity reached 1 dollar and the reactor went prompt

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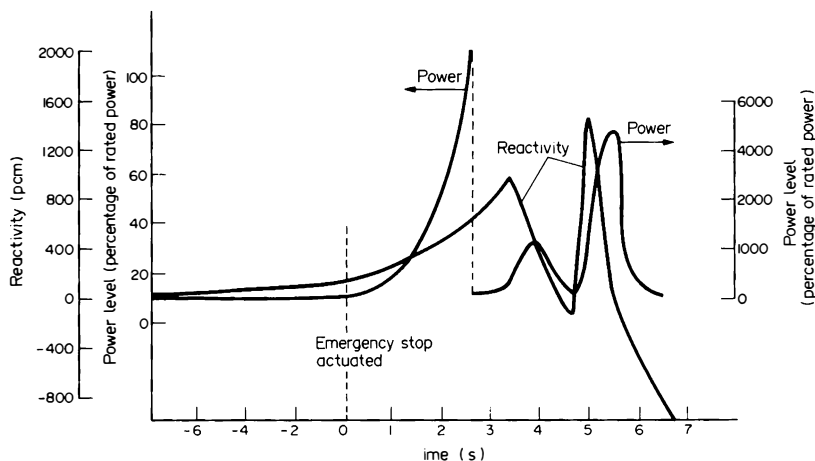


FIG. 13.15. Time variation of reactivity and power in the simulation of the Chernobyl accident. Adapted from USSR State Committee on the Utilization of Atomic Energy, *The Accident at the Chernobyl Nuclear Power Plant and Its Consequences*, Information compiled for the IAEA Experts' Meeting, 25–29 August 1986, Vienna

critical about 2 seconds later. The reactivity rose to about 1000 pcm or 2.5 dollars at time 3.5 seconds, after which it decreased and passed a minimum before it increased steeply to about 1500 pcm (3.8 dollars) at about 5 seconds.

The (average) power level rose rapidly from about 10% of nominal 3200 MWth to 100% in 2.5 seconds to reach a first maximum of about ten times nominal power at approximately 4 seconds. The peak power level corresponds to a heat rate of about 200 watts per gramme of fuel. The power then decreased and passed a second maximum corresponding to a peak heat rate of about 1000 W/g. Thus, there are two power peaks within 1.5 seconds.

The analysis shows that the reactor was on a positive reactivity ramp, estimated at 250 pcm/s, due to the positive void coefficient, already at time zero, when emergency shutdown was actuated. The scram system was far too slow to shut the reactor down within the time scale of the accident. Instead, the reactivity ramp caused the power to increase with a doubling time of about 0.2 seconds.

When the power increases, energy is deposited in the fuel and a negative reactivity contribution is obtained due to the Doppler effect (3.3.4). With an estimated Doppler coefficient of -0.7 pcm/°C, a temperature increase of about 1500°C is required to compensate for the positive ramp reactivity. The first power excursion is therefore probably limited by the Doppler effect. The peak fuel pellet enthalpy (sum of deposited and stored energy) in the first power pulse is estimated at about 200 cal/g UO₂. This will cause dryout but probably no serious fuel damage if the coolant flow is sustained.

The coolant flow continued to decrease, however, and the pressure in the fuel channels increased, so as to eventually block the coolant flow completely. At this time, at about 5 seconds, there was an abrupt increase of the voidage and the reactivity to superprompt criticality. Since the fuel temperature was already high, the Doppler effect was not sufficient to limit the excursion, and the fuel melted and disintegrated. The disruption of the fuel introduced negative reactivity and terminated the second power excursion.

The peak fuel pellet enthalpy in the second power pulse is estimated at more than 400 cal/g U_2 , which is sufficient to destroy the fuel (cf 3.4.7). When particles of destroyed fuel were ejected into the coolant, a violent interaction resulted that caused a rapid and abrupt pressure increase in the fuel channels and ruptured the pressure tubes. This is estimated to have occurred at about 7 seconds.

When the pressure tubes ruptured, the main recirculation pumps could again supply water to the core. However, at this stage the flow was no longer directed into intact channels but into the reactor space. The steam generation and the rapid rise in core temperature created the appropriate conditions for the metal–water reaction (cf 3.4.6) and other exothermal reactions. As a result, a mixture of gases was formed containing hydrogen and carbon monoxide which then led to a chemical explosion upon mixing with oxygen in the air. This mixing became possible after the upper shield (see Fig. 13.13) had been blown off.

The energy required to destroy the fuel, rupture the pressure tubes and throw off the 3 m thick upper shield could have been supplied by fuel–coolant interaction or by the thermal energy already stored in the fuel channels. It is estimated (1318) that any of these energy sources might yield mechanical work of the order of 1 GJ. This compares with rough estimates in the range 0.2–2.0 GJ of the work done in blowing off the upper shield. Rough estimates also show that the nuclear energy released in the power excursions was much less than the chemical energy released in the metal–water reaction and the gas explosion, and several orders of magnitude less than that of a small nuclear explosion.

In summary, the Chernobyl accident was triggered by a prompt-critical reactivity excursion causing a rapid power surge, severe fuel destruction, and violent fuel–coolant interaction. It was due to fundamental design deficiencies and erroneous operator action under abnormal operating conditions. No unknown phenomena or mechanisms were revealed. The accident started as a reactivity-induced accident (RIA) and proceeded as a loss-of-coolant accident (LOCA).

13.7.5 Radioactive releases

When the upper shield was blown off and the reactor building destroyed, hot fuel fragments together with vapours of volatile fission products were ejected directly into the atmosphere. Most of the particulates were deposited in the vicinity of the plant, but the heat from the hot steam and gases made a large part of the smaller particles rise more than a thousand metres in the atmosphere. A radioactive cloud was formed and transported in a north-westerly direction.

The graphite fire promoted a high level of continuing activity release during the following days, but the dumping of material onto the core debris led to a steady reduction in activity release until 2 May. During this time additional particles of graphite and dust with attached radioactive substances were raised although probably not as high as during the initial stage. This material settled mainly within a few tens of kilometres from the reactor site.

When the dumping had ceased, the core temperature, driven by decay heat, rose during 3–5 May and a steady increase in activity release occurred, especially of iodine. A second peak in the activity release resulted on 5 May. A sharp decline occurred on 6 May, coinciding with the injection of nitrogen under the core debris for cooling.

The Soviet account of the source terms is shown in Table 13.14. Some 100% of the noble gases, 10–20% of the volatile fission products iodine, cesium and tellurium, and 3–4% of all other radionuclides escaped to the environment over a 10-day period from 26 April to 6 May. In total, about 1.85 EBq (50 MCi) of released activity was present in the environment on 6 May.

The magnitude of the release in terms of the core inventory roughly agrees with the predictions in the worst cases of the Reactor Safety Study (see Table 11.11). However, the extended release period contrasts strongly with the release periods of at most a few hours predicted in the analyses of severe accidents for the light water reactors. It is likely that UO_2 oxidation played a key role in determining the magnitude as well as the release rate of the fission products (1319).

It is interesting to compare the activities of iodine-131 and cesium-137, released into the atmosphere at the three most-discussed reactor accidents: Windscale, Three Mile Island and Chernobyl (Table 13.15). For comparison the estimated release of cesium-137 from all nuclear weapons tests is also shown.

13.7.6 Radiation doses

The exposure rate in Pripyat about 5 km from the reactor site was low initially but started to rise rapidly about 20 hours after the accident. There-

TABLE 13.14. *Core inventories and releases in the Chernobyl accident*

Element	Half-life (d)	Core inventory ^a (Bq)	Percentage released
Krypton-85	3930	3.3E16	100
Xenon-133	5.27	1.7E18	100
Iodine-131	8.05	1.3E18	20
Tellurium-132	3.25	3.2E17	15
Cesium-134	750	1.9E17	10
Cesium-137	1.1E14	2.9E17	13
Molybdenum-99	2.8	4.8E18	2.3
Zirconium-95	65.5	4.4E18	3.2
Ruthenium-103	39.5	4.1E18	2.9
Ruthenium-106	368	2.0E18	2.9
Barium-140	12.8	2.9E18	5.6
Cerium-141	32.5	4.4E18	2.3
Cerium-144	284	3.2E18	2.8
Strontium-89	53	2.0E18	4.0
Strontium-90	1.02E4	2.0E17	4.0
Neptunium-239	2.35	1.4E17	3
Plutonium-238	3.15E4	1.0E15	3
Plutonium-239	8.9E6	8.5E14	3
Plutonium-240	2.4E6	1.2E15	3
Plutonium-241	4800	1.7E17	3
Curium-242	164	2.6E16	3

^aDecay corrected to 6 May 1986 and calculated as prescribed by the Soviet experts.

Source: USSR State Committee on the Utilization of Atomic Energy, *The Accident at Chernobyl' Nuclear Power Plant and Its Consequences*, Information compiled for the IAEA Experts' Meeting, 25-29 August 1986, Vienna

TABLE 13.15. *Comparison of activity releases*

Accident	Activity release Iodine-131	(PBq) ^a Cesium-137	Cs-137 over Sweden
Windscale	0.75	0.02	0
TMI-2	0.0005	0	0
Chernobyl	300	50	4
All nuclear weapons tests	?	1000	1

^a 1 PBq = 10¹⁵ Bq.

Source: B Lindell, Radiation Risks and Chernobyl, *Vår föda*, Vol 38, Supplement 3, Swedish National Food Administration, 1986

fore, the town was completely evacuated, which was accomplished within 3 hours about 30 hours after the accident. It is estimated that the inhabitants received whole-body doses of 15–50 mSv from gamma radiation and skin doses of 100–200 mSv from beta radiation. These doses are insufficient to cause early radiation effects. The collective dose to the inhabitants of Pripyat is estimated at 1500 manSv (1317).

Because of increasing radiation levels, the whole surrounding area up to a radius of 30 km was evacuated after a few days. The estimated radiation dose to the population in the vicinity of the reactor site is shown in Table 13.16.

Because of the evacuation, the individual doses were less than 1000 mSv, which means that nobody suffered acute radiation sickness.

TABLE 13.16. *Estimated radiation doses near the reactor site*

Distance km	Number of places	Thousands of persons	Average dose mSv	Collective dose manSv
Pripyat	1	45	33	1500
3- 7	5	7	540	3800
7-10	4	9.0	460	4100
10-15	10	8.2	350	2900
15-20	16	11.6	52	600
20-25	20	14.9	60	900
25-30	16	39.2	46	1800
Total	72	134.9	120	15,600

Source: Information compiled for the IAEA Experts' Meeting, 25-29 August 1986, Vienna

At distances larger than 30 km, no evacuation was undertaken. The ground deposit at 30 km resulted in doses about five times larger than those at 100 km. The total integrated doses, including ingested activity in contaminated foodstuffs, is estimated at a few hundred mSv in the region from 30 to 100 km. These doses are of the same order as the highest doses received by evacuated residents in the inner zone. This means that the residents near to the plant are not expected to run a higher risk of late effects than those living farther away.

At distances of more than 100 km, wet deposition during periods of rainfall caused a marked patchiness in the environmental activity concentration. It is those ground doses and the food doses which determine the future integrated collective doses. The total collective dose, summed over all countries in Western and Eastern Europe (except the USSR), is estimated at 1.8×10^5 manSv (1318), about equally divided between ground dose and ingestion dose. The corresponding figure for the USSR is estimated at 5×10^5 manSv.

13.7.7 Health effects

At the time of the accident, there were three persons in the control room and four or five in the turbine building. Two persons died immediately of burns. About 500 people were hospitalized, including employees at the

plant and firemen, who made heroic efforts to fight the fires in the reactor and turbine buildings. About 150 suffered acute radiation sickness, twenty-eight of whom died (Table 13.17).

The medical treatment of patients in categories 3 and 4, i.e. with doses in excess of 4 Gy, was complicated since the exposure was very non-uniform, with severe thermal and beta radiation burns. Twenty-six people died within 10 and 50 days after the accident. In many cases, already the skin damage was fatal. The attempts to carry out bone marrow transplantation had limited success.

The latent cancer effects can be estimated on the basis of the linear dose-risk relationship. Using a risk coefficient of 0.02 per mansievert, the total number of cancer fatalities over the next 50-year period are estimated at 10,000 in the USSR and 4000 in the rest of Europe. During the same time, approximately 35 million people would ordinarily die of cancer in the USSR. This means that Chernobyl may cause 0.03% additional cases.

TABLE 13.17 *Acute fatalities and radiation exposure at Chernobyl*

Category	Number hospitalized		Estimated doses Gy	Fatalities 25 Aug. 1986
	Kiev	Moscow		
4	2	20	6-16	20
3	2	21	4-6	6
2	10	43	2-4	2
1	74	31	1-2	

Source: Verbal information at the IAEA Experts' Meeting, 25-29 August 1986, Vienna

13.7.8 Implication for light water reactors

Although the Chernobyl RMBK reactor had little in common with light water reactors, the accident highlighted several important aspects of reactor design, operation and safety analysis. Many of these aspects were also highlighted by the Three Mile Island accident, and as a result have been extensively studied against current criteria and practice in the countries operating light water reactors.

The basic difference between the Three Mile Island and the Chernobyl accidents is that the former was a loss-of-coolant accident (LOCA) leading to relatively slow core melting, while the latter was a reactivity-induced accident (RIA) with rapid fuel disruption.

At least three RIAs are known to have occurred prior to Chernobyl: in the experimental reactors NRX, EBR-1 and SL-1. NRX is a heavy water moderated reactor at Chalk River, Canada, which was severely damaged in a power excursion in 1952. EBR-1 was a liquid sodium cooled fast reactor

in Idaho, USA, which was destroyed in a fast reactivity excursion in 1955. SL-1 was a U.S. experimental light water reactor destroyed in 1961 by a power excursion when an operator withdrew a control rod too far.

Many deliberate experiments and extensive analyses of RIA in light water reactors have been carried out. The general conclusion is that this type of accident must be prevented to a high degree of reliability. Rapid reactivity insertion by control rod ejection is avoided by design. Too fast control rod withdrawal during start-up is precluded by interlock arrangements. Although transients involving superprompt criticality cannot be ruled out in light water reactors, studies show (cf 9.6.1 and Fig. 9.14) that the resulting power excursions will be limited by the Doppler effect before excessive energy deposition occurs and the fuel is seriously damaged.

At an early stage it was verified by experiment that light water reactors normally have a strongly negative void coefficient. This fact alone excludes the possibility of a Chernobyl-like accident in a light water reactor. The void coefficient may be slightly positive under certain circumstances, such as in a PWR at room temperature with a large boron concentration in the moderator. Criticality is avoided in these conditions by prohibiting cold start-up. The void coefficient may become positive also in very closely packed PWR lattices outside the range of today's core design.

The reverse of the negative void coefficient is the positive pressure coefficient of reactivity in boiling water reactors. The pressure must therefore be carefully controlled and sudden pressure increases avoided. Pressure transients within the design basis are subjected to analysis in the licensing process (cf 9.6.4). The Chernobyl accident has stimulated interest also in the analysis of pressure transients beyond the design basis.

Since the Three Mile Island accident, the studies of severe accidents have been mostly devoted to relatively slow core meltdown processes due to insufficient core cooling. Powerful steam explosions when a core melt falls under gravity into water are considered physically impossible (cf 11.1.2). In Chernobyl, the destruction of fuel occurred very rapidly and fragments of partly molten fuel were ejected under high pressure, violently interacting with the coolant water. In this case the fuel was fragmented into fine particles, allowing very rapid steam generation, a steam explosion. The detailed mechanisms in this type of steam explosion are insufficiently known.

Another lesson learned from Chernobyl is that large amounts of radioactive materials can be released without coherent core melting. The Chernobyl release was very energetic and prolonged. While probably unique to RBMK type of reactors, certain phenomena may have occurred that can also be of interest to light water reactors. These include mechanical release of radionuclides from core debris, revaporization and resuspension of previously deposited radionuclides, the transport of various forms of iodine, and hydrogen generation from dispersed fuel fragments (1320).

Fuel oxidation was a major release mechanism in the Chernobyl accident.

Oxidative release from fuel can arise in the containments of PWR and BWR, following steam explosion or high-pressure melt ejection, but the conditions are very different from those at Chernobyl.

The Chernobyl accident underlines the importance of a high-integrity reactor containment for limiting activity releases following severe accidents. However, it is doubtful whether any containment could have resisted the loadings caused by the chemical explosions in the Chernobyl accident.

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14

Safety Improvement

Nuclear power plant safety is constantly scrutinized by the utilities, the supervisory agencies and the mass media. Modifications for improving plant safety are implemented as a result of operating experience and safety review. Occasionally problems arise which are common to a particular type or class of reactor. Some of these “generic” issues are discussed in this chapter, for U.S. and Swedish conditions. This is followed by a review of provisions for risk reduction as a result of the Three Mile Island accident.

14.1 Generic Safety Issues

In 1978 the USNRC established a Programme for the Resolution of Generic Issues Related to Nuclear Power Plants (1401). The programme comprised the three steps:

- identification of problems,
- establishment of priorities,
- implementation of measures.

Some hundred issues were identified, of which seventeen were given highest priority as *Unresolved Safety Issues* (1402). The progress of the programme is reported annually to the U.S. Congress. It has been possible to resolve several issues by establishing new safety requirements and implementing the required changes. Additional issues are identified as a result of increasing operating experience, research results and safety reviews. Selected issues are presented in the following subsections.

14.1.1 Pipe cracking in BWR

The cracking of pipes belonging or connected to the primary system has been observed in U.S. boiling water reactors since the mid-1960s. The cracks, which mainly occur in austenitic stainless steel pipe welds, were first observed in 100–250 mm diameter piping, and later on also in larger pipes. The cracks are generally discovered during ultrasonic testing and by leakage

from penetrating cracks. The frequency of observed cracks has increased in proportion to the number of plants and the operating time.

The mechanism has been identified as *intergranular stress corrosion cracking* (cf 3.5.3). This type of cracking requires the interaction of three factors (1403):

- precipitation of a chromium carbide in the grain boundaries of the material, known as *sensitization*, which weakens the grain boundaries enabling the crack to extend;
- mechanical tension above the yield stress of the base material;
- presence of oxygen in the reactor coolant.

Sensitization mainly occurs in heat-affected zones during the welding of pipes and connections.

Welding can also cause high residual stresses which are added to the normal pipe strains. A relatively high oxygen content in the primary coolant system is characteristic of boiling water reactors in contrast to pressurized water reactors. Therefore, stress corrosion has only been observed in exceptional cases in the primary system of pressurized water reactors.

Crack growth occurs slowly and produces “leak-before-break” (cf 3.5.2). If not earlier, the crack is detected by the leakage, and corrective action can be taken before a break occurs. Pipe cracks are therefore not considered to be a major safety issue, but rather an operating and maintenance problem. However, the USNRC has on several occasions called for the shutdown of reactors for inspection of pipe cracking. Conditions have been prescribed for continued operation involving requirements of repair, improved methods for ultrasonic testing and leakage detection as well as long-term measures which eliminate the problem.

The development of remedies has focused on the basic conditions for cracking, for example the use of materials which are not as susceptible to sensitization, or of improved welding methods which do not result in high residual tensile stresses, or the addition of hydrogen to the feedwater to reduce the oxygen content in the coolant. The latter has been implemented in Swedish BWR units (1404).

The Swedish boiling water reactors were spared from stress corrosion cracking for a long time. This is considered to be due to the choice of a stainless steel material with low carbon content, which minimizes the susceptibility to sensitization. In spite of this, small leaks in tubes connected to the primary system of Ringhals 1 were detected in 1982 and shown to be due to intergranular stress corrosion. All the pipes in the systems concerned were replaced during the 1983 refuelling outage with pipes of a material with a still lower carbon content. Isolated indications of similar cracking have also been found in other Swedish reactors.

Another kind of crack in stainless steel piping has occurred in the connect-

ing pipeline between the feedwater system and the shutdown cooling system. Large areas with transgranular cracks were observed in non-sensitized material. They are caused by *thermal fatigue* (cf 3.5.3) due to the temperature fluctuations which occur when the hot (270°C) reactor coolant mixes with the cold (180°C) feedwater.

14.1.2 Steam generator tube integrity

The steam generators are the largest components in pressurized water reactors next to the reactor pressure vessel. Each steam generator is up to 20 m high and has a diameter of 3–4 metres. It contains several thousand thin-walled tubes of stainless steel, usually a chromium-nickel alloy, surrounded by a carbon steel shell (see Fig. 5.7). The tubes are rolled and welded onto a thick plate in the bottom head and supported by plates at intervals. The reactor coolant passes through the tubes, while the feedwater flows outside the tubes.

There is usually a thin oxide layer on the tube walls to protect the material against chemical attack. In certain conditions, the layer is penetrated which results in corrosion. Most corrosion attacks occur in stagnant areas such as immediately above the tube sheet and in the crevices between the tubes and the tube sheet/support plates. Impurities in the feedwater can collect in these areas and form a reactive sludge. Corrosion causes cracking or thinning of the walls, gradually leading to leakage and fracture. Since a leaky tube necessitates reactor shutdown, it is of vital importance to avoid corrosion and other phenomena which can threaten tube integrity.

Most pressurized water reactors have suffered from steam generator problems. Defective tubes are plugged to prevent leakage. To a certain extent, this can be carried out without power reduction since the steam generators are designed with a considerable excess heat transfer capacity. According to a review of steam generator operating experience (1405), about 2% of the almost 1.6 million tubes in service in the world had been plugged by 1982.

Figure 14.1 shows the cumulative number of defective tubes per reactor as a function of the operating time. Each point in the diagram corresponds to one reactor. The three lines represent different failure rates, i.e. percentage of failed tubes per number of effective operating years. The higher the failure rate, the higher the cost of forced outages, inspections and repairs. If the number of tube defects is greater than about 10%, it may be necessary to reduce the power or replace the steam generator. As of 1984, such replacements had been carried out in seven PWRs, worldwide, after 10–14 years of operation.

It can be seen that the data differ for reactors with the same operating time. Certain plants have experienced no failures at all for a period of up to 10 years, while others have had more than 20% defective tubes. Several

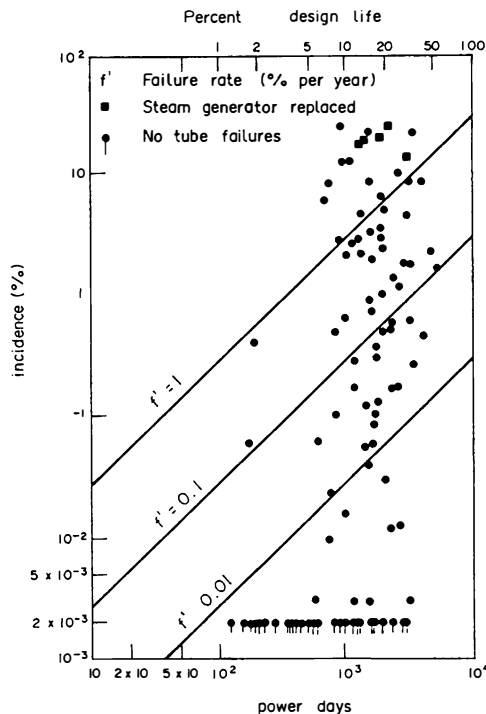


FIG. 14.1. Operating experience of PWR steam generators up to 1982. From O S Tatone, R S Pathania, Update on World-Wide Steam Generator Experience, *Nucl. Eng. Int.*, Vol 30, 1985

factors account for this: steam generator design, choice of material, water chemistry on the secondary side, type of cooling water (fresh, brackish or salt water), turbine condenser tightness, etc. In isolated cases, tube rupture has occurred during operation, resulting in loss of coolant and high release levels (cf Table 13.11). These events are mitigated by shutting down the reactor and isolating the damaged steam generator. If the safety systems function as intended, the environmental consequences will be negligible.

More than 90% of all defects have been caused by some kind of corrosion. At first, the most common kind of corrosion was *stress corrosion* from the secondary side due to alkali enrichment by local evaporation on the tube walls. During the mid-1970s, *wastage* caused by the attack of sodium phosphate posed a considerable problem. Sodium phosphate was added to the feedwater to reduce the chloride content and to counteract the general corrosion of heat transfer surfaces. As a result, many utilities changed to alkaline volatile treatment (AVT) of the feedwater. However this resulted in *denting*, i.e. the compression of tubes near the support plates due to corrosion in the crevice between the tube and the plate. By a combination of different methods, this type of degradation has been almost eliminated.

Alkaline stress corrosion has reappeared as a dominant cause of failure. In addition, another kind of intergranular attack is appearing on the *inside* of the tubes in areas with high mechanical stress, e.g. in U-bends and in tube-to-tubesheet welds. Other kinds of corrosion such as *corrosion fatigue* and *fretting corrosion* due to flow-induced vibration have also occurred.

It is evident that the problem is very complex. No fully effective remedy has as yet been found. By improving the design and using new materials it may be possible to avoid some of the tube degradation types so far observed. However, experience is still limited. As regards water chemistry on the secondary side, the tendency is towards the use of AVT and full-flow condensate polishing. With respect to turbine condenser tube material, there is a tendency to change from traditional copper alloys to the more corrosion-resistant titanium. The methods for inspection and repair of defective tubes have been considerably improved so that it should be possible to avoid tube rupture during reactor operation.

Each of the Swedish pressurized water reactors has three steam generators with vertical U-tubes of Inconel 600, two turbine condensers with tightwelded tubes of titanium and alkaline volatile feedwater treatment with partial flow condensate polishing. Ringhals 2, which started commercial operation in 1975, had condenser tubes of aluminum brass until 1979–80 and phosphate chemistry during the start-up period in 1974. After changing to AVT, denting was observed in 1977. As a preventive measure, about 200 tubes were plugged. From 1974 to 1980 condenser leakage was detected on a total of forty-two occasions which resulted in a high chloride content in the feedwater. Since the changeover to titanium tubes, no condenser leakage has occurred and denting has been arrested.

The first tube leakage in the Ringhals 2 steam generators occurred in 1979. Some sixty tubes were plugged as a preventive measure. Since then, further tube leakage has been observed, mostly in the tube sheet region due to crevice corrosion and stress corrosion cracking. In mid-1986 about one-third of the some 10,000 tubes had been plugged or sleeved. Since then the unit has been operated at 80% power. A decision has been taken to replace the steam generators in 1989.

After less than a year of operation with a new type of steam generator, a tube leak occurred in Ringhals 3 in October 1981. The leak was caused by mechanical fretting due to flow-induced vibration at the steam generator preheater inlet. This problem, which was also observed in Ringhals 4, was resolved through intensive development work carried out in a joint programme with utilities and the vendor.

Ringhals 3 and 4 have also experienced steam generator leakage due to stress corrosion cracking. Preventive measures are taken in the form of shot-peening of the inside of the tubes in the hot part of the tube-sheet region. In this way the mechanical stresses in the tube wall are reduced.

14.1.3 Pressure vessel thermal shock

The reactor vessel is normally in such a condition of pressure and temperature that brittle fracture cannot occur. This means that the base and welding materials are in the region of high fracture toughness above the brittle-to-ductile transition temperature (cf 3.5.2). If the temperature drops below the transition temperature at high reactor pressure, crack growth may occur. The risk is greatest in the part of the vessel surrounding the core. The risk increases with operating time since the transition temperature increases with the neutron fluence (time-integrated fast neutron flux).

There are two types of abnormal events which are of importance to reactor vessel safety:

- overcooling transients* when the vessel wall comes into contact with colder than normal coolant, i.e. is exposed to thermal shock;
- cold pressurization*, e.g. if the system pressure is increased too rapidly in connection with start-up.

Cold pressurization is avoided by careful adherence to prescribed procedures for reactor system heat-up from the cold shutdown state. Overcooling transients can occur during operation, for example when the emergency core cooling system is taken into operation in connection with a pipe break in the primary system, or as a result of a sudden increase of the feedwater flow.

An overcooling transient threatens the integrity of the reactor vessel when several factors interact:

- the transition temperature amounts to 100–150°C;
- there is a crack in the vessel which is large enough to propagate;
- the vessel comes into contact with cold water resulting in high thermal stresses and a wall temperature which falls below the transition temperature;
- the reactor pressure remains high or is increased from a lower level as the vessel temperature decreases.

Modern pressure vessel steel has a transition temperature of –20 to –10°C and which lies below 50°C even after long irradiation. The operating temperature remains well above the transition interval during the entire reactor lifetime. In some older reactor vessels with weld material containing impurities of copper and phosphorus, embrittlement occurs more rapidly. It is largely with respect to these older vessels that thermal shock can represent a limit to the service life.

For example, some U.S. pressurized water reactor vessels were found to have a transition temperature of 60–119°C after about 10 years of operation.

Also, embrittlement of the most exposed vessel welds was found to occur more rapidly than predicted in the Finnish Loviisa reactors (PWR). The fast neutron fluence at the vessel wall and hence the embrittlement rate was reduced by replacing a number of peripheral fuel assemblies with steel bundles.

The only reactor vessel in Sweden with material containing copper is Oskarshamn I. The surveillance tests at this plant show that the embrittlement proceeds at a rate which results in a predicted vessel lifetime of about 40 years.

By analysing reports on safety-related events, an attempt to identify precursors of overcooling transients was made in the USA (1406). Of a total of about 160,000 reports for forty-seven PWRs with a total of 329 operating years from 1963 to 1981, thirty-four events were considered significant with regard to thermal shock. Most of the transients were mild and only four events were considered serious. Two of these events are included in Table 13.11, namely Rancho Seco and Crystal River 3. In both cases, the loss of non-nuclear instrumentation resulted in erroneous signals which led to loss of coolant, safety injection and too rapid decrease of the reactor coolant temperature. However, the reactor vessel was not damaged.

14.1.4 Anticipated transients without scram

During certain transients it is essential for safety that the power be rapidly reduced, i.e. that reactor scram is successful. When scram does not occur as intended, this is known as an Anticipated Transient Without Scram (ATWS). The ATWS issue has attracted great interest in the USA. The debate has centred around whether the ATWS probability is low enough to warrant the exclusion of ATWS from the design basis.

A malfunction of the scram system can be electrical if the actuation signal fails, or mechanical, if one or several control rods fail to enter the core on receipt of a signal. More than two control rods must normally fail in order for scram to be ineffective. In pressurized water reactors, the control rods drop into the core by gravity when the magnetic coils holding the rods out of the core are de-energized. In boiling water reactors, the rods are pushed into the core from below by hydraulic pressure.

Automatic scram is considered to be very reliable. The Reactor Safety Study estimated the unavailability at about 1 per 20,000 demands. If the automatic system fails, scram can be initiated manually. There is also the possibility of shutting down the reactor by other means; in PWRs by boron injection, and in BWRs by reducing the speed of the main recirculation pumps so that more steam is produced in the core, which makes the reactor subcritical. In Swedish BWRs, it is also possible to motor the rods into the core by the fine-motion control rod system. Both fine-motion control rod insertion and recirculation pump runback are automatically initiated on

receipt of a scram signal. As an extra precaution, boron can be injected into the primary coolant by manual actuation.

Because of the severe consequences of certain anticipated transients without scram, the USNRC suggested several means for improving safety in such events (1407). The aim was to reduce the estimated contribution of ATWS to the core damage frequency to about one in a million reactor years. This can be achieved in two ways: by increasing the reliability of the scram system or by reinforcing the possibilities of alternative methods for reactor shutdown.

Vendors and utilities in the USA have questioned whether the tightening of requirements was necessary and justified. The probability of ATWS was considered so low that such events were not believed to represent a safety issue (1408). However, some incidents have occurred (see Table 13.11), which indicate that scram system reliability may be less than previously thought.

Final requirements on risk-reducing measures were set down by the NRC in 1984. The rules specify that pressurized water reactors must be equipped with independent and diversified systems for both the actuation of scram and the initiation of the auxiliary feed water system and turbine stop valve closure. Similar requirements for the actuation of scram and recirculation pump runback were prescribed for boiling water reactors. An increased capacity of the boron injection system was also required for these reactors.

U.S. experience and requirements are not directly applicable to Swedish boiling water reactors due to differences in design. The Swedish safety studies indicate a very low core damage frequency for ATWS events, e.g. about 3×10^{-7} per reactor year for Ringhals 1. No special requirements for improving safety in ATWS events have been proposed in Sweden.

14.1.5 Station blackout

Station blackout is defined as the complete loss of AC electric power. Since many systems required for core cooling, decay heat removal and containment cooling depend on AC power, the consequences of station blackout are severe. In fact, station blackout is a major contributor to the estimated core damage frequency in many cases, for example by causing leakage of the main coolant pump seals in PWRs, and containment pool heat-up in BWRs. Station blackout may also include loss of AC power to safety-related equipment supplied by the DC/AC converters, if the battery system fails.

Operating experience in the USA indicates that a loss of offsite power occurs about once per 10 site-years, Table 14.1. The typical duration is of the order of one-half hour. However, at some power plants the frequency of offsite power loss has been substantially greater than the average, and at

TABLE 14.1. *Total loss on offsite power at U.S. nuclear power plant sites, from 1968 to 1983*

Causes of loss of offsite power	Number	Frequency of occurrence (per site-year)	Median duration (hours)
Plant-centred	30	0.056	0.3
Grid blackout	10	0.019	0.7
Severe storm	6	0.011	2.6
Total	46	0.086	0.5

Source: *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, USNRC Report NUREG-1032, U.S. Nuclear Regulatory Commission, January 1985

TABLE 14.2. *Diesel generator availability at U.S. nuclear power plants. Number of diesel generator years: 450*

Category	No. of demands	No. of failures	Failures/demand	No. of auto start failures	Auto start failures/demand
Test	13,665	253	0.019	55	0.004
Loss of offsite power	100	5	0.05	3	0.03
All emergency demands	539	14	0.026	5	0.009

Source: *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, USNRC Report NUREG-1032, U.S. Nuclear Regulatory Commission, January 1985

other plants the duration of the power outages has greatly exceeded the average.

During loss of offsite power events, on-site emergency AC power sources were available to supply the power needed by vital safety equipment. However, in some instances one of the redundant energy power supplies was unavailable, and in a few cases there was a complete loss of AC power. During these events, power was restored in a short time without any serious consequences. As shown in Table 14.2, there have been numerous instances at operating plants in which emergency diesel generators failed to start and run during surveillance tests.

A U.S. study (1409) summarized the characteristics of station blackout events in the USA as follows:

- The estimated station blackout probability ranges from approximately 10^{-5} to 10^{-3} per reactor year.
- The capability of restoring offsite power in a timely manner has a significant effect on accident consequences.
- The estimated core damage frequency for station blackout events ranges from approximately 10^{-6} to 10^{-4} per reactor year.

The study proposed a rule for the resolution of the station blackout issue, based on the expectation that the core damage frequency from station blackout could be maintained around 10^{-5} per reactor year or lower. To reach this level, a plant would have to be able to cope with station blackout at least 4 and perhaps 8 hours long and have emergency diesel availabilities of 0.95 per demand or better, with relatively low susceptibility for common cause failures.

Many PWRs and BWRs are provided with a steam-driven auxiliary feed-water pump. If battery power is also available, these plants can withstand station blackout for several hours. In addition, it is essential that adequate procedures and training for the rapid restoration of AC power are ensured, and that improved methods for diesel generator operations and maintenance are developed and implemented.

Outside the USA, plant modifications have been introduced in several countries to cope with station blackout. French PWRs, for example, have been provided with a special steam-turbine driven generator which supplies power to the high-pressure seal injection pumps and the battery chargers. In German PWRs, additional auxiliary feedwater pumps with a dedicated diesel generator have been installed in a separate bunkered building. In Sweden, the Ringhals 1 BWR has been equipped with a special coolant make-up system with a dedicated diesel generator.

14.2 Impact of the Three Mile Island Accident

The Three Mile Island accident resulted in a major effort worldwide to review existing plant designs and reassess potential risks to the public. Two weeks after the accident, the President of the United States appointed a commission to analyse the accident and its consequences and to propose measures to raise the level of safety. The USNRC formulated a detailed plan of action. Already a week after the accident, the Swedish Nuclear Power Inspectorate proposed certain modifications of Ringhals 2, the only pressurized water reactor in operation in Sweden at that time. The Swedish Government appointed a committee to re-evaluate the overall risks associated with reactor operation.

14.2.1 The Kemeny Report

The President's Commission on the Accident at Three Mile Island, called the Kemeny Commission after its chairman, submitted its report in October 1979, about 7 months after the accident (1410). The report confirmed that the actual release of radioactive substances was negligible and that the main health effect was mental stress. The fundamental message was the importance of the human factor to reactor safety. It was considered that plant equipment had performed well enough for the accident to have become

only a minor incident if human error had not been involved. The general conclusion was that while plant equipment could and should be improved, basic safety issues are closely connected with the people who operate the plants and the role, procedures and attitudes of the plant vendors, utilities and supervisory bodies.

According to the Commission, the reactor designers, operators and supervisors had been lulled into the belief, after many years of accident-free nuclear power plant operation, that the plants were safe enough. The USNRC had established a comprehensive system of rules and regulations which, if complied with, were considered a guarantee of safety. The Commission found that the regulations focused too much on the technical equipment and not enough on the human factor.

According to the Commission, the prevailing safety philosophy concentrated too heavily on design basis accidents such as large pipe break in the primary system. If these very improbable "worst" events could be mitigated, it was believed unnecessary to analyse other, more likely but small events in detail. Large breaks require rapid and automatic execution of safety functions. Small events, on the other hand, generally occur more slowly and often require human mitigative action. TMI-2 was an example of how an originally harmless incident can develop into a severe accident through human error.

The conclusion of the Commission was that a change in the attitude towards safety was required by plant operators, utilities, vendors and authorities. The deterministic safety approach and the fixation on design basis accidents should be supplemented by a more diversified safety analysis. A general recognition of the fact that severe accidents can occur should permeate all stages of safety work. The man-machine interface should be improved, e.g. in the design of the control room so as to improve the possibility of the operator to identify potential accident sequences and adopt countermeasures.

The Commission considered that operator training at TMI-2 had been deficient, that the procedures for dealing with abnormal events had been unclear and that lessons had not been learnt from earlier similar incidents. This led the Commission to generally advocate improvements in the training of operating and maintenance personnel, the formulation of adequate operating rules for accident situations and the systematic collection, evaluation and feedback of operating experience.

While the focus of safety work should remain on preventive action, the Commission felt that more attention should be paid to mitigating the consequences of an accident, should an accident arise.

Both internal and external emergency preparedness should be reinforced. The public's rights to information should be better complied with than in the TMI-2 case.

It should be noted that the findings and recommendations of the Kemeny

Commission were applicable to the U.S. situation and are not necessarily relevant to other countries.

14.2.2 The TMI Action Plan

Immediately after the accident, the NRC closed down five U.S. pressurized water reactors of the same design as that of TMI-2. After implementation of certain measures, the reactors were placed into operation again. The sister unit, TMI-1, was restarted in 1985. Clean-up operations were started on TMI-2 (see 13.5.4). This work is expected to be finished in 1989 and is estimated to cost about one billion dollars. The recovery plan aims at future use of the plant.

The NRC immediately launched an investigation which resulted, as soon as 4 months after the accident, in comprehensive proposals for risk-reducing measures (1411). Based on this investigation and the recommendations of the Kemeny Commission, a detailed action plan was prepared which covered a broad spectrum of measures and requirements for plants already in operation as well as for new plants (1412). The actions were grouped into the following task areas:

- I Operational Safety.
- II Siting and Design.
- III Emergency Preparedness and Radiation Effects.
- IV Practices and Procedures.
- V NRC Policy, Organization and Management.

The items within Task I aimed at reducing the number of events which could result in accidents and at improving the possibility of the operators identifying such events and adopting corrective action. Among the prioritized actions were:

- improved operator training,
- upgraded requirements on control room manning,
- new guidelines for control room layout,
- procedures for experience feedback.

Task II comprised both long-term and short-term action. Short-term improvements were required for:

- equipment for the ventilation of non-condensable gases from the primary system,
- plant shielding to provide access to vital areas and protect safety equipment for post-accident operation,
- post-accident sampling in the primary system and reactor containment,

- instrumentation for monitoring accident conditions.

Long-term action included:

- development of improved methods and equipment for controlling the formation of hydrogen in the containment and for minimizing the risk of hydrogen explosions,
- probabilistic safety analyses on specific plants to provide a basis for selecting measures for improving safety.

The President's Commission recommended centralized external emergency preparedness planning which would be carried out by a special federal organization in co-operation with federal and local bodies. This measure was adopted in 1979 and, as a result, Task III in the NRC Action Plan largely dealt with internal emergency preparedness and radiation protection. Tasks IV and V were specific to the NRC.

As a result of the TMI Action Plan, numerous modifications to U.S. light water reactor plant designs and operating procedures have been made. Major programs were begun to reassess the role that severe accidents could have in NRC's regulatory process. The NRC developed and issued a Severe Accident Policy Statement (1413) followed by an Implementation Plan (1414). This plan provides for the resolution of severe accident issues through a systematic examination of plants by industry for risk contributors, and the regulatory use of improved source terms information.

14.2.3 The Swedish Reactor Safety Investigation

The Swedish Reactor Safety Investigation Committee was appointed in 1979 and submitted its final report 7 months later (1415). Based on an independent examination of the accident sequence at TMI-2 and an analysis of the safety in Swedish reactors, the investigators arrived at a number of findings and conclusions. These findings led to a series of forty-nine recommendations under the following headings:

—Roles and Responsibilities

The main task of the supervisory agencies should be to provide goals for the safety work of the utilities and to evaluate their organization and methods for achieving these goals.

—Design and Construction

Probabilistic methods should be used in the assessment of safety. Special analyses should be carried out for each plant.

—Consequence Mitigation

The risk of accidental off-site releases should be reduced beyond the level of protection provided by the existing reactor containments.

—*Man–Machine Interaction*

Measures should be adopted to reduce the risk of human error, for example by facilitating operator action in stress situations.

—*Recruiting and Training*

Training should be broadened to include maintenance personnel and to place more emphasis on operational disturbances and accident situations.

—*Normal Operation*

Normal operation was found to be satisfactorily regulated by the Technical Specifications for reactor operation, but the supervisory agency should formulate requirements for the quality assurance work carried out by the utilities.

—*Emergency Preparedness*

The on-site emergency plans should be reviewed with regard to organization, staffing and training.

—*Feedback of Experience*

An improved system for the systematic gathering, review, analysis and feedback of operating experience should be set up in co-operation between the utilities, the supervisors and the vendors.

—*Reactor Safety Research*

Research should be intensified, for example on human reliability and measures for limiting radioactive releases.

Most of the proposals were put into action. The decision in 1981 by the Swedish Government to install a system for filtered venting of the Barsebäck reactor containments deserves special mention. This project is described in 14.3.2.

14.3 Plant Modification

Modifications of existing plants to reduce the accident risk might be broadly grouped into preventive changes and mitigative changes. A preventive change is one that reduces the frequency of core damage. A mitigative change is one that reduces the accident consequence. Some important features have both preventive and mitigative function; a few can be positive in one respect and negative in another.

Probabilistic risk analysis makes possible a quantitative assessment of risk-reducing changes. The fundamental approach taken is to examine the benefits and costs of any risk-reducing option. The benefits are expressed as averted accident costs, i.e. the benefits are monetized for comparison with the costs.

The following subsections give examples of modifications undertaken in Swedish nuclear power plants.

14.3.1 Preventive changes

The oldest Swedish unit, Oskarshamn I, has been in commercial operation since 1972. Forsmark 3 and Oskarshamn III were commissioned in 1985. This means that plant designs are based on safety requirements which have developed over a decade. During this time, the safety requirements have been successively sharpened. Changes have been made in the older plants in order to raise their level of safety to that of the new plants. This is known as *backfitting* or retrofitting.

Table 14.3 presents some examples of preventive backfitting. It has largely been possible to implement the changes during planned outages, and the plant load factor has only been slightly affected.

TABLE 14.3. *Examples of backfitting in Swedish reactors*

Plant	Modification	Year of completion
All BWR	Change of spray nozzles for emergency core cooling	1974
Ringhals 1 and 2	Improvement of sea water intake	1975
All plants	Improvement of physical protection	1976
All BWR	Installation of back-flushing system for the emergency core cooling water strainers in containment pool	1977
All BWR	Reinforcement of equipment in containment pool	1978
All PWR	Replacement of thermal insulation of high energy piping	1979
Oskarshamn 1	Installation of backup system for power supply to safety-related equipment	1980
All LWR	Replacement of components and instruments to improve durability and increase measuring range during accidents	1980
All plants	Implementation of alternative means of residual heat removal	
Forsmark 1 and 2	Replacement of bolts for securing fuel assembly guide rails	1982
All BWR	Change of blowdown pipe outlet geometry to reduce dynamic forces in containment pool	1983
Ringhals 1	Change of stainless pipes connected to reactor main coolant system	1983
Ringhals 3 and 4	Modification of feedwater inlet to steam generators	1983

14.3.2 Mitigative changes

According to the proposal by the Swedish Reactor Safety Investigation for increased efforts to limit radioactive releases, a research project, called FILTRA, was carried out from 1980 to 1982 (1416). A study was made of the possibility of reducing the offsite consequences of accidents involving high pressure in the reactor containment, by the combination of two functions:

- pressure relief of the reactor containment through a “safety valve” which opens before the failure pressure is reached;
- filtering of escaping steam and gas for the removal of any radioactive particulates.

The study showed that a good filtration effect and steam condensation could be achieved in a large volume gravel bed.

In 1981 the Government decided that the two reactor containments of the Barsebäck power plant should be equipped with a common filtered venting system. The FILTRA plant was placed into operation in November 1985. It consists of a gravel bed condenser with a 10,000 m³ volume, connected to the wetwell of each containment via a large vent line (1417) (Fig. 14.2). The gravel bed condenser is normally isolated from the containment by a rupture disc for which the burst pressure is set at 0.65 MPa, which is 0.15 MPa above the containment design pressure. There are also two small pipes which connect the gravel bed condenser to the drywell via two isolation valves in series, which are normally closed. These pipes allow for depressurization even if the containment is partly filled with water or if manual depressurization is initiated before the containment pressure reaches the set point of the rupture disc. The gravel bed is vented via an off-gas line to the stack.

After the rupture disc there are two shut-off valves in series which are normally open. The flow of steam and gases to the FILTRA plant is distributed in the upper layer of the gravel bed. When the steam and gases flow downwards into the gravel column, steam condenses on the initially cold pebble surfaces. The condensate is collected in the lower part of the condenser. The inner surfaces of the condenser have a steel liner. The vessel is filled with nitrogen to prevent hydrogen combustion and growth of organic material in the gravel bed.

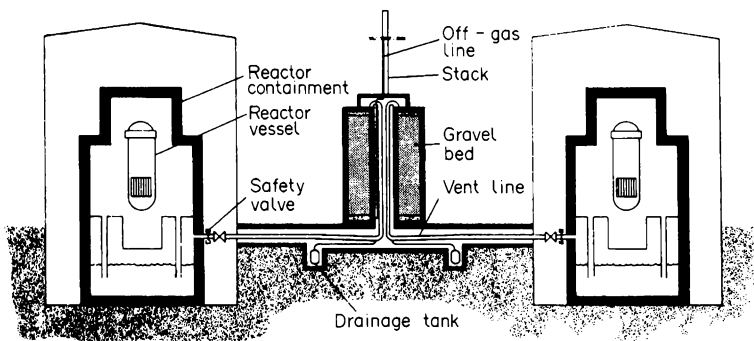


FIG. 14.2. Schematic layout of FILTRA

FILTRA is designed so that 99.9% of all radionuclides in the core (except noble gases) are retained in the reactor containment and the gravel condenser after a severe core damage accident. The plant is designed to function passively for 24 hours during the accident. The single failure criterion is applied (except for the rupture disc) and the plant is designed to withstand a ground acceleration of 0.15 g during an earthquake.

The safety analysis for FILTRA showed that the venting precludes containment overpressure which greatly reduces risk in Barsebäck-type reactors. The filtering provides additional risk reduction for events which also involve core melting. On the other hand, FILTRA does not provide any risk reduction for core melt sequences which do not result in high containment pressure.

The government decision in 1981 also established that mitigative measures should be implemented in other nuclear power plants before 1989. Therefore a research project, called RAMA, was undertaken in co-operation with the Nuclear Power Inspectorate and the utilities. The aim of the research project was to provide a design basis for containment behaviour and source term analysis during severe accidents. Some of the results are presented in Chapter 11.

Based on the results of the research project and of design studies by the utilities, in 1985 the Nuclear Power Inspectorate proposed an action plan for mitigative plant modification in Forsmark, Oskarshamn and Ringhals. The plan suggested that all reactor containments should be equipped with pressure relief devices. In addition, it was recommended that Forsmark type BWRs with annular condensation pool (see Fig. 4.7) should have equipment for flooding the lower drywell in severe accident situations and special reinforcement of vulnerable penetrations and load-bearing parts. The proposal was based on the same requirements as those of Barsebäck, namely that accidental releases to the environment should be kept below about 0.1% of the radionuclide inventory, excluding noble gases, in a core of approximately 1800 MW thermal output.

In 1986 the Government agreed on the proposal. The technical solution adopted is based on the use of an improved containment spray system and a filtered venting system (1418). The filter is a new design, a submerged multi-venturi scrubber.

The improved containment spray utilizes the ordinary spray water penetrations and nozzles. Outside the containment, connections are made to the plant's fire protection system. Hence, spray can be initiated using any of three direct diesel-driven pumps in the fire protection system without having to rely on auxiliary power. Spray is initiated manually, and it is predicted that spray start will be needed in a time interval of 5–8 hours after the beginning of a severe accident, depending on the particular sequence. The spray system is also able to flood the containment to above the original core level.

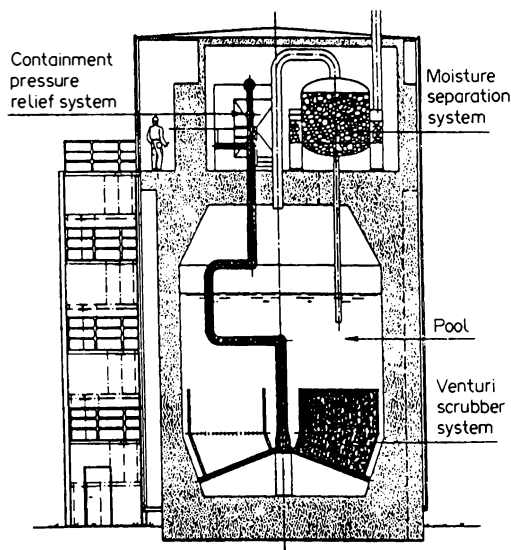


FIG. 14.3. Filtered containment venting by the Multi Venturi Scrubber System.
Courtesy AB Asea-Atom

The vent filter system is capable of acting as an alternative depressurization device, passively initiated by a rupture disk, should the spray not come into operation. It is otherwise needed only to discharge the compressed atmosphere following containment flooding. The vent line connects to the drywell.

The Multi Venturi Scrubber System (MVSS) (Fig. 14.3) is a design previously used for flue-gas cleaning. The containment pressure drives the venturis, which are submerged in a water pool, also acting as an iodine trap. The number of venturis utilized is determined by the static pressure in the header, which allows each venturi to operate close to optimal conditions. The MVSS water volume is 200–300 m³ for the BWR plants and about 500 m³ for PWR plants, as compared to the 10,000 m³ gravel bed volume for the FILTRA system.

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15

Reactor Safety Research

In the early days safety research went hand in hand with reactor development and design. Later on independent research programmes were initiated by the regulatory agencies. During the 1970s the emphasis was placed on the verification of design criteria for the emergency core cooling systems and the reactor containment. In terms of cost, the research programmes were dominated by large-scale thermohydraulic experiments simulating large LOCA. As operating experience accumulated, research was more and more directed to operational safety and accident prevention. After TMI-2, substantial efforts were devoted to the study of core melt accidents, containment behaviour and consequence mitigation. This chapter highlights reactor safety research within the major areas, with examples mainly from U.S. and Swedish research programmes.

15.1 Heat Transfer and Fluid Flow

The emergency core cooling systems are designed to prevent core overheating after a postulated large pipe break in the main coolant system, i.e. during large LOCA. Between 1971 and 1973 the USNRC established licensing requirements which are also applied in many other countries (see 9.2.1). A principal aim of the research was to develop calculational methods for LOCA analysis and to verify that the licensing requirements are fulfilled. This requires a thorough understanding of the thermohydraulic processes in the primary system and the reactor containment as well as of the fuel behaviour during accident conditions.

15.1.1 Thermohydraulics

Thermohydraulic experiments and modelling have concentrated partly on studying separate effects, and partly on integral experiments and calculational methods where the entire sequence of blowdown, refill and reflood is simulated (Fig. 15.1). Separate effects are studied in test facilities with electrically heated fuel bundles simulating real fuel assemblies. Correlations of heat transfer and fluid flow parameters have been developed which make it possible to predict critical heat flux and post-dryout heat transfer.

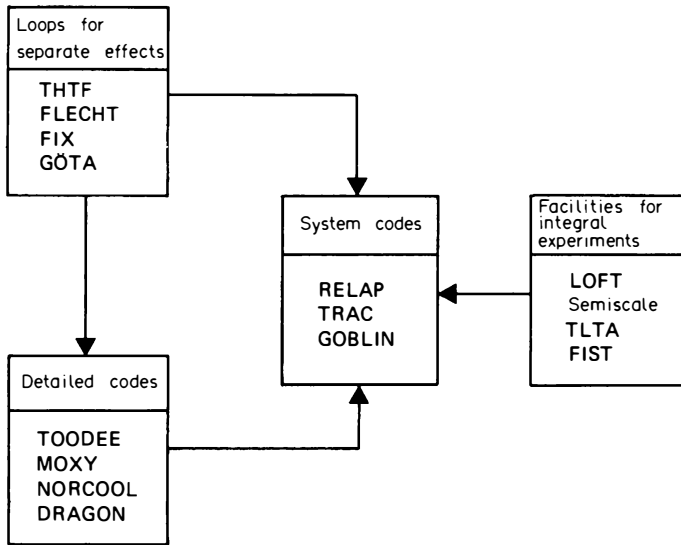


FIG. 15.1. LOCA experiments and modelling with examples of U.S. and Swedish test facilities and computer codes

The time to critical heat flux during blowdown and the heat transfer during subsequent boiling have been studied in the THTF loop in the USA for PWR conditions (1501). Rewetting and heat transfer during the reflood phase were studied in FLECHT (1502). For BWR conditions, the time to dryout and heat transfer during post-dryout were tested in FIX (Fig. 15.2) (1503). The clad temperature history after the initiation of spray cooling was investigated in the GÖTA loop (1504).

The experimental results are used to determine the cladding-to-coolant heat transfer coefficient during the various stages of blowdown and emergency core cooling. If the heat transfer coefficient is known, the fuel and clad temperature can be calculated, e.g. with the computer code MOXY for boiling water reactors and TOODEE for pressurized water reactors. The codes NORCOOL and DRAGON, indicated in Fig. 15.1, were developed in a joint Nordic project and by Asea-Atom, respectively, and are used to calculate the coolant state and the heat transfer coefficient during emergency core cooling in a BWR coolant channel.

Special codes have been developed to describe the thermohydraulics of the entire primary system during LOCA. Examples of such system codes are RELAP and TRAC which were produced in the USA for both pressurized and boiling water reactors. Versions of these codes, adapted to Swedish reactors, are also available in Sweden (1505). Asea-Atom have developed an independent system code, GOBLIN, for their boiling water reactors.

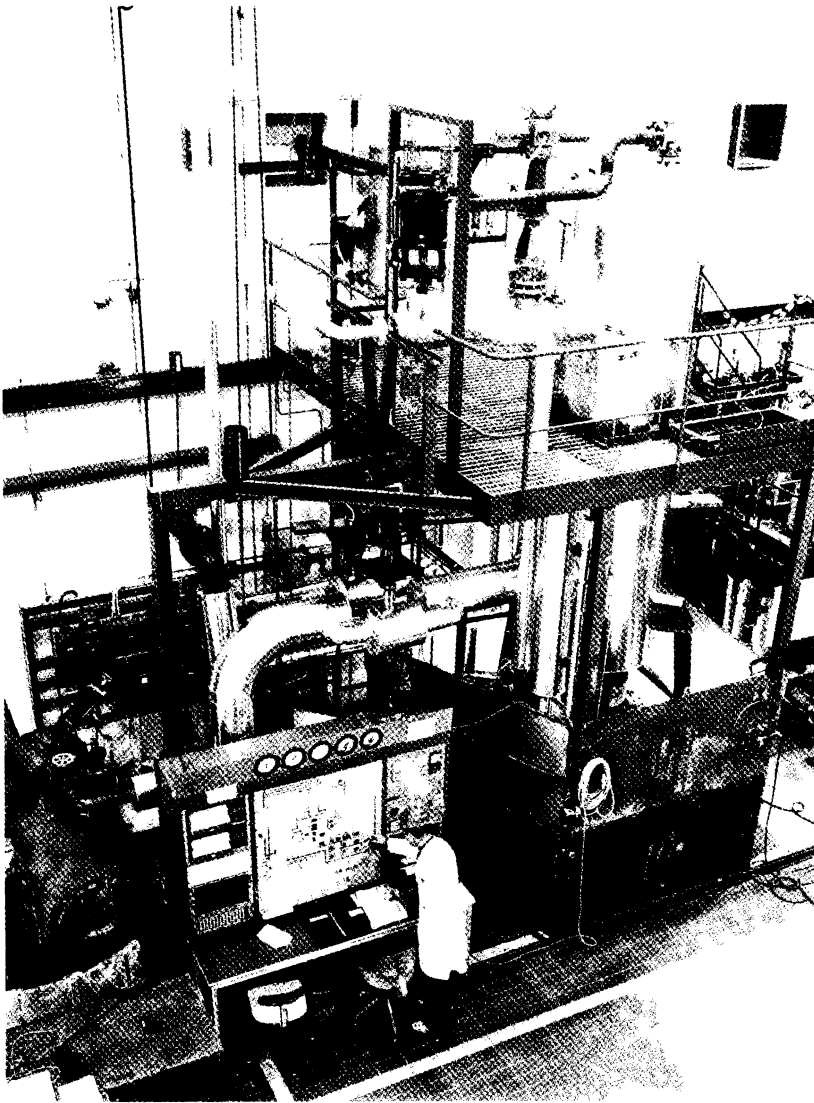


FIG. 15.2. The FIX loop in the Studsvik thermal laboratory

15.1.2 Integral experiments

Integral experiments, which simulate entire LOCA and transient sequences, are performed in order to verify the licensing requirements and validate the computer codes. Experimental facilities in the USNRC's LOCA programme have included two facilities for pressurized water reactors: LOFT (Loss Of Fluid Test) and Semiscale, located at the Idaho

National Engineering Laboratory (INEL), and two boiling water reactor experimental loops: TLTA (Two Loop Test Apparatus) and FIST (Full Integral Simulation Test) at General Electric's laboratories in California.

LOFT was a 55 MWth pressurized water reactor in a 1:5 model of a full-scale reactor. In the USNRC LOFT programme some thirty LOCA and transient experiments with nuclear heating were carried out during 1978–82. The experiments on large LOCA show that after early DNB during blow-down rewetting is rapidly obtained due to the flow maintained by the main coolant pumps (Fig. 15.3). Cooling during subsequent reflooding is more efficient than assumed in the calculational models prescribed for licensing. This means that the margin to the critical clad temperature, 1204°C (2200°F), is several hundred degrees.

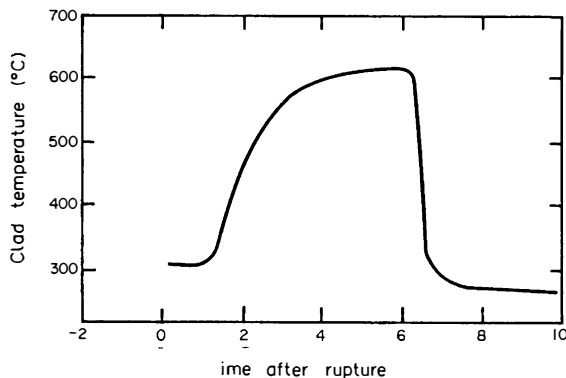


FIG. 15.3. Schematic diagram of the measured clad temperature during a large LOCA in LOFT (Experiment L2-3). From M L Russel, Loss-of-Fluid Test Findings in Pressurized Water Reactor Core's Thermal-Hydraulic Behaviour, in *Proc. on Nuclear Reactor Core's Thermal-Hydraulics*, Vol 1, American Nuclear Society, 1983

Eight additional integral experiments with nuclear heating were carried out in the OECD LOFT programme during 1983–5, including two experiments with significant fuel damage. The last experiment was designed to provide information on the release and transport of fission products and fuel aerosols in a severe accident, simulating a V-LOCA with ineffective emergency core cooling, where cladding temperatures reached 1800°C and above.

While LOFT had nuclear heating, other test facilities have used electrically heated rod bundles to simulate fuel assemblies. LOCAs initiated by steam generator tube rupture were simulated in Semiscale. The most unfavourable response, i.e. the highest cladding temperatures, was obtained after a rupture of between twelve and fifty tubes.

Semiscale was also used to investigate alternative methods of supplying emergency core cooling water to the pressurized water reactor. An effective method, which “quenches” the core quickly, was demonstrated to be the injection of water into the region below the core rather than into the cold leg of a main cooling loop as is usually done.

Large LOCA integral experiments for U.S. boiling water reactors, where part of the primary system flow is recirculated by external centrifugal pumps and part by internal jet pumps, have been carried out in the TLTA loop. A large margin was observed in the peak clad temperature as compared to the results of licensing calculations (1506). It was shown that countercurrent steam flow in the inlet of the coolant channels is important for delaying the loss of coolant in the channels during blowdown and for rapidly refilling the channels by the low-head safety injection system. The USNRC have approved a LOCA analysis model, developed by General Electric, predicting a 250–500°C lower peak clad temperature than the original licensing models.

Once the essential thermohydraulics during large LOCA had been determined, the integral experiments focused on small LOCA and transients, involving loss of feedwater, recirculation pump trip, etc. Such events have been simulated in Semiscale and LOFT for pressurized water reactors. The results show that natural circulation is sufficient to transfer the decay heat to a steam generator even if most of the primary coolant is lost. Heat transfer then takes place by steam condensation and reverse flow of the condensate to the core. Cooling by natural circulation in the reflux condenser mode has also been demonstrated in the West German PKL loop (1507).

Small LOCA in jet pump boiling water reactors have been simulated in the FIST loop in the USA and in ROSA-III in Japan. In boiling water reactors, the clad temperature variation exhibits a similar shape during large and small LOCA (Fig. 15.4). This is because small and medium breaks threatening to uncover the core are intentionally transformed into large “breaks” by automatic depressurization (see 9.4.3). The size of the break changes the time to dryout and rewet, but not the phenomena as such or the form of the clad temperature curve.

15.1.3 Fuel behaviour

Fuel behaviour during LOCA and transients is affected by many factors (Fig. 15.5). Maximum values of clad temperature, clad oxidation and hydrogen gas formation as well as requirements on core heat removal are established in the licensing criteria (see 9.2.1). Assumptions and models for licensing calculations are intended to give results on the safe side. Such calculations can be carried out with the previously mentioned TOODEE and MOXY codes (Fig. 15.1).

Measurements of clad oxidation in steam at temperatures in the range

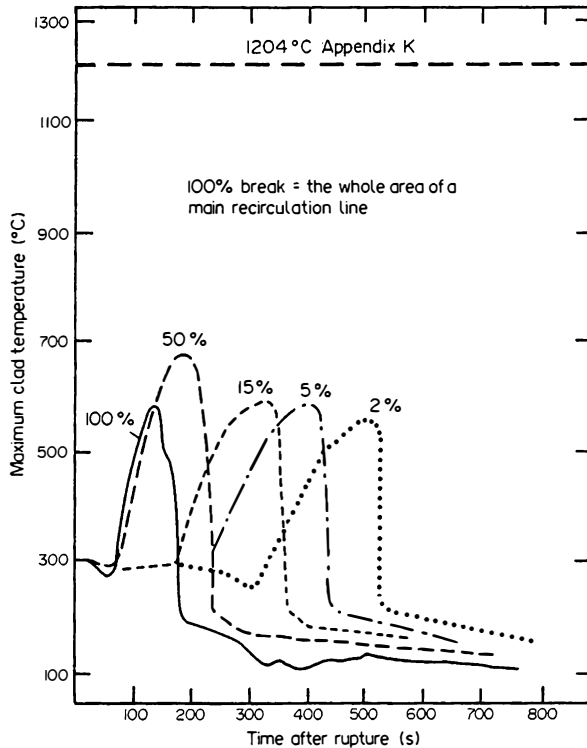


FIG. 15.4. Clad temperatures for various break sizes during simulated LOCA in jet pump boiling water reactors. Experiments in ROSA-III. From M Shiba et al, *Small-Break LOCA Experiments in ROSA III*, Paper IAEA-CN-36/39 at Int. Conf. on Current Nuclear Power Plant Safety Issues, Stockholm, 20-24 October 1980

700–1400°C have shown that the maximum oxidation rate is about 25% lower than assumed in the original licensing model (1508). Clad creep in high temperature steam has been studied at Studsvik and elsewhere, and a calculational model has been developed (1509). Tests in the materials testing reactor PBF (Power Burst Facility) at INEL show that clad deformation and oxidation are generally moderate during LOCA. The creep rate is influenced by the gas pressure in the gap between the cladding and the pellet. During certain conditions, a kind of unstable clad swelling occurs (“ballooning”) which may block the coolant flow and lead to clad failure. Another possible failure mechanism is brittle fracture from thermal shock when the oxidized hot cladding is rewetted during the reflood phase.

The gas pressure in the gap, the fuel swelling and the clad deformation affect the heat conductance of the gap and hence the temperature and the stored heat in the fuel. The GAPCON code, developed in the USA, is used to calculate these and other fuel parameters during steady state conditions

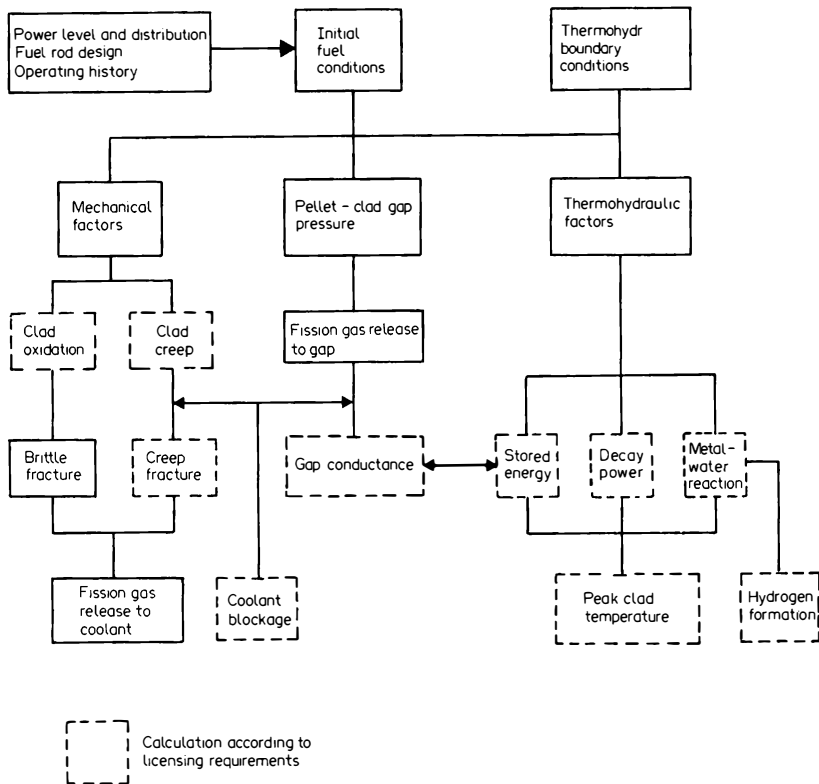


FIG. 15.5. Factors affecting fuel behaviour during LOCA and transients

valid at the onset of a LOCA. The code has been validated by comparison of calculated and experimental results of the gap conductance under various conditions (1510).

If core cooling ceases, the heat stored in the fuel is redistributed. The fuel and clad temperatures will equalize at a rate determined by the time constant of the fuel rod, which is about 5 seconds. Even if the reactor is rapidly shut down and the fission power cut off, the clad temperature will rise several hundred degrees because of this redistribution.

Heat continues to be generated in the fuel due to fission product decay even though the nuclear chain reaction has stopped. The decay heat decreases with time. A standard curve based on measurements carried out in the 1950s with a 20% allowance for uncertainties was established for licensing calculations. New measurements (1511) have shown that the decay heat is lower for short cooling times than indicated by the standard curve and that the uncertainty is generally less than previously assumed. A new standard for decay heat has therefore been adopted in the USA (see 3.4.5).

15.1.4 Containment behaviour

In the event of a large pipe break in the primary system (DBA-LOCA), a large amount of steam will escape and result in a rise of the containment pressure. The containment is designed to withstand the maximum pressure during DBA-LOCA. The pressure increase in the large dry containment of a PWR is limited by the large volume of the containment. In the BWR, the pressure increase is suppressed by discharging the escaping steam to the containment condensation pool.

Special computer codes have been developed for the calculation of containment pressure and temperature during DBA-LOCA and similar events. COPTA (1512) is such a code, developed at Studsvik and validated by comparison with results from full-scale experiments in the Marviken facility. COPTA can be used for both large, dry containments and pressure suppression containments.

The Marviken experiments were conducted from 1972 to 1982. The aim of the first series of experiments was to study the pressure and temperature conditions during blowdown in a pressure suppression containment. The effects of the energy content in the water and the steam in the reactor pressure vessel, the location and size of the simulated pipe break, the temperature of the condensation pool and the depth of vent pipe submergence in the condensation pool were investigated (1513). In the second test series, the dynamic processes in the blowdown lines and the condensation pool were studied in greater detail (1514). These phenomena include pressure oscillations and pressure surges through the compression of non-condensable gases in the blowdown pipes and their subsequent expansion in the condensation pool or through unstable gas condensation.

The magnitude of the break flow is important for the progression of a DBA-LOCA. When the flow velocity reaches the speed of sound, which cannot be exceeded, *critical flow* conditions are obtained. The aim of the third series of Marviken experiments was to determine the critical mass flow rate of a two-phase mixture of steam and hot water from large diameter pipes (1515). The mass flow rate was shown to be 5–20% lower than that prescribed for licensing calculations.

The force of the water jet from the break can result in damage to equipment in the containment. The effects of large-scale two-phase jet impingement were studied in the fourth Marviken experiments (1516).

15.1.5 Licensing requirements

Traditional licensing calculations for LOCA analysis are performed with conservative versions of computer codes which have been approved by the regulatory agencies (cf 9.3.1). As previously noted, the assumptions in these codes may be over-conservative for several reasons:

- the decay heat is about 20% lower than assumed;
- the clad oxidation rate is about 25% lower than predicted with the prescribed recipe;
- rewetting of the fuel rods seems to occur even in the blowdown phase, which is not credited in the licensing models;
- the heat transfer from cladding to coolant during refill is higher than predicted with the approved correlations;
- the break flow is up to 20% lower than predicted with currently approved formulae.

Best-estimate models which draw on the improved theoretical and experimental basis available since the adoption of the 10 CFR 50 Appendix K licensing models, result in several hundred degrees lower peak clad temperatures (Fig. 15.6). It should therefore be possible either to modify the licensing requirements or replace the original licensing models with more realistic models, the results of which can be evaluated by comparison with experiment (Fig. 15.7). Realistic models could also be applied to small and medium LOCA for which it is sometimes difficult to determine whether or not the Appendix K models (which are primarily applicable to large LOCA conditions) give results on the safe side.

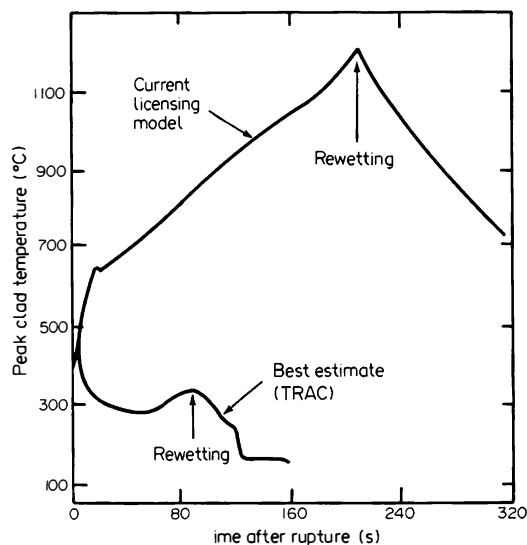


FIG. 15.6. Comparison of calculations with licensing and best-estimate (TRAC) models for a large LOCA in a U.S. boiling water reactor. From G E Dix, BWR Loss of Coolant Technology Review, *Proc. on Nuclear Reactor Thermal-Hydraulics*, Vol 1, American Nuclear Society, 1983

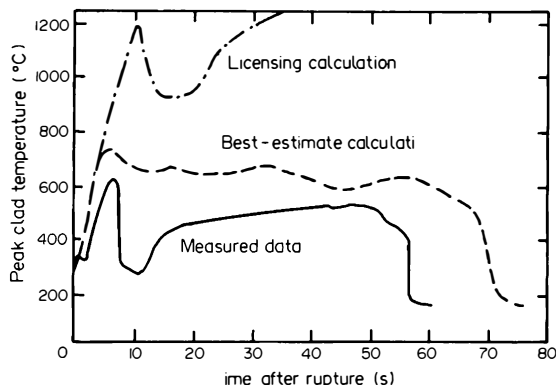


FIG. 15.7 Comparison of a LOCA experiment (L2-3) in LOFT and calculations with licensing and best-estimate (RELAP4/Mod 6) models. From M L Russel, Loss-of-Fluid Test Findings in Pressurized Water Reactor Core's Thermal-Hydraulic Behaviour, *Proc. on Nuclear Reactor Core's Thermal-Hydraulics*, Vol 1, American Nuclear Society, 1983

15.2 Fuel and Cladding

The fuel and the cladding are the first barriers against the release of radioactive fission products. The fuel performance directly affects the availability and the load factor of the plant. Fuel failure must therefore be avoided from the standpoint of both safety and economy. This requires an understanding of the basic phenomena and mechanisms for fuel behaviour under various operating conditions, which can only be acquired through experimental investigation and operating experience. Fuel irradiation testing under controlled circumstances and post-irradiation examination of the irradiated fuel is necessary. Such studies require a realistic reactor environment (Fig. 15.8), and radiation-protected remote manipulation of irradiated samples (Fig. 15.9).

Models for fuel performance are developed on the basis of experimental results and theoretical considerations. From the aspect of safety, the aim is to predict fuel behaviour in accident situations, i.e. during transient conditions. For this to be possible, fuel behaviour under steady state conditions must first be thoroughly understood. One of the primary tasks of fuel research is therefore to improve the understanding of fuel behaviour and failure mechanisms during normal operation. The computer code GAP-CON, mentioned in section 15.1.3, is an example of a mechanistic calculational model for steady state conditions.

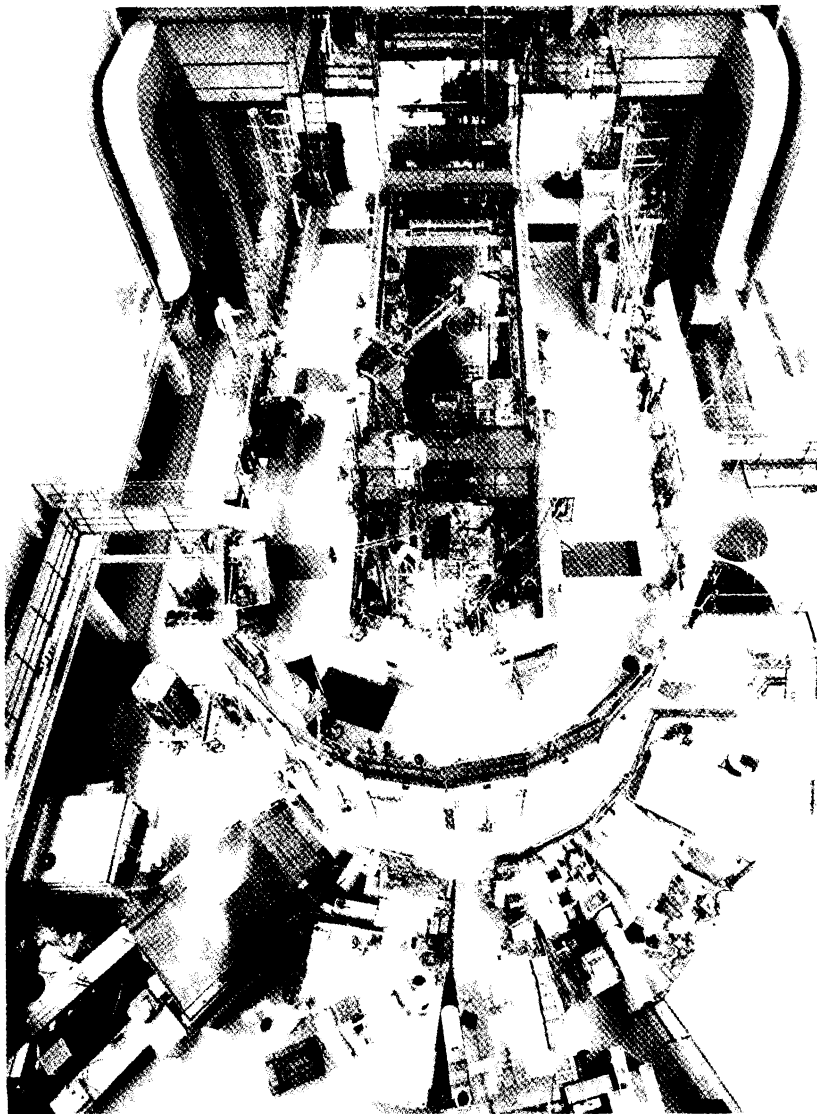


FIG. 15.8. View from above of the R2 materials testing reactor (50 MWth) in Studsvik. Fuel test samples can be inserted for irradiation in loops in the reactor core

15.2.1 Fuel densification

In the manufacture of fuel pellets, a slightly lower than the theoretically possible uranium dioxide density is desirable in order to leave enough room for the fission products formed during fuel irradiation. Hence, fresh fuel

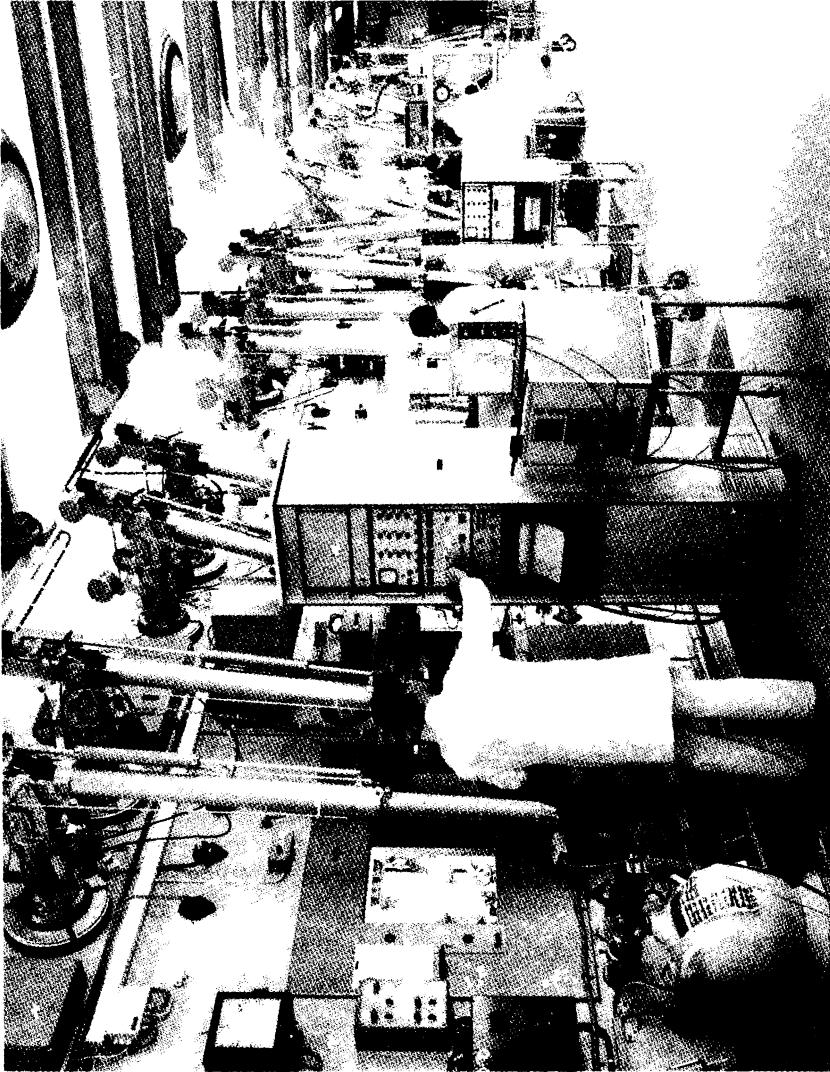


FIG. 15.9. Investigation of irradiated samples in Studsvik hot cells

incorporates small pores which are about a thousandth of a millimetre in diameter. In the early 1970s it was discovered in some U.S. reactors that the volume of the fuel decreased after a period of operation. Since such densification of the fuel could have a bearing on safety, a research programme was initiated to clarify the causes and mechanisms involved.

In a series of investigations at the Pacific Northwest Laboratories of the Battelle Memorial Institute, the effects of various parameters could be clarified (1517). The fuel densification was attributed to radiation-induced sintering, i.e. the dissolution of pores after a short period of burn-up. Once the mechanism had been established, fuel densification could be avoided by an appropriate sintering procedure during fabrication so that the desired pore distribution and grain size was obtained. By controlling the densification to counteract the simultaneous swelling due to fission gas release, an almost dimensionally stable fuel can be achieved during the early irradiation phase.

15.2.2 Pellet-clad interaction

The fuel material comes into full or partial contact with the cladding through thermal expansion, swelling, cracking and relocation. Since the fuel pellets expand more than the cladding, the cladding is subjected to severe stress, especially when the power is suddenly increased. Possible cracks may then extend and lead to clad failure (Fig. 15.10). This phenomenon, known as PCI (Pellet-Clad Interaction), has been extensively studied at Studsvik (1518).

A test procedure has been developed which involves base irradiation of fuel samples, and then, at a certain power level, subjecting the samples to a rapid linear power increase, a power ramp, in the R2 reactor. The systematic variation of burn-up, power level and ramp rate on well-characterized samples has made it possible to determine the influence of relevant parameters.

The significant mechanism is identified as stress corrosion in the reactive environment inside the cladding, created by certain volatile fission products, primarily iodine. A crack, initiated at a microscopic defect on the inside of the cladding, propagates until the stress in the remaining load-bearing part of the cladding exceeds the ultimate tensile strength, resulting in clad failure.

The risk of pellet-clad interaction has made it necessary to limit the rate of power change, which reduces the freedom in regulating the reactor power. Various remedies have been tried, such as introducing a zirconium liner on the inner surface of the cladding to reduce the tendency for stress corrosion, or coating the outside of the pellet with graphite to provide "lubrication" during contact with the cladding. Another method is to provide "rifles" on the inner surface of the cladding in order to control and limit the pellet-clad contact areas.

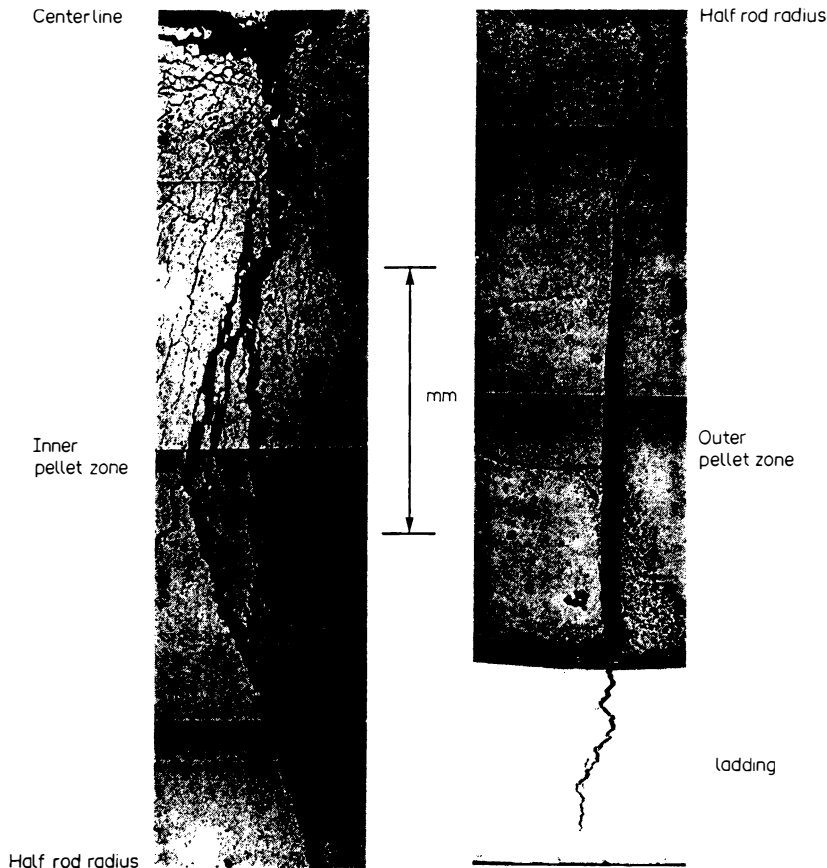


FIG. 15.10. Pellet-clad interaction. Cross-section of a fuel rod after ramp testing in the R2 reactor at Studsvik. A crack has appeared in the cladding opposite to a crack in the uranium pellet

15.2.3 Fission product release

Gaseous fission products collect in the microscopic pores of the uranium dioxide. The gas pressure causes the pores to grow and the pellet to swell. The swelling increases with temperature and burn-up. Fission gas release is relatively minor at temperatures below 1500°C. At higher temperatures, grain growth occurs, and the pore structure changes, so that fission gas is released. Release can also occur at lower temperatures if the pores become saturated with fission gas, as is the case at large burn-up, above about 20 MWd/kg U.

The released fission gas diffuses via grain boundaries and cracks to the gap between the pellet and the cladding. At high temperature and burn-up,

the fission gas pressure inside the cladding is high. Usually, noble gases such as krypton and xenon are major contributors. At high temperatures, volatile fission products, mainly iodine and cesium, add to the gas pressure. If the cladding is damaged, the inventory of gaseous fission products in the gap is released to the coolant.

Comprehensive research programmes have been carried out to determine the contribution of the gaseous fission products to the total gas pressure inside the cladding and to predict the quantity and composition of the fission products released from a damaged rod. The results show that the release can be approximately described by mechanistic models, although the understanding of the chemical form of the released fission products is still incomplete (1519).

15.2.4 Cladding properties

The identification of stress corrosion as a clad failure mechanism has led to intensive research for determining relevant failure criteria. It is not possible to specify simple criteria such as a critical stress or a critical strain. Several metallurgical, mechanical and chemical factors and the burn-up are important. Efforts have been directed into analysing the various stages of clad failure: crack initiation, crack growth and ultimate failure.

Crack growth normally occurs through the mechanical-chemical breakdown of the oxide layer on the inner cladding surface in the presence of iodine. The growth rate depends on the stress at the tip of the crack and a number of other parameters. It has been found that in unirradiated Zircaloy, some plastic deformation is necessary for stress corrosion to occur. Since the yield strength must therefore be exceeded, it would be expected that irradiated material would require higher stress for crack propagation. However, studies have shown that irradiated Zircaloy is susceptible to stress corrosion cracking far below the yield strength limit (1520). This may be interpreted as a considerably higher crack growth rate in irradiated than in unirradiated material.

15.3 Materials and Mechanics

The integrity of the reactor pressure vessel and primary system envelope is fundamental to reactor safety. A large pressure vessel rupture would have catastrophic consequences. The probability of pressure vessel failure must be so low that a rupture can be considered incredible. This is achieved by the application of well-proven design standards with large safety margins, by the selection of the best material possible and by the detailed specification and control of the manufacturing process. The requirements also apply to any connecting pipes and systems which are pressurized from the reactor,

although reactor safety systems are designed to cope with a maximum pipe break without significant offsite consequences.

Considerable research has been devoted to finding suitable materials and determining their properties, to establishing criteria and estimating probabilities for failure as well as to designing suitable test methods. Research in this area is carried out in the HSST (Heavy Section Steel Technology) programme of the USNRC, which has been in progress since the early 1970s. Important materials research is also being carried out in West Germany, Japan and Sweden.

15.3.1 Material properties

Steel can be given a high strength with suitable alloy materials. For pressure vessel steel, a high fracture toughness is desirable. This is achieved by eliminating any impurities and alloy elements. A fair compromise between the requirements for high fracture toughness and high yield strength is attained in the *low-alloy steels* used as reactor pressure vessel material. These steels contain small amounts of manganese and nickel (see Table 3.6).

The properties of pressure vessel steels have been determined for the base material as well as for welds and heat affected zones (1521). Certain changes can be expected during the operating lifetime of the pressure vessel due to neutron irradiation and ageing. The changes are manifested as an increase of the yield strength and the transition temperature from the ductile to brittle state. Test methods have been developed to follow the changes in material properties with time.

An example of a Swedish research contribution in this field is the measurement of the dynamic fracture toughness at operating temperature (Fig. 15.11). The result shows that the fracture toughness above the transition temperature varies with temperature and strain rate, i.e. the rate of the load change which the pressure vessel may be subjected to during reactor transients.

15.3.2 Fracture mechanics

Fracture mechanics deals with the relationship between material properties, stress state and crack occurrence. The condition for brittle fracture can be expressed by a *critical crack size* for rapid, unstable crack growth. In the elastic range, the critical crack size can be calculated using linear elastic fracture mechanics (3.5.2). In the ductile area, a substantial plastic deformation in front of the crack is required for crack growth to continue. The linear theory does not apply in this case, and elastic-plastic fracture mechanics must be used.

The theory of linear and non-linear fracture mechanics has largely been

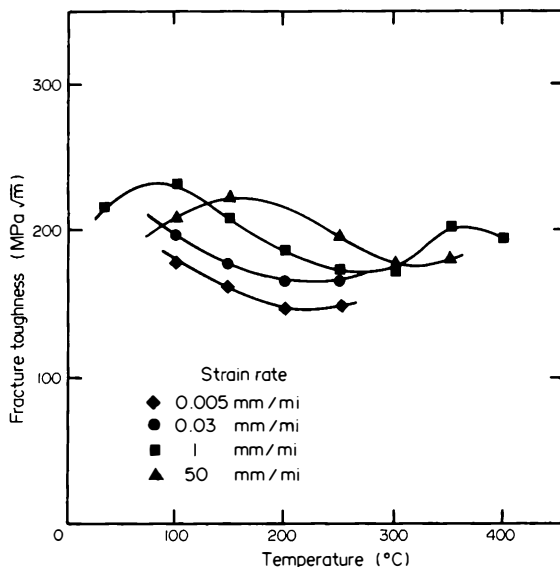


FIG. 15.11. Dynamic fracture toughness of pressure vessel steel A533B versus temperature for various strain rates. From B Östensson, R Westin, *The Fracture Toughness of A533B Pressure Vessel Steel at Low Strain Rate*, Studsvik Report S-573, 1977

confirmed by experiment. Extensive experiments have been carried out in the HSST programme, including hydrostatic testing of model vessels to failure. Theory and experiment show that failure cannot occur at the stress and strain levels to which a real reactor pressure vessel is subjected as long as it remains in the ductile region (1522).

Nevertheless, one can never be absolutely sure that an unfavourable combination of material properties, state of stress, and crack size will not occur, since these factors are stochastic in nature. The failure probability of reactor vessels has been estimated using assumed probability distributions for the parameters concerned. Extremely low values are obtained even with pessimistic assumptions (1523). This confirms the qualitative conclusion that the reactor pressure vessel is a very safe component.

Probabilistic fracture mechanics has also been used to estimate the failure probability of pipes. The results indicate that the fracture probability is very low for the pipes and loads occurring in a reactor (1524). The estimated leak probability is much larger, which confirms the conclusion of the deterministic analysis on "leak before break". These results have led to a relaxation of design criteria for the reactor primary system piping in the USA and West Germany. The LOCA criteria are not affected, however.

15.3.3 Test methods

Even if unstable crack growth cannot occur in the reactor vessel at operating temperature, a situation where the temperature falls below the transition temperature while the vessel is subjected to stress cannot be ruled out. It must therefore be assured that no cracks larger than the critical size are present. This is achieved by careful manufacture, testing and inspection prior to start-up as well as regular in-service inspections. The quality control is carried out by non-destructive test methods, particularly using ultrasound.

Ultrasonic testing is based on the fact that high frequency sound waves propagate as a beam in homogeneous material but are reflected by any discontinuities in the material. Cracks and other defects can be located by recording the reflected beam energy. The resolution is of the same order of magnitude as the wave length. For example, the wave length in steel is 2.7 mm for ultrasound with a frequency of 2.25 MHz. However, there are several theoretical and practical problems which limit the use of the conventional technique.

In an international research programme, called PISC (Plate Inspection Steering Committee), samples with hidden defects were independently investigated by various groups. It was found that 25 mm cracks could only be detected with a 50% probability as opposed to the expected 95% using methods prescribed in the U.S. ASME Boiler and Pressure Vessel Code Section XI (1525). In general, accumulations of smaller defects could not be detected. Alternative methods using focused sound beams or double probes led to considerably better results.

15.4 Corrosion and Water Chemistry

Reactor structural materials are exposed to various kinds of corrosion. A distinction is made between general corrosion and localized corrosion. General corrosion is a uniform attack of the entire metal surface. The resistance to corrosion in the reactor environment is based on the spontaneous formation of a thin protective layer on the surface of the material. General corrosion is very moderate, a few hundred millimetres per year in carbon steel and low-alloy steel and even less in stainless steel. Whilst this amount is of no consequence to the strength of the material, the corrosion products which are formed and released into the coolant can affect reactor operation and maintenance.

If the protective oxide layer is damaged, either mechanically or chemically, localized attack can result by the initiation and extension of a crack due to the mechanical stress at the tip of the crack, which is called stress corrosion (cf 3.5.3). The crack growth rate is affected by the varying loads to which the component may be exposed during reactor start-up, shutdown

and transients. This is known as corrosion fatigue. Localized attack is more serious than general corrosion since the attack extends inwards instead of sideways.

15.4.1 Corrosion fatigue in pressure vessel steel

Pressure vessel steel does not normally come into contact with the coolant, since it is protected by a stainless steel liner on the inside of the vessel. If the liner is penetrated, the vessel may be exposed to corrosion fatigue if there are defects in the material. Growth occurs slowly in subcritical cracks. Limit values for the growth rate have been established in the U.S. pressure vessel code, ASME XI.

In order to improve the experimental information on corrosion fatigue in pressure vessel steel, the USNRC and EPRI (Electric Power Research Institute) launched an international research project in 1977. Identical samples were analysed at several laboratories. An example of the results is shown in Fig. 15.12.

It can be seen that results vary greatly. To a certain extent, this can be explained by the fact that the crack growth rate depends upon the oxygen

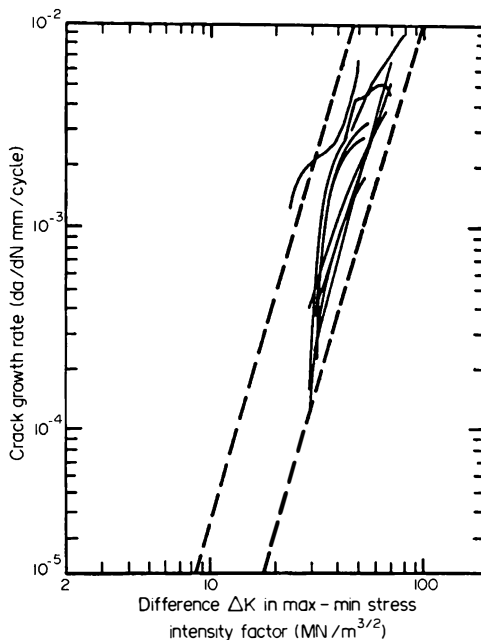


FIG. 15.12. Measured growth rate during corrosion fatigue of pressure vessel steel A533B in reactor water. The dashed lines indicate the crack growth rate limits as specified in ASME XI for air (lower line) and "reactor water". From K Gott, B Östensson, *Corrosion Fatigue of Pressure Vessel Steel A533B*, Studsvik Report EI-80/2, 1980

content in the reactor water, and that this and other conditions were different in the cases investigated.

15.4.2 Stress corrosion in stainless steel

Austenitic stainless steel, which is used in the main and auxiliary coolant systems, is susceptible to stress corrosion under certain circumstances. Stress corrosion cracking is a generic problem for boiling water reactors (see 14.1.1). The mechanism of intergranular stress corrosion cracking (IGSCC) has been clarified through systematic research, mainly in the USA. It has been found that IGSCC requires the interaction of three factors: the weakening of grain boundaries in the material through sensitization, the mechanical stress exceeding the yield strength, and the presence of oxygen in the coolant. In order to counteract IGSCC, it is sufficient to eliminate one of these factors.

In Sweden tests have been made on the injection of hydrogen into the feedwater for reducing the oxygen content in the coolant (1526). In 1979 and 1981, short-term tests were conducted in Oskarshamn II which demonstrated that it was possible to obtain such a low oxygen content that IGSCC was not expected to occur. In 1983 and 1984 further experiments were carried out in Ringhals 1 and Forsmark 1 where sensitized samples were subjected to stress in a real reactor environment. The experiments showed that a considerable oxygen reduction could be obtained with a moderate hydrogen dosage, thus preventing IGSCC without any unfavourable side effects. It was also found that small concentrations of impurities in the coolant have a greater effect on the risk for stress corrosion than previously believed.

15.4.3 Water chemistry

Pressurized water reactors are susceptible to corrosion in both the primary and the secondary system. The corrosion is directly connected to water quality. The primary coolant contains boric acid for reactivity control (cf 5.4.1). In order to minimize general corrosion, the coolant is treated with an alkalizing agent, such as ammonia or lithium hydroxide. By adjusting the dosage of the alkalizing agent to the boric acid concentration so that a suitable pH value is maintained, the general corrosion level can be reduced and the solubility of the corrosion products in the coolant minimized.

Oxygen formation through *radiolysis*, i.e. the decomposition of water due to radiation, is lower in pressurized water reactors than in boiling water reactors. Hydrogen is added to the coolant to further reduce oxygen formation. Although the basic radiation chemistry water is rather well known, the understanding of the conditions during reactor operation is still incomplete, especially for boiling water reactors.

The corrosion of steam generator tubes is one of the most important causes of forced outages in pressurized water reactors (see 14.1.2). There are several mechanisms at work which have called for changes in the chemical treatment of the feedwater. The most important parameters to be kept under control are the pH, the cation conductivity, and the chloride content. However, it has so far been difficult to correlate the observed corrosion to the water chemistry.

15.4.4 Decontamination

With the dissolution of corrosion products in coolant and the subsequent redeposition on other surfaces, radioactive material is transported from the core to other parts of the primary system. All surfaces in contact with the coolant become radioactive, making servicing and maintenance difficult. One way of reducing potential radiation doses is to remove the radioactive deposits. This is known as decontamination (cf 6.5.4). Decontamination is especially important in large operations, such as PWR steam generator repair, replacement of BWR high energy piping, and reactor decommissioning.

Although the radioactive deposits mainly consist of iron, nickel and chromium, the radiation level is dominated by the isotopes of cobalt, Co-58 and Co-60. The oxide layer can be removed by using concentrated inorganic or organic acids sometimes preceded by an oxidation step with concentrated alkaline potassium permanganate. These “hard methods” are mainly intended for the decontamination of components which are removed from the reactor or for the decommissioning of the entire reactor.

Large research efforts have led to the development of “soft methods” which use certain diluted solutions of reducing and complexing agents (1527). One of the advantages of these methods is that they are not corrosive. They can therefore be used for periodic decontamination, e.g. prior to scheduled outages for service and maintenance.

15.5 Instrumentation and Control

Reactor performance is continually monitored. The information from sensors and detectors is processed to provide input signals for the automatic protection and control systems. Operating data are displayed in the control room and provide the basis for operator action. Control and monitoring systems must be designed to optimize the operator's possibilities to follow the reactor processes and carry out the required action. Research in this field has to a large extent concentrated on the man-machine interface in the design of the control room and on various forms of operator support.

15.5.1 Control room design

Traditionally, data are displayed in the control room on analog instruments and in the form of alarm signals. The wealth of information makes necessary a careful selection of data to be presented. The ergonomic layout and location of controls and displays is of great importance. New process computers have been installed in the Swedish reactor units for computer-based display to supplement the conventional data presentation via instruments.

Traditional control rooms are designed for normal reactor operation and design basis accident conditions (see 7.3). The operator plays an important role during normal start-up, shutdown and power changes. In abnormal events which require prompt response, the necessary action is initiated automatically, and human intervention is only required if the automatic systems fail. For example, in Swedish reactors no manual action is required within 30 minutes after the initiation of a design basis accident.

Since TMI-2, attention has turned towards the management of accidents beyond the design bases. Requirements are being established on how plant data should be monitored and displayed also for severe accident conditions. Although present-day control rooms largely meet these requirements, certain improvements and modifications may be necessary. They could involve the selective grouping of process information for diagnosing the state of the plant before, during and after the accident, and the identification of critical safety functions for mitigative action (1528).

The working conditions and the behaviour of the control room crew during complex sequences have been studied in the Swedish nuclear power plants (1529). The studies confirm that the control rooms function well. Some modifications have been implemented, mainly for maintaining and improving the operator's feel for and understanding of the reactor processes as the control operations are increasingly automated and computerized. Research has also provided a basis for improving the training of control room personnel.

15.5.2 Operator support

Normal operator action, such as during start-up and shutdown, is based on well-practised procedures. There are special instructions for action in abnormal situations. Experience from TMI-2 indicates that the usual operating rules are inadequate in situations which deviate from the design bases. Emergency Operation Procedures (EOP) have therefore been established to supplement the traditional operating rules. The focus of the Emergency Operation Procedures is to ensure that critical safety functions are fulfilled and mitigative action adopted in response to symptoms of abnormal conditions.

One of the lessons learnt from TMI-2 was that the operators possessed inadequate knowledge of plant conditions during the accident. It was therefore suggested to provide the control rooms with a Safety Panel Display System (SPDS) showing a selection of safety-related parameters. The display should be symptom-oriented instead of event-based and provide an overview of the state of the critical safety functions (1530).

Another kind of computer-based operator support has been developed in West Germany and the USA, namely *on-line disturbance analysis* (1531). This means that in addition to indicating safety-related critical parameters, the computer tries to diagnose the event immediately and propose mitigative action. The diagnosis is performed by comparison of the real event sequence with a series of pre-calculated sequences stored in the memory of the computer. The computer then displays information on the probable cause of the disturbance, the operational consequence if the disturbance remains, and proposals for corrective action.

Although computers are not yet used for the direct control of safety-related processes in light water reactors, a development in this direction is to be expected. It is therefore important to study the reliability and quality assurance issues associated with computer-controlled safety systems. These issues particularly relate to the specification, design, verification and documentation of the computer software.

15.5.3 Accident instrumentation

Safe reactor operation requires comprehensive instrumentation to actuate the reactor protection system if necessary. In order to follow the progression of an accident, information is required on the status of individual safety systems and on whether or not a safety function has been carried out. The corresponding instrumentation is usually adapted to design basis accident conditions. Experience from TMI-2 indicated several deficiencies in the traditional instrumentation, e.g. that the measuring range was too limited or that the instrument failed.

Requirements on extending the range and improving the reliability as well as on the ability of the instruments to withstand more severe operating conditions have therefore been established (1532). This made it necessary to review and upgrade the existing instrumentation. New instruments have been developed, e.g. for in-vessel liquid-level detection. In some cases, it has been difficult to satisfy the requirements for instruments to withstand accident conditions. The entire measuring chain must be tested to prove that it can withstand the severe environment which may arise in the reactor containment during an accident. Because of the potentially severe conditions, electrical equipment is placed outside the containment as far as possible.

15.6 Reliability and Uncertainties

The Reactor Safety Study was a breakthrough in the application of reliability analysis to reactor safety. The basic event tree–fault tree methodology has been further developed and, in combination with an extended data base, is found to be a useful tool for the quantification of nuclear power plant safety and risk. Development continues in order to improve the treatment of dependent failure and human reliability as well as of uncertainty and incompleteness.

15.6.1 *Methods development*

The reliability analysis of nuclear power plants is a complex process involving several steps:

- identification of initiators and sequences which can result in severe core damage;
- modelling of systems and components including dependences and operator action;
- determination of failure probabilities for base events, including human error;
- estimation of core damage frequencies, including uncertainty analysis.

Several methods have been developed to identify event sequences and to construct system models (1533). The traditional event tree–fault tree methodology which was introduced in the Reactor Safety Study is still dominant. The borderline between event trees and fault trees varies from study to study. There is a tendency to use small event trees and large fault trees, as the capacity of the computer codes for fault tree analysis increases. Computer-based methods have also been developed for the construction of fault trees (1534).

The development of data bases for fault tree quantification includes data collection and analysis of base events, selection of suitable reliability models, and documentation. A centralized bank of failure data from nuclear power plants has existed for several years in Sweden. A handbook of reliability data for components in Swedish boiling water reactors has been published (1535).

Special computer codes have been developed for quantitative fault-tree analysis. One kind of code is used for the calculation of minimal cut sets for a given fault tree. A problem associated with such codes is that large fault trees require large storage capacity and long search time due to the large number of cut sets. Various methods of reducing the computer time, such as eliminating cut sets with low probabilities, have therefore been developed.

A comparison of methods and data for reliability analysis was carried out

in a joint Nordic project (1536). Studies of the reliability of a typical PWR safety injection system and of the modelling and quantification of a BWR loss of feedwater transient were performed independently at four Nordic research institutes. The first study showed the sensitivity of the results to the choice of baseline data. The second study demonstrated the significance of different methods of system and component modelling.

15.6.2 *Dependent failures*

Dependent failures or *common cause failures* (CCF) tend to increase the frequency of multiple, simultaneous failures. The common cause may be an external event, a manufacturing defect or a manoeuvring error. Propagating failures are a type of CCF when a component failure causes a change of the conditions and environment which results in further component failures.

A combination of several methods is usually used in the analysis of dependent failures. First, the dependences must be identified, which may be done by examining the fault trees, visiting the plant, interviewing operating and maintenance personnel, etc. The fault trees are then modified and new failure probabilities estimated for the components concerned, using some parametric model. The beta-factor model is an example of such a model (see 10.2.5). This model has been extended for application to systems with high levels of redundancy (1537). Another category of methods uses special computer codes to search for dependences between minimal cut sets in the fault trees (1538).

The lack of data for validation of the parametric models is an essential weakness in the analysis of dependent failures. To a certain extent this can be compensated for by means of sensitivity analysis, in which the model parameters are varied or alternative models are used. Sometimes the elimination of the dependence by physical segregation or diversification is justified. Intensive efforts are being devoted to improving the classification, modelling and data bases for dependent failures.

15.6.3 *Human reliability*

A quantitative analysis of human error in connection with reactor safety was first attempted in the Reactor Safety Study. The effects of erroneous action during testing and maintenance and of deviations from standard procedures during normal operation and abnormal events were studied. Fault trees were constructed in the same way as for component and system analysis. This method, known as *mechanistic* human reliability analysis, has been further developed and described in a handbook (1539). A general problem with this method is the difficulty of quantifying the failure probabilities.

Mechanistic models for human action are best suited to the analysis of routine procedures. Action in unexpected situations is more difficult to

represent. Human error differs from equipment failure in that it can be corrected, given enough time, through the feedback of information and knowledge-based behaviour. Attempts to model knowledge-based behaviour have been made (1540). The models indicate a very complex interaction of factors which are impossible to quantify at present.

Simplified *dynamic models* have been developed which can be used to quantify knowledge-based behaviour in accident situations (1541). These models are based on the fact that the nature of the event must be determined before the appropriate corrective action can be selected and implemented. In order to facilitate the analysis, operator action fault trees are constructed (see 10.2.6). The trees are quantified using reliability–time curves (Fig. 10.13) which express the probability of human error as a function of the available time. The determination of failure probabilities in both the mechanistic and dynamic models suffers from a lack of statistical data.

The aim of the dynamic models is to simulate the way in which humans react in abnormal situations. An important cause of operator error is the wrong diagnosis of an abnormal event, which can result in omitted or erroneous action. Estimates of human error probabilities are often based on expert opinion. Various methods of structuring expert opinion have been developed (1542). The results will depend on the level of knowledge among the experts. Experience shows that experts often tend to underestimate the failure probabilities of knowledge-based behaviour.

15.6.4 Uncertainties

Plant safety analysis usually provides point estimates of core damage frequencies for various event sequences. The frequencies of the individual sequences are summed to obtain the total core damage frequency. Uncertainty arises partly from the stochastic variation of base data, and partly from shortcomings of data and models. The latter contribution to the uncertainty can be reduced by expanding the data bases and improving the models.

Uncertainties in the base data are propagated through the fault trees and event trees, to a resulting uncertainty in the core damage frequency for an event sequence. More uncertainty is added when the frequencies are summed to obtain the total core damage frequency. There is as yet no generally accepted method for propagating and combining the uncertainties in probabilistic safety analysis. This is partly due to the fact that the probabilities for the base events are a mixture of objectively verifiable and subjectively estimated data.

A qualitative assessment can be made by estimating the upper and lower bounds of the most important contributors to data and model uncertainty. The effect on the result is then determined by sensitivity analysis. Several methods have been used for quantitative analysis. In the Zion Probability

Safety Study (1543), base data are characterized with statistical distribution functions and the error propagation is studied analytically or numerically using special computer codes.

Another kind of uncertainty arises from the impossibility of guaranteeing the completeness of the analysis. The questions to be asked are: Have all important sequences been considered and all important physical processes been modelled? Have all dependences and possibilities of human error been identified? The quantification of these uncertainties is impossible in principle. The uncertainty can only be reduced by further analysis. Through the systematic way in which current analyses are performed, it is improbable that significant sequences and failure sources remain hidden.

15.7 Core Melting and Containment Behaviour

The Reactor Safety Study concluded that accidents involving severe core damage were major contributors to the environmental risk. After TMI-2, considerable research efforts were directed to improving the understanding of core meltdown processes and containment behaviour for accidents with insufficient core cooling. In this section, model development and experimental verification are briefly described, and the uncertainties assessed.

15.7.1 Modelling

During an accident with insufficient core cooling, the core overheats and melts. Molten core material collects at the bottom of the reactor vessel, which is soon penetrated. Depending on the particular accident sequence, the melt then either falls by gravity (low pressure case) or is ejected at high pressure into the reactor containment where it is eventually cooled. Steam and gases are generated during the melting process and in the interaction between the molten corium, water and concrete. This increases the containment pressure and temperature, and can result in containment failure.

Physical models have been developed which describe the thermohydraulic processes in the primary system and containment. The models form part of computer codes for calculating the pressure, temperature, hydrogen formation, concrete attack, etc., as a function of time after the initiating event. The accident progression is largely determined by the initiating event, the design and performance of the reactor coolant system and containment, and by any operator action undertaken. The codes must be adapted to the specific plant under study and must be able to describe the effects of human intervention.

The first computer codes for the thermohydraulic analysis of severe accidents were MARCH, developed by Battelle Columbus Laboratories on behalf of the USNRC, and MAAP, produced within the IDCOR programme (Industry Degraded Core Rulemaking Programme) set up by the

U.S. nuclear industry. MARCH and MAAP, which have been issued in successively improved versions, are based on simplistic models to provide fast-running codes for survey calculations. Detailed models for separate effects in the accident progression are also developed. The models are validated, i.e. their accuracy is tested, by comparison with experimental data.

When fuel melts, volatile fission products and other substances are released. The vaporized materials may condense on surfaces in the reactor coolant system or in the gaseous phase, forming aerosols. The laws governing melt release and aerosol formation are not yet completely understood, nor are the chemical forms in which the various substances may exist.

Release rates for fission products from overheated fuel are primarily determined by diffusion phenomena in the fuel. For the main fuel components, uranium and zirconium, and other structural materials in the core, direct vaporization determines the release rate. Eutectics may form which melt at a lower temperature than the UO_2 itself.

Diffusion and vaporization models are included in the CORSOR and FPRAT computer codes which calculate the release rate of fission products from fuel. The vaporization of other substances in the core is also calculated. CORSOR and FPRAT are used in combination with MARCH and MAAP, which determine the temperature history of the core. Special codes have been developed for calculating the release of fission products and other substances during melt-concrete interaction.

Detailed mechanistic codes for predicting the core condition, fission product release, etc., are being developed in the USNRC's research programme (1544). SCDAP (Severe Core Damage Analysis Package) models core melting and fission product release, while TRAP-MELT describes the transport of the released substances in the reactor coolant system. The second generation MELPROG code integrates the description of in-vessel processes and the release to the containment at vessel breach. Core-concrete interaction is modelled by CORCON and release from the core debris by VANESA. Aerosol transport and retention in the containment is described by MEAROS and containment loads from hydrogen burn by HECTR. The ex-vessel models are integrated in the second generation CONTAIN code.

The mechanistic codes in the USNRC development program are summarized in Fig. 15.13, which also shows the corresponding codes developed by U.S. nuclear industry, and in the German PNS (Projekt Nukleare Sicherheit) project (1545).

In the aerosol codes, the reactor plant is divided into a number of compartments in which the gases and gas-borne particles are assumed to be well mixed. The concentrations change by transport to other compartments and by the effects of various removal mechanisms. Along with the natural mechanisms indicated in Fig. 15.14, special engineered systems, such as filters, the containment spray system and the condensation pool (boiling water reactors), are effective in reducing the aerosol concentration. The computer

1. Detailed Mechanistic Codes

Sponsor	In-Vessel Processes			Ex-Vessel Processes			
	Thermal Hydraulics	Core Melting	Release from Fuel	Transport in RCS	Vessel Failure	Concrete Interaction	Release from Debris
NRC ^a	RELAP-5 TRAC PSAAC	SCDAP CORMLT	MELPROG	TRAPMELT RAFT IMPAIR		CORCON	VANESA CONTAIN
EPRI ^b PNS ^c						WECHSL	MAEROS SUPRA NAUA
							HECTR COCMEL

2. Integrated Codes

NRC	1st generation	MARCH1-3, STCP
IDCOR ^d		MAAP1-3
NRC	2nd generation	MELCOR

^a Nuclear Regulatory Commission^c Electric Power Research Institute^b Projekt Nukleare Sicherheit^d Industry Degraded Core Programme

FIG. 15.13. Survey of U.S. and German mechanistic codes for severe accident phenomena

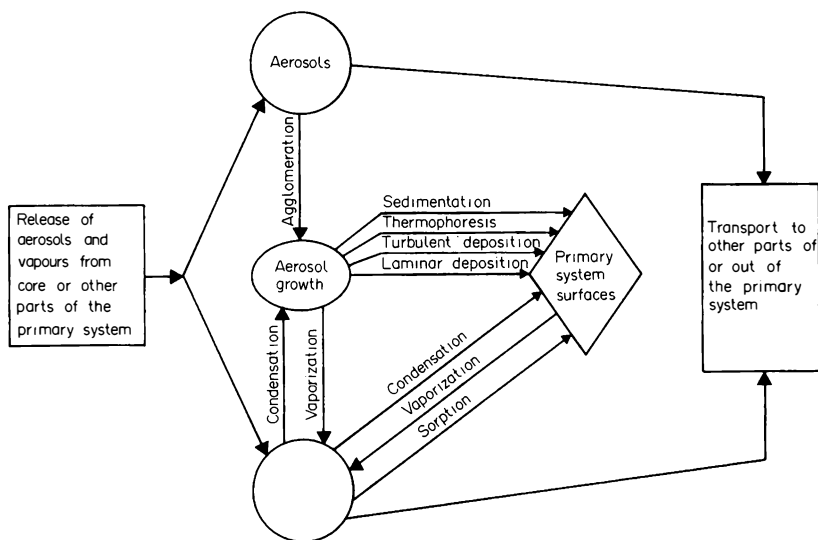


FIG. 15.14. Mechanisms for aerosols and vaporized material in the reactor coolant system

codes calculate the concentration in various compartments as a function of time. Releases to the environment can also be determined, if and when the containment is penetrated.

The aerosol code CORRAL was used in the Reactor Safety Study in a four-compartment version for pressurized water reactors and a six-compartment version for boiling water reactors. More detailed models have since been developed in West Germany and the USA. The German code NAUA calculates aerosol behaviour in a closed volume with an atmosphere, containing steam which may condense on the aerosol particles. The USNRC codes SPARC and ICEDF calculate the effectiveness of suppression pools and ice condensers in retaining or releasing fission products from the containment structures.

In the Reactor Safety Study it was conservatively assumed that essentially all iodine was released and transported in gaseous form as elemental iodine. There is strong evidence that the major part of the iodine combines with cesium to cesium iodide which is less volatile and dissolves in water or forms aerosol which is deposited in the reactor containment. The most important chemical reactions and their effects are modelled in the IMPAIR code (Iodine Matter, Partition and Iodine Release) developed at the Karlsruhe Nuclear Research Centre on the basis of extensive experimental research.

A characteristic of the early severe accident codes was that heat transport

and fission product transport were calculated separately. The coupling due to the fission product decay heat, which can result in revaporization and relocation of fission products was not represented. The revaporization of condensed substances in the reactor vessel can be decisive for the magnitude of the release from the containment. *Integrated codes*, incorporating revaporization and covering the accident sequence from beginning to end, have therefore been developed. Examples of integrated codes are the Source Term Code Package (STCP), developed by Battelle Columbus Laboratories (1546), and the extended versions of MAAP. The USNRC is sponsoring the development of a second generation integrated code, called MELCOR.

For estimating the offsite consequences, the Reactor Safety Study used the CRAC (Calculation of Reactor Accident Consequences) code. Its successor, CRAC 2, has been used in many subsequent risk studies. A second generation offsite consequence code, called MACCS (MELPROG Accident Consequence Code System) was used in the Reactor Risk Reference Study. MACCS (1547) represents a major development from the CRAC series of codes, including the use of a multiplume atmospheric dispersion model that can represent time-varying release paths, improved deposition models and health effects data (1548).

15.7.2 Experiments

Since TMI-2, many experiments have been carried out to simulate severe accidents as a basis for validation of the calculational models. The experiments are often carried out in international co-operation. Some large experimental programmes are listed in Table 15.1.

The SFD (Severe Fuel Damage) programme forms an important part of the NRC severe accident research programme. A series of integral fuel bundle tests are carried out in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory, in the NRU reactor at Chalk River, Canada, and in the Annular Core Research Reactor (ACRR) at the Sandia National Laboratories. The fuel rods undergo nuclear heat-up to 2200°C. Fission product release, clad oxidation, hydrogen formation, aerosol production, etc, are measured in the experiments. The SFD programme has been in progress since 1982.

The aim of the West German research project BETA (BETonAnlage) is to provide a basis for the calculation of melt-concrete interaction. The research facility consists of a concrete crucible which holds the inductively heated simulated molten core. It is possible to work with melt quantities up to 600 kg. Two kinds of experiments are carried out: one in which the melt is kept at high temperature, about 2300°C, corresponding to the start of concrete attack, and another in which the melt is nearly at the point of

TABLE 15.1 *Experiments for the simulation of severe accident phenomena*

1. THERMOHYDRAULICS	Performed at
Core meltdown SFD	INEL ^a , SNL ^b
Concrete melt interaction BETA	KfK ^c
System behaviour LOFT	INEL
Hydrogen combustion	EPRI ^d
2. FISSION PRODUCT RELEASE	
SASCHA	KfK
CORE MELT	ORNL
3. AEROSOL BEHAVIOUR	
MX-V	Studsvik
DEMONA	KfK
LACE	EPRI
4. HIGH PRESSURE MELT EJECTION	
HIPS	SNL

^aIdaho National Engineering Laboratory, USA.^bSandia National Laboratories, USA.^cKernforschungszentrum, Karlsruhe, West Germany.^dElectric Power Research Institute, USA.^eOak Ridge National Laboratory, USA.

solidification, about 1500°C. The penetration rate in the radial and axial directions is measured as well as the release and composition of gaseous substances.

The BETA experiments were used to validate the WECHSL code. Figure 15.15 illustrates the concrete erosion in one of the experiments, as compared

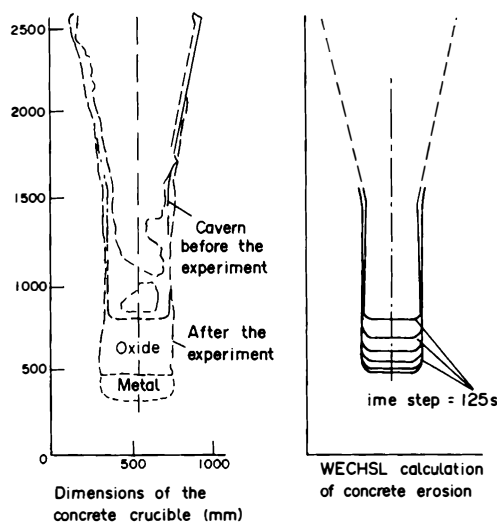


FIG. 15.15. Concrete erosion in a BETA experiment as compared to WECHSL calculations. From H-H Hennies et al, Forschungsergebnisse zum Kernschmelzunfall in einem modernen 1300-MWe-DWR, *Atomwirtschaft*, November 1986

to WECHSL results. The experiments showed that the initial erosion rate was higher than pre-calculated, but that the totally eroded volume was about the same as predicted. This means that the initial rate of hydrogen generation from core-concrete interaction is higher than previously assumed, which would lead to an unfavourable situation in the reactor containment.

The original aim of the LOFT project was to study LOCA sequences in a pressurized water reactor and to verify that the safety requirements are fulfilled in the design basis events (see 15.1.2). During most of the experiments the core remained intact as expected. In the last two experiments, however, the temperature of the core was intentionally so high that the central part of the core was damaged and fission products were released.

In the first experiment, although the maximum clad temperature was limited to 1100°C, it was enough to cause clad failure. The amount of gaseous and volatile fission products released, transported and removed in the primary system was studied. The second experiment was designed to provide clad temperatures in excess of 1800°C. An interfacing LOCA ("V-LOCA") was simulated, involving a direct release path from the reactor coolant system to the auxiliary building. To accomplish this, a special central fuel assembly was built. The released fission products and aerosols passed through a pipeline, simulating the low pressure injection system, to a suppression tank.

The experiment was successfully run in July 1985. The heat-up phase was close to expectations as the assembly was uncovered. The temperature rose rapidly as the zirconium-steam reaction began to dominate the heat release at a clad temperature of about 1500°C. Fuel temperatures were maintained above 1800°C for 4½ minutes. The experiment was terminated by the injection of emergency coolant.

Hydrogen is produced in severe core damage sequences, due to zirconium-steam reaction and core-concrete interaction. The hydrogen contributes to containment pressure build-up and can ignite and burn in the presence of air and steam. Detonation can occur at certain mixture ratios (see 11.1.4). In order to study hydrogen burn, an international research project co-ordinated by the Electric Power Research Institute (EPRI), was carried out during 1981-4 in the USA.

The project comprised several series of large-scale experiments on the ignition of hydrogen, steam and air mixtures. The experiments showed that the pressure and temperature rise due to hydrogen combustion was moderate in conditions corresponding to those existing in PWR dry containments during severe accidents. Hydrogen detonation occurred only at high concentrations in particular geometries.

Fission product release from simulated molten corium was studied in the SASCHA facility in Karlsruhe during 1974 to 1984 (1549). SASCHA mainly consisted of a high frequency furnace, a crucible containing molten corium,

and equipment for aerosol collection and analysis (Fig. 15.16). The corium consisted of UO_2 , Zircaloy and simulated fission products. Small amounts of stainless steel and control rod material were added so that the composition was representative of that of a molten core. The mass of the simulated corium melt was 200–250 grammes. The time and temperature dependence of the released substances was measured by collecting aerosol particles on filters and subsequent radiochemical analysis. The results were used to determine functions which describe the time-dependent release of fission products from the fuel during core melting.

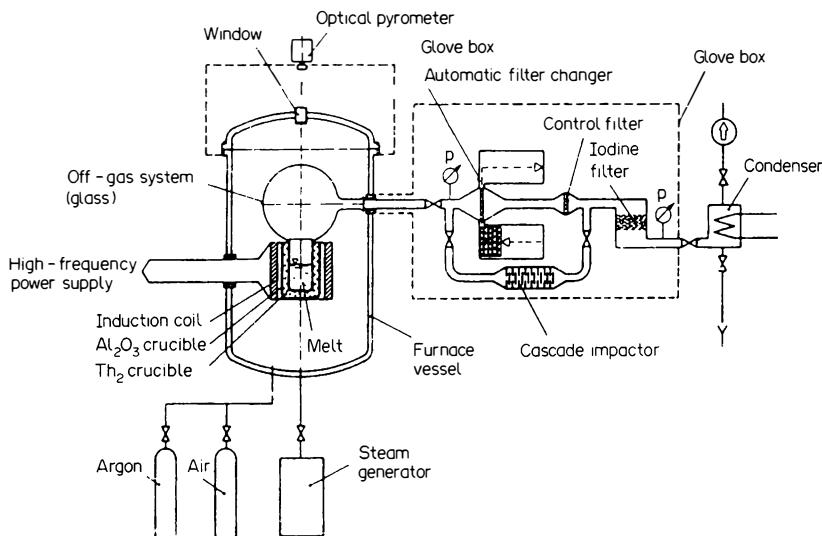


FIG. 15.16. The SASCHA experiment on fission product release from molten core material

From 1982 to 1985 an international project involving aerosol transport in a large-scale model of the reactor coolant system was carried at the Marviken facility in Sweden. Two kinds of substances were studied: simulated fission products ("fissium") of iodine, cesium and tellurium, and simulated corium, corresponding to the core material of a pressurized water reactor. The fissium and corium were vaporized in a special aerosol generator and their transport and removal in a model of a reactor vessel, pressurizer and pipes were studied.

Some conclusions and observations from the Marviken experiments are (1550):

—The dominant deposition mechanisms are gravitational settling and inertial impaction.

- The mass median diameter of aerosol particles leaving the reactor vessel was about 12 microns.
- In tests with only fission, the CsOH, CsJ and Te species were transported together. In tests with both corium and fission, there was evidence of differing transport behaviour.
- Most of the aerosol collected in the water-filled relief tank.

The aim of the international DEMONA (DEMONstration of NAture) project was to demonstrate in large scale the natural removal mechanisms for aerosols under simulated severe accident conditions. The experiments were carried out at Battelle Frankfurt (West Germany), using a 640 m³ experimental facility which is 1:4 model of the containment of a West German pressurized water reactor, Biblis A. The tests were variations of a reference case, simulating the low-pressure scenario of a core melt accident with late containment failure.

A typical example of the results is reproduced in Fig. 15.17, showing the time history of the measured aerosol concentration in the model containment in two reference tests, as compared to predictions with NAUA and COCMEL (cf Fig. 15.13). The aerosol concentration decreases four orders of magnitude within less than 6 hours. While the agreement is relatively good in the range of large mass concentration, the experiments show that the time-integrated aerosol mass concentration is consistently overpredicted with NAUA.

LACE (LWR Aerosol Containment Experiment) is an international project, managed by EPRI. The experiments are carried out at the Hanford Engineering Development Laboratory in Washington. The containment is simulated by an 852 m³ steel tank. The aerosols are generated outside the tank and carried into the tank via a pipe system. Both the thermohydraulic

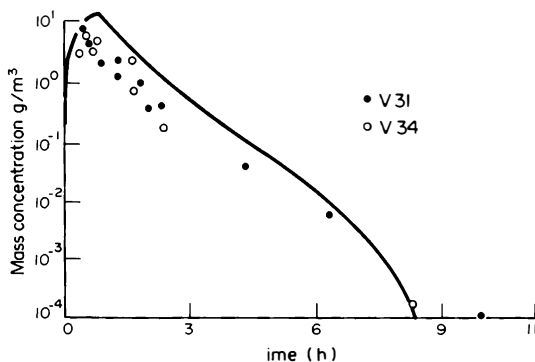


FIG. 15.17. Comparison of experimental and theoretical aerosol mass concentration as measured in DEMONA and calculated with NAUA. From J P Hosemann, K Hassmann, Methoden zur Quelltermbestimmung und experimentelle Absicherung, *Atomwirtschaft*, January 1987

conditions and the transport and deposition of aerosols in the pipe system and tank are studied. The main programme consists of six tests, focusing on three types of accident situations: containment bypass, failure to isolate the containment, and delayed containment failure.

The High Pressure Melt Streaming (HIPS) programme at Sandia National Laboratory, USA, aims at studying high pressure melt ejection. It has been found that the ejected melt is not a coherent stable stream but that the jet expands and breaks up. The ejection process is accompanied by significant aerosol generation. The expansion and break-up of the jet are attributed to the rapid evolution of the pressurizing gas dissolved in the melt (1551).

15.7.3 Assessment of uncertainties

Substantial improvement has been made in the modelling and experimental verification of severe accident processes since the Reactor Safety Study. Large uncertainties remain, however, and new issues have been uncovered. The state-of-the-art in 1987 is reviewed in the USNRC Reactor Risk Reference Document (1552), and the Swedish RAMA project (1553), for example. In this subsection, some of the remaining phenomenological uncertainties are briefly discussed.

In the simplistic severe accident codes, the core melt temperature is a user specified parameter, which has been shown to have a large influence on accident progression. In reality, there is a range of melting temperatures corresponding to the various core materials. There is uncertainty as to the formation of alloys and eutectics with lower melting points than those of the constituents. For example, the early meltdown of control rods could affect the accident progression by creating pathways for melt and steam flow. In addition, it could imply recriticality when the core is reflooded.

The simplistic codes assume that molten corium is collected in a bowl which blocks the core flow and subsequently fails by melt-through of the bottom of the bowl. In reality, the processes are probably much more complex, e.g. regarding the composition and possible stratification of the molten corium. This would affect the slumping of the core and the subsequent events. The uncertainty in core relocation is difficult to quantify.

Hydrogen is generated by metal–water reactions and during core–concrete interaction. There is some uncertainty in the amount of hydrogen produced and on the effects of hydrogen burn. Swedish studies have concluded that hydrogen burn is not a problem for Swedish BWRs due to their inerted containments. The Swedish PWRs, which have large air-filled containments, are predicted to be able to withstand hydrogen deflagration corresponding to oxidation of 85–170% of the zirconium in the core. Local hydrogen detonation does not cause containment failure. Global detonation is considered extremely unlikely.

Direct containment heating (cf 11.4.2) appears to be associated primarily

with pressurized water reactors. In order for direct heating to threaten containment integrity, the reactor coolant system must be at high pressure at the time the reactor vessel bottom is penetrated, a large fraction of the core must be molten in the lower plenum and ejected from the vessel, the molten core material must be aerosolized and dispersed throughout the containment, and the debris must transfer heat rapidly to the containment atmosphere. While the uncertainties are large, the Reactor Risk Reference Study (1552) could not dismiss the probability of early containment failure due to direct containment heating, or due to the combined effects of hydrogen burn and steam spikes.

Another area of uncertainty concerns the coolability of core debris in the reactor containment. According to Swedish design philosophy, the core melt will fall into water and form a coolable debris bed. The phenomena of melt fragmentation and steam explosion during melt–water interaction are insufficiently known, as is the heat transfer from molten debris to water covering the debris.

The detailed mechanistic modelling of physical and chemical phenomena has led to a greater predicted removal of aerosols in the reactor coolant system and containment than calculated in the Reactor Safety Study. The retention is more effective the longer the time to containment failure. There is, however, considerable uncertainty as to the efficiency of spray washing and pool scrubbing. Another key source of uncertainty is the revolatilization of iodine, cesium and tellurium from reactor coolant system surfaces.

Attempts to quantify the uncertainty in the prediction of source terms were made in the Reactor Risk Reference Study. For the volatile groups of radionuclides the uncertainty ranges are typically one to two orders of magnitude, and for the more refractory groups of radionuclides, two to three orders of magnitude.

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16

Secure Reactors

16.1 Safety Philosophy

The high level of safety in nuclear power plants has been achieved primarily by preventive measures to avoid operational disturbances and equipment malfunction. Whenever such events still occur, protective systems are provided to prevent incidents from developing into accidents. In terms of probabilistic safety analysis, it is a matter of reducing the frequency of initiating events and improving the reliability of the safety systems. The analysis of incidents and accidents and the feedback of experience provides the basis for safety improvement.

Development has led to a substantial raising of the safety level. For example, the estimated core-damage frequency is 4.1×10^{-6} per operating year for the pressurized water reactor Sizewell B under construction in England (1601). The corresponding value for the Surry-2 reactor, built 15 years earlier, was estimated at 5.6×10^{-5} per operating year in the Reactor Safety Study. The improvement has been mainly achieved through a higher degree of redundancy and diversification in the safety systems, e.g. more pumps for safety injection, greater diesel generator capacity, and a reserve control room, physically separated from the main control room.

At a level of 10^{-6} per year for the core damage frequency, the value of further risk-reducing measures is doubtful due to the diminishing returns and the uncertainties of the analysis. There is hardly any reasonable argument for attempting to achieve a lower core damage frequency. The USNRC has proposed the use of a safety performance guideline, which implies that the overall mean frequency of a large release be less than 10^{-6} per year of reactor operation (1602). It should be borne in mind that the release frequency is generally only a fraction of the core damage frequency, depending on the conditional probability of containment failure.

Even if the safety design and operation of today's reactors do meet very high standards, the calculated core damage frequency is by its very nature a probabilistic estimate, which means that there can be no absolute guarantee against the occurrence of a severe accident. The fundamental reason for this apparent paradox is that safety depends on the performance of mechanical and electrical systems and on human action. Experience has shown that very high reliability can be achieved, but that failure cannot be ruled out.

16.2 The PIUS Principle

In order to completely eliminate the possibility of core melting, safety must be based on inherent characteristics of the reactor system, independent of safety system performance and operator intervention. This is the basis of the PIUS (Process Inherent Ultimate Safety) principle, pioneered by Asea-Atom (1603).

The PIUS principle means that core safety is guaranteed by the laws of gravity and thermohydraulics alone. Core overheating will be avoided if the core is kept submerged and well-cooled, i.e. the core power must not exceed the cooling capability of the coolant. One way to ensure this is to have a sufficient amount of water constantly available to the core for decay heat removal by evaporation. The water must be available at operating pressure. It must contain a neutron poison, such as boric acid, which is capable of stopping the nuclear chain reaction.

A 2000 MWth (about 600 MWe) reactor is found to require at least 2000 m³ of water for 1 week of decay heat removal by evaporation. For this, a 10 MPa reactor vessel is needed with an internal volume of 3000–4000 m³, which can in practice only be built of prestressed concrete. In a design study (1603), the pressure vessel is given an inner diameter of 13.4 m, a height of 32.8 m and a wall thickness of 8–10 m. The vessel has a stainless steel liner. As an extra precaution against leakage, another leaktight steel barrier is embedded in the concrete. The vessel is free from penetrations except at the top. Therefore, there are no conceivable events which could lead to loss of coolant through the vessel walls. Loss of coolant can only occur when steam, carrying decay heat, is discharged through the valves in the upper part of the vessel. With the core located at the bottom of the vessel, the requirement that the core should be submerged is fulfilled.

Heat generation and cooling is illustrated in Fig. 16.1. To produce useful energy, hot water with a sufficiently low boron content must be pumped through the core to a heat exchanger (steam generator). The coolant circuit includes a riser and a pressurizer as well as interfaces (at A and B in the diagram) which connect the circuit with the surrounding pool water.

During normal operation, the circulation flow is adjusted so that the sum of the dynamic pressure loss from A to B and the static pressure difference in the lighter, hot reactor water is exactly the same as the static pressure difference in the denser, cold pool water. The interfaces at A and B are designed as density locks where the hot water forms a stagnant layer above the cold water. The lower hot/cold water interface at the lower density lock is controlled by temperature sensors, and the level is maintained by adjusting the speed of the recirculation pump.

This principle means that the mechanical energy supplied by the recirculation pump is used to keep the fluid system at a higher potential energy level than that of the equilibrium state. If the pump trips, the system will

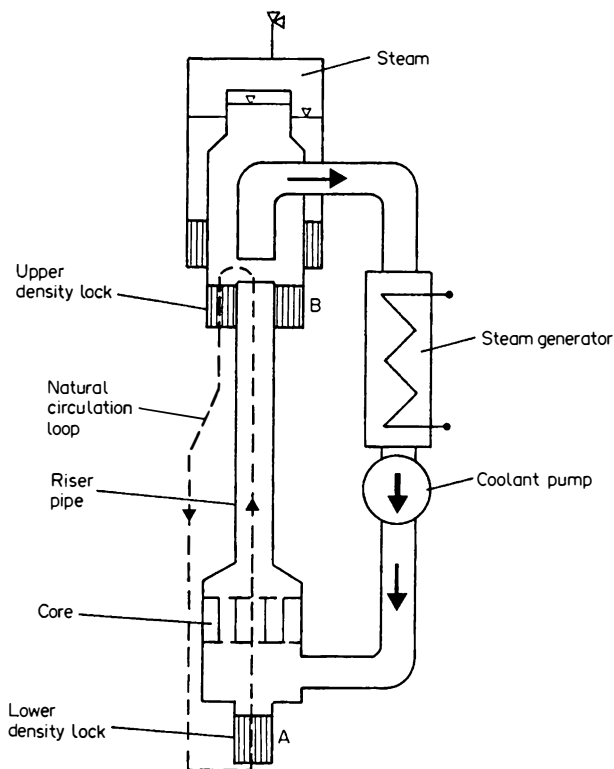


FIG. 16.1. The PIUS flow arrangement principle. From K Hannerz, *The SECURE reactors: Goals and Principles*, *Nucl. Europe*, October 1984

revert to its equilibrium state by the expulsion of hot water at B and the ingress of cold borated water at A, which will shut the reactor down. Further cooling is by natural circulation as shown in Fig. 16.1.

If the core generates more heat than can be removed by the steam generator, the reactor water will be heated to the boiling point. Steam will be produced in the core and the bubbles will move up through the riser pipe. The steam bubbles will further enhance the buoyancy of the coolant and increase the core flow. Since the recirculation pump has a limited capacity, the core flow will be partly drawn from the pool when it reaches a certain level, resulting in reactor shutdown.

Thus, the system is self-protecting in any conceivable abnormal situation. Rapid reactivity insertion is not possible, since there are no mechanical control rods. Reactivity is solely controlled by the boric acid concentration and the negative temperature coefficient of the reactor water.

16.3 SECURE-H

The PIUS principle was introduced in the mid-1970s in a joint Swedish-Finnish study project on using a nuclear power plant for district heating. The project was called SECURE (Safe Environmentally Clean Urban Reactor). SECURE-H nuclear power plants are now marketed by ABB Atom (formerly Asea-Atom) for district heating or for supplying heat to process industries using temperatures below 160°C (1604). A general layout of the SECURE-H heating reactor is shown in Fig. 16.2. The concrete reactor pressure vessel is placed with its closure at ground level. The coolant pumps and heat exchangers are located outside the reactor vessel together with a blowdown chamber containing a pressure suppression pool. The primary cooling system and the blowdown chamber are housed in a containment building below ground.

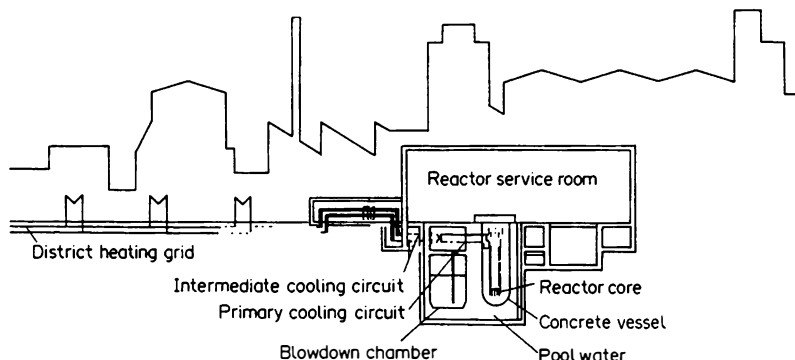


FIG. 16.2. The SECURE-H main cooling system. From C Pind, *The SECURE Heating Reactor*, *Nucl. Technol.* Vol 79, November 1987

The primary cooling system delivers heat from the core through the primary heat exchanger to an intermediate cooling system, connected to the district heating grid through a secondary heat exchanger. The intermediate cooling system operates at a higher pressure than the primary system, and its water has a high content of boron. In this way a tube rupture in the primary heat exchanger results in a flow of highly borated water into the primary system.

The main coolant lines are provided with venturi flow limiters at the reactor pressure vessel pipe penetrations. They act as safeguards against low pressure in the vessel or high coolant temperature. In these cases the coolant will boil in the throat section of the venturi flow limiters, which increases the pressure drop and reduces the flow in the primary coolant circuit. This results in the ingress of highly borated pool water and a reduction of the reactor power.

The main plant data are presented in Table 16.1. As can be seen, the core power ratings are low which gives very good margins against fuel failure during normal and transient conditions. The pressure drop over the core is only 0.011 MPa, since the PIUS principle implies that the core pressure drop be equal to the pressure difference caused by the density difference between pool water and riser water.

16.4 SECURE-P

Several electricity generating versions of SECURE, called SECURE-P, have been studied for the unit output range of 400–800 MW_{el} (1603). In the final choice, a modular design was selected, where each module constitutes a complete steam generating system with an integrated core, steam generator and coolant pump. The modules supply steam to a common turbine and can be operated independently. One to four modules can be placed in a concrete reactor pressure vessel. Main data for a three-module unit are shown in Table 16.2.

TABLE 16.1 *SECURE-H main data*

Thermal output	MW _{th}	400
Fuel power density	W/g U	15.0
Number of fuel assemblies	—	308
Number of fuel rod positions per assembly		8 × 8
Active core height	m	1.845
Equivalent core diameter	m	2.51
Core flow	kg/s	2300
Primary system operating pressure	MPa	2.0
Coolant inlet temperature	°C	150
Coolant outlet temperature	°C	190
Inner diameter of concrete RPV	m	9.5

Source: C Pind, The SECURE Heating Reactor, *Nucl. Technol.* Vol 79, November 1987

TABLE 16.2. *Main data for SECURE-P*

Thermal output	MW _{th}	2000
Electrical output	MW _{el}	625
Number of fuel assemblies	—	213
Number of rod positions per assembly	—	16 × 16
Active core height	m	1.97
Equivalent core diameter	m	4.03
Primary system operating pressure	MPa	9.0
Coolant inlet temperature	°C	261
Coolant outlet temperature	°C	293
Inner diameter of concrete RPV	m	13.4

Source: C Pind, *loc. cit.*

From a nuclear point of view, SECURE-P is a pressurized water reactor with moderate performance data. The reactor pressure is lower than that of a conventional PWR which results in a certain loss of efficiency. Further technology development will mainly focus on the steam generator, which is of a new design, and on the qualification of the hot/cold interfaces and the thermal insulation of the primary system against the pool. A large-scale integral experiment has demonstrated the thermohydraulics of the system during abnormal events (1604).

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